August 12, 1982



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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| In the Matter of | ) | |
| | ) | |
| METROPOLITAN EDISON COMPANY | ) | Docket No. 50-289 |
| | ) | (Restart) |
| (Three Mile Island Nuclear | ) | |
| Station, Unit No. 1) | ) | |

LICENSEE'S RESPONSE TO THE

ATOMIC SAFETY AND LICENSING APPEAL BOARD'S

ORDER OF JULY 14, 1982



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United States of America
Nuclear Regulatory Commission

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| In the Matter of<br><br>METROPOLITAN EDISON COMPANY<br><br>(Three Mile Island Nuclear Station, Unit No. 1) | Docket No. 50-289<br>(Restart) |

## AFFIDAVIT OF ROBERT W. KEATEN

County of Morris )
)
State of New Jersey )

ROBERT W. KEATEN, being duly sworn according to law, deposes and states as follows:

1. I am Directorof the Systems Engineering Department of GPU Nuclear Corporation and have presented testimony before the Atomic Safety and Licensing Board in this proceeding on several occasions. A statement of my professional qualifications is set forth in the evidentiary record of this proceeding following Tr. 4588.

2. The information provided in Licensee's responses, dated August 12, 1982, to the Atomic Safety and Licensing Appeal Board's questions contained in its July 14, 1982 Order was prepared by me or under my supervision by employees of GPU Nuclear Corporation and is true and accurate to the best of my knowledge and belief.

Robert W. Keaten
ROBERT W. KEATEN

Subscribed to and sworn before me this 12th day of August, 1982.

Alice J. House
ALICE J. HOUSE
NOTARY PUBLIC OF NEW JERSEY

Question I. Update of Restart Requirements

The Appeal Board requested that Licensee provide a report on the status of certain of the restart modifications listed in Appendix A to the Board's July 14, 1982 Order. The attached chart presents the current status of these modifications.

## I. UPDATE OF RESTART REQUIREMENTS

| ITEM | PERCENT COMPLETE | ESTIMATED COMPLETION DATE | NOTE |
|---|---|---|---|
| Short Term | | | |
| Order Item 1 EFW Reliability | | | |
| 1a-3 Auto EFW Load to Diesels | 100% 1/ | | |
| 1a-4 EFW Technical Specification | 100% | | TSCR 103 (5/18/81), Rev. 1 (11/13/81), Rev. 2 (5/20/82) pending NRC approval. |
| 1a Additional Items | | | |
| 1 CWST Level Alarm | 95% | 9/1/82 | Need to replace level gauge. |
| 6 EFW Initiation Independent of AC (OTSG Level Indication independent of ICS) | 100% 2/ | | |
| 7 EFW Operability in Steam Environment | 10% | Cycle 6 Startup | Long Term - EFW Safety-Grade Modification |
| 8 Cross-Tie Break | 100% | | Weld inspection. Results to be sent to NRC. |
| Order Item 2 IE Bulletins | | | |
| 79-05B-3 PORV Set Point | 100% | | |
| 79-05B-5 Anticipatory Reactor Trip (Safety Grade) | 100% | | |

| ITEM | PERCENT COMPLETE | ESTIMATED COMPLETION DATE | NOTE |
|---|---|---|---|
| Order Item 4 Separation of TMI-1 & 2 | | | |
| (a) Liquid Radwaste | 100% | | |
| (b) Gaseous Radwaste | 100% 2/ | | |
| (c) Solid Radwaste | 100% | | |
| (d) Sampling System | 100% | | |
| Order Item 8 Lessons Learned - Short Term | | | |
| 2.1.1 Emergency Power Supply | | | |
| - Pressurizer Heaters | 100% 2/ | | |
| 2.1.3a Valve Position Indication | 99% | 9/1/82 | |
| 2.1.3b Inadequate Core Cooling | | | |
| - Existing Instrumentation & Saturation Meter | 100% | | |
| 2.1.4 Containment Isolation | 80% | 11/1/82 | |
| 2.1.5c Install Recombiner | 99% | 9/1/82 | |
| 2.1.8c Iodine Instrumentation | 60% | 10/1/82 | |
| Long Term (LT) (NUREG-0737 numbers) | | | |
| LT-1 (II.K.2.9) ICS FMEA Modifications | 98% 1/ 2/ | 9/1/82 | |

| ITEM | PERCENT COMPLETE | ESTIMATED COMPLETION DATE | NOTE |
|---|---|---|---|
| LT-3 Lessons Learned Category B from NUREG-0578<br>2.1.3b (II.F.2.3) ICC Instrumentation<br>-Backup Incore Thermocouples (safety grade) | 75% 1/ Modification / Env. Qual. 10% | 10/1/82 / (see note) | Environmental qualification of this modification will be accomplished in accordance with the Final Rule. |
| 2.1.5a (II.E.4.1) Dedicated $H_2$ Penetrations<br>- Install | 100% | | |
| 2.1.6b (II.B.2) Plant Shielding<br>- Plant Modifications | 100% MCC / DHRS 30% | / Cycle 6 Startup | MCC: Motor Control Center<br>DHRS: Decay Heat Removal System |
| 2.1.7a (II.E.1.2) EFW Auto Initiation<br>- Safety Grade | 100% | | |
| 2.1.7b (II.E.1.2) EFW Flow Indication<br>- Safety Grade | 100% 1/ 2/ | | |
| 2.1.8a (II.B.3) Post-Accident Sampling<br>- Modifications (long-term Category B) | 75% | 12/1/82 | Short-term (Category A) modifications complete |
| 2.1.8b (II.F.1) Radiation Monitors<br>- Effluent Monitors | 75% | 1/1/83 | Monitors have been sent to Batelle for calibration |
| - Iodine/Particulate Monitors | 75% | 10/1/82 | |
| Additional Items<br>#1 (II.F.1) Containment Pressure | 90% | 10/1/82 | |
| #2 (II.F.1) Containment Water Level (safety-grade) | 95% | 10/1/82 | control grade complete |
| #3 (II.F.1) Containment Hydrogen | 70% | 12/1/82 | |

| ITEM | PERCENT COMPLETE | ESTIMATED COMPLETION DATE | NOTE |
|---|---|---|---|
| LT-4 Emergency Preparedness<br>Emergency Communications<br>- Install control room emergency telephone | 100% | | |
| - Connect emergency telephone equipment to vital power | 100% | | |
| Emergency Facilities<br>- Install high radiation monitoring alarm system | 90% | 9/1/82 | |
| Board Imposed Requirements (December 14, 1981 PID)<br>Plant Design, Modification and Procedures Findings<br>II.E. Pressurizer Heaters<br>- Demonstrate RCS pressure control w/HPI | 100% | | Test data to be forwarded to NRC by 9/1/82 |
| II.K Computer<br>- Incore thermocouple backup display (not safety grade) | 75% | 10/1/82 | |
| II.M Safety System Status Panel<br>- System Status Administrative Controls | 100% | | |

| ITEM | PERCENT COMPLETE | ESTIMATED COMPLETION DATE | NOTE |
|---|---|---|---|
| II.N Control Room Design<br>- Correct NUREG-0752 deficiencies | short-term 97%<br>long-term 10% | 10/1/82<br>Cycle 6 Startup | |
| II.P Systems Classification<br>- Upgrade Pressurizer Level Instrument Power Supplies | 95% | 9/1/82 | |
| II.Q EFW Reliability (see detailed question on long-term order Item B.2.1.7a)<br>- Safety grade automatic EFW control | 10% | Cycle 6 Startup | |
| - Install following long-term EFW modifications<br>(a) EFW cavitating venturis | 100% 1/ | | |
| (b) CWST level alarm (safety grade) | 10% | Cycle 6 Startup | |
| (c) OTSG high level alarm | 10% | Cycle 6 Startup | |
| (d) Safety grade isolation of MFW on OTSG overfill | 0% | Cycle 6 Startup | |
| (e) Upgrade main steam rupture detection system to safety grade | 0% | Cycle 6 Startup | |

NOTES - 1/ Construction Complete awaiting plant acceptance

2/ Construction Complete awaiting testing during hot functional testing or power escalation testing.

Question II.A: In letters dated April 22, 1982 and May 13, 1982, the licensee notified this Board that certain steam and water tests exhibited valve instability that resulted in damage to the safety relief valve. Throughout the hearing, licensee maintained that the feed and bleed mode of forced core cooling relied upon these valves to provide a release pathway for excess coolant. In light of these tests results, how does the licensee plan to ensure that safety relief valves are capable of performing their function during feed and bleed when they may be called upon to open and close frequently with both steam and water flow mixtures?

LICENSEE RESPONSE

Prior to addressing the actions Licensee will take in response to the results of the EPRI valve testing program, Licensee believes that it would be appropriate to briefly review the situations in which the feed and bleed mode of core cooling might be utilized. Feed and bleed cooling is not required except when postulating events which are beyond the plant design basis, i.e., an extended loss of all main and emergency feedwater or certain accident conditions in conjunction with an extended loss of all feedwater. See Jones, ff. Tr. 4588, at 3; Tr. 5201 (Jones). Secondly, it should be noted that, while the analyses of feed and bleed cooling capability have assumed the use of the safety valves for the bleeding function, the PORV may be utilized to perform this function if it is available. Keaten and Jones, ff. Tr. 4588, at 7-8; Tr. 8761 (Jones).

The EPRI steam and water tests, referred to in our letters dated April 22, 1982 and May 13, 1982, in which the test valve exhibited instabilities, were performed on a long inlet (loop seal) configuration. This configuration is representative of the TMI-1 plant specific inlet piping. The EPRI test results for safety valves and subcooled fluid discharge have shown that the safety valves exhibited stable performance for all fluid inlet conditions when tested on a short inlet configuration. Based on these results and a review to ensure that the EPRI test conditions bound the TMI-1 specific requirements, Licensee believes that the TMI-1 safety valves will perform in a stable manner if they are on a short inlet. Therefore, Licensee is presently planning to modify, by restart, the inlet piping to eliminate the loop seal and move the valves into a short inlet configuration at the nozzles on the pressurizer.

Upon completion of these modifications, the safety valves will be capable of performing their function during the feed and bleed mode of core cooling when they may be called upon to open and close frequently with both steam and water flow mixtures.

Question II.B: The status list indicates that the installation of the Emergency Feedwater (EFW) automatic initiation is completed as control grade equipment (Item A.8.2.1.7a) but that further modifications up to safety grade will be partially completed by August 1982, and a footnote indicates that additional long term modifications are scheduled for the first refueling after restart. During the hearing, the staff testified that emergency feedwater modifications should be completed by late 1982 (Ross, Tr. 15,577).

1. Which, if any, of the modifications discussed in paragraphs 1028-1034 of the partial initial decisiom (PID) LBP-81-59, 14 NRC 1211 (1981), will not be completed before restart?

2. What are the reasons for the delay beyond the completion date estimated by the staff during the hearing?

LICENSEE RESPONSE

The EFW modifications described in I.D., ¶¶ 1028-1034 are all short-term modifications undertaken in accordance with the terms of the Commission's August 9, 1979 Order and Notice of Hearing, CLI-79-8, 10 N.R.C. 141, 144. *See* Staff Ex. 1 at C1-1 through C1-12 and C8-34 through C8-40. Each of these modifications, described below, will be fully implemented prior to restart.

1. Safety-grade, automatic initiation of EFW on loss of all four reactor coolant pumps (4 RCPS) or loss of both feedwater pumps (2 FWPS) has been installed.
2. Redundant, safety-grade flow indicators for EFW flow to each steam generator have been installed.
3. The EFW flow control valves have been modified to fail open on loss of instrument air.

4. Operator control of EFW flow to each steam generator independant of the ICS has been provided.

5. A redundant two hour air supply to furnish instrument air to the EFW control valves and related systems has been installed.

6. Alarms, signifying a 20 minute supply of water remaining in the condensate storage tanks, have been provided.

7. Redundant, safety-grade steam generator level indication, used in conjunction with item 4 above, has been provided in the control room.

In addition to the short-term modifications discussed above there are certain long-term EFW modifications associated with Item II.E.1.1 of NUREG-0737 which are being undertaken. I.D., ¶¶ 1037-1038, 14 N.R.C. 1211, 1364. These are the modifications referred to by Dr. Ross at the transcript page cited by the Appeal Board. As Dr. Ross testified, the NUREG-0737 implementation date for these long-term modifications was January 1, 1982, but it was thought that procurement and design problems might result in a delay in implementing certain of the design modifications until the Cycle 6 refueling outage (i.e., approximately 1 year after restart). Tr. 15,577 (Ross); see also I.D., ¶ 1038, 14 N.R.C. 1211, 1364. The current status of the long-term modifications is set forth below:

1. Cavitating venturis, one per steam generator, have been installed.

2. Safety-grade low level alarms with the same setpoint as short-term item 6 above will be installed during the cycle 6 refueling outage.

3. Safety-grade steam generator high level alarms will be installed during the cycle 6 refueling outage.

4. Safety-grade isolation of main feedwater on overfill of a steam generator (hi-hi level in downcomer) will be installed during cycle 6 refueling outage.

5. The main steam rupture detection system will be upgraded to safety-grade during the cycle 6 refueling outage.

6. An additional safety-grade signal, based upon steam generator low-low level, will be provided for EFW initiation.

In conjunction with these six long-term modifications, as noted by the Appeal Board in Question II.B., Licensee will further upgrade the EFW system by providing safety-grade automatic control of EFW flow to the steam generators. It is this long-term modification which is referred to by the Staff in footnote 3, p. 6 of Appendix B to the Appeal Board's Order of July 14, 1982. To clarify, the TMI-1 EFW system at restart will have safety-grade automatic initiation (i.e., automatic starting of the EFW pumps) as described in snort-term item 1 above, but will not have safety-grade automatic control. See I.D., ¶¶ 1036, 14 N.R.C. 1211, 1363. Redundant, safety-grade automatic control of EFW to each steam generator, based upon steam generator level, will be installed during the cycle 6 refueling outage.

At the time that testimony was presented on the TMI-1 EFW system, it was thought that restart would occur in late 1981, and that most of the long-term modifications could be accomplished during the cycle 6 refueling outage, then scheduled for late 1982. However, it must be realized that the provision of safety-grade automatic EFW flow control and long-term modifications 3, 4 and 5 above required the design and procurement of an entirely new four channel safety-grade system. The design engineering for this system required the performance of additional analyses beyond those originally projected, thereby resulting in a delay in the original implementation schedule.1/ Further delays have been created by the long lead time for delivery of properly qualified hardware. In view of the time and labor required for installation, the

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1/ The additional engineering analyses were required due to unanticipated complexities inherent in attempting to integrate the new system with existing plant systems, i.e., assuring that there are no unacceptable interactions with existing non-safety-grade systems and resolving human factors considerations with respect to consistency of displays. Additionally, engineering work on the long-term modifications was delayed approximately six months by the need to concentrate engineering resources on resolving the TMI-1 steam generator problems.

modifications will require an extended outage and will therefore not be completely implemented until the cycle 6 refueling.

Question II.E: During the hearing, the licensee indicated that the high point vents were planned to be installed prior to restart (Tr. 16, 580). NUREG-0737 requires the installation to be complete by July 1, 1982. The status list indicates that the completion date is "to be determined." What progress has been made in complying with the requirements of NUREG-0737 for the installation of high point vents? Are the vents and their controls fully safety-grade? If the high point vents will not be installed prior to restart, what is the justification for allowing operation TMI-1 before the vents are installed?

LICENSEE RESPONSE

Licensee's system for providing the capability to vent noncondensible gases from the Reactor Coolant System (RCS) is described in Section 2.1.2.2 of the Restart Report (Lic. Ex. 1). The RCS Venting System will consist of three separate sub-systems: vents from the top of the pressurizer, discharging to the Reactor Coolant Drain Tank (RCDT); and vents from the top of both hot legs and from the top of the Reactor Vessel Head, which will discharge directly to the containment. The pressurizer vent has been installed and will be operable at restart. The design of the balance of the RCS venting system has progressed through the production of flow diagrams, piping drawings, pipe support drawings and electrical and instrumentation details. The entire RCS venting system will be safety-grade. See Lic. Ex. 1, §§ 2.1.2.2.1, 2.1.2.2.6.

The schedule for implementation of this modification as set forth in NUREG-0737 has been superceded by a recent revision2/ to 10 C.F.R. § 50.44(c)(3)(iii), which requires

2/ See Final Rule, Interim Requirements Related to Hydrogen Control, 46 Fed. Reg. 58,484 (Dec. 2, 1981).

installation of these vents "by the end of the first scheduled outage beginning after July 1, 1982 and of sufficient duration to permit required modifications..." (emphasis added). In accordance with this requirement, Licensee plans to install the balance of the RCS venting system during the first refueling outage following restart (i.e., the cycle 6 refueling outage).3/

The Appeal Board has also requested a justification for allowing operation of TMI-1 prior to the installation of the high point vents. In that the TMI-1 vents will be installed in accord with the schedule for all operating reactors, Licensee does not believe there is a need to provide special justification for permitting TMI-1 to restart. However, it should be noted that the high point vents are solely a back-up which will be provided to mitigate a beyond design basis event -- the generation of noncondensible gases -- which is not expected to occur in the future. Tr. 4991-93 (Jensen).

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3/ Licensee notes that the installation of the high point vents was never a pre-restart commitment, although installation by restart was previously thought to be possible. See Tr. 16,580 (Keaten).

Question III.B: In Paragraph 771 of its PID, the Licensing Board directed the staff to verify that procedures to connect the pressurizer heaters to the diesels include provisions to assure that the heaters would not be reconnected to onsite power until stabilization of the event that caused their disconnection. The status list attached to SECY-82-250 indicates that this item is complete. What provisions have been included in the procedures to comply with the Licensing Board's direction?

LICENSEE RESPONSE

In Paragraph 771 of its PID the Licensing Board directed the Staff to "verify that the plant procedures include provisions to assure that desired pressurizer heater loads will not be reconnected to the on-site power supply after they have been automatically separated until stablization has been achieved following the event that caused their disconnection." (Emphasis supplied.) Licensee understands the Board's direction to refer to the stabilization of electric supply to all systems connected to the diesel generator following the event which caused the disconnectionof the pressurizer heater load, rather than stabilization of the event itself. Thus, a small break LOCA could result in an ES signal which would automatically disconnect the pressurizer heaters as well as actuate the emergency core cooling systems. Stabilization of the LOCA event itself could require a substantial period of time. Stabilization of the electric power supply to the emergency core cooling systems or other connected loads would normally occur in a much shorter interval. In other situations, however, stabilization of the event which caused the disconnection could be synonomous with stabilization of electric power supply. Thus a fault in a pressurizer heater could cause both overcurrent and undervoltage, either of which would automatically result in disconnection of the power supply. Maintenance of a stable power supply could not be accomplished upon reconnection without correction of the fault condition. In this situation therefore Licensee's procedures call for a full evaluation of the cause of disconnection.

A "caution" has been added to Revision 17 of Emergency Procedure 1202-29, "Pressurizer System Failure",4/ which requires evaluating the cause of the pressurizer heater trip and verifying stabilization of electric supply to all systems connected to the diesel generator prior to the reconnection of the heaters. The procedure caution, which is applicable when the diesel generators are supplying plant load, is set forth verbatim below:

> CAUTION: Should the pressurizer heaters be tripped out as a result of an ES signal, overload or undervoltage condition, they are not to be reconnected until the cause of the trip has been fully evaluated and stabilization has been achieved following the event. Stabilization shall be considered to be achieved when block loading is completed, voltage is at its normal value and the load on the diesel does not exceed 2850KW.

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4/ An earlier version of this procedure (Revision 15) was admitted as Licensee Ex. 50. The new caution has been added following Step L at page 12.1 of Licensee Ex. 50.

Question III.C: PID Paragraph 943 listed measures that have been or will be taken at TMI-1 to improve protection against small break LOCAs. One of those measures was the improvement of the HPI systems by adding cavitating venturis and cross-connection lines. It was also stated that the system being installed will automatically perform the balancing of HPI flow. How is this to be accomplished and what is the completion status of these HPI modifications?

LICENSEE RESPONSE

The HPI System modification, adding cavitating venturis and cross-connection lines, has been completed. Prior to restart, testing will be performed to demonstrate system performance. Tr. 5605 (Jensen); Lic. Ex. 1, Supp. 1, Response to Question 36c.

A complete description of the HPI system modifications and performance evaluation is contained in the Restart Report (Lic. Ex. 1) at Section 3; Supplement 1, Part 1, Questions 36b, 36c and 37; and, Supplement 1, Part 3, Questions 1, 2 and 3. The modifications provide for assuring adequate HPI flow in the event of either a break in the HPI lines or in the event of a makeup value failing to open. In the first case, the installed flow-limiting devices (the cavitating venturis) will limit the amount of coolant injected into the broken line, thereby limiting the fluid discharged out of the break. In the event that one of the valves supplying HPI fluid (the MUV-16 valves) fails to open, the cross-connect devices will function and direct HPI flow to all four HPI lines. Prior to these modifications, the control room operators were required to manually limit or redirect the HPI flow. See Jensen, ff. Tr. 5496, at 7; Tr. 5605 (Jensen).

Question III.D: In Paragraph 1064 of its PID, the Licensing Board directed the staff to certify to the Commission that the licensee has made reasonable progress in initiating a program for long-term solution of the steam generator bypass logic problem. What progress has been made by the licensee in solving this problem? What interim methods will be used to ensure that plant operators are aware of the problem and the actions to be taken in the event of isolation of both steam generators?

LICENSEE RESPONSE

In order to eliminate the concern raised by the Licensing Board in I.D., ¶¶ 1060-1064 (i.e., isolation of all feedwater flow to both steam generators), Licensee has proposed implementing two changes to the Main Steamline Rupture Detection System (MSLRDS). The proposed changes consist of the addition of cavitating venturis to the EFW lines and the deletion of the MSLRDS signal to the EFW system. The proposed design changes were submitted for Staff approval by letter dated August 2, 1982, from H. D. Hukill to John F. Stoltz. This letter, which includes a safety evaluation of the proposed change, is attached hereto as Attachment A. Licensee anticipates implementing this design change prior to restart, subject to review and approval by the Staff.

Question III.I: In a footnote to Paragraph 919 of the PID, the Licensing Board indicated that the licensee planned to perform an in-plant communications study in 1981. What is the status of that study? If completed, please briefly summarize results and present status of implementation.

LICENSEE RESPONSE

As noted by the Licensing Board, Licensee planned to begin an in-plant communications study in 1981 and to complete this study in 1982. I.D., ¶ 919, n. 109. The performance of this study, however, received a lower priority than many of the other actions being taken by Licensee prior to restart and some slippage in schedule has occurred. Proposals to perform a communications study at TMI-1 were solicited from four consulting firms in April, 1982. Three firms responded in June of this year and their proposals are currently being reviewed by Licensee. Selection of the consultant, awarding the contract and commencement of the study should be completed in the fall of 1982. The study is expected to take six to nine months.

Question III.K: The Licensing Board indicated in PID Paragraph 1264 that a tunnel-like barrier for personnel passage between the Unit 1 control tower and the Unit 1 auxiliary building will be completed before restart. What progress has been made in completing this modification?

LICENSEE RESPONSE

The commitment in I.D., ¶ 1264 to construct a tunnel like barrier which will provide personnel passage between the Unit 1 control tower and the Unit 1 auxiliary building, and which will also form part of a barrier that will seal the open areas between the Unit 1 auxiliary building and the Unit 1 fuel handling building has been completed. Licensee is currently in the process of designing a program to test the adequacy of its phase I ventilation separation program, which will be submitted to the Staff for approval. This program will include a test of the adequacy of this barrier.

Question IV.B: In the event that the pressurizer heaters fail to operate while the plant is operating at full power,

(1) how much time would it take to achieve RHR system initiation conditions and then cold shutdown?

(2) how would pressure control be performed during cooldown to conditions allowing RHR system operation?

(3) how soon after shutdown from full power conditions does the RHR system have sufficient decay heat removal capability?

LICENSEE RESPONSE

(1) Based upon data concerning several actual TMI-1 shutdown and cooldown events, Decay Heat Removal (DHR) System5/ initiation will occur between 8 and 12 hours following initiation of the shutdown. Cold shutdown conditions (less than 200°F) are normally achieved in an additional two hours. However, in the event of a serious plant casualty, a controlled shutdown/cooldown to less than 200°F could be achieved in approximately 5 hours.6/

(2) The TMI-1 normal cooldown procedure requires that the pressurizer heaters be turned off; therefore, failure of the pressurizer heaters would not adversely impact a normal plant cooldown. RCS pressure control during a normal plant cooldown is achieved by use of the pressurizer sprays, to reduce pressure as necessary.

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5/ The Appeal Board question refers to the RHR (residual heat removal) system. This system is designated as the Decay Heat Removal System at TMI-1.

6/ The DHR System can be actuated at approximately 250°F and 320 psi. *See* Tr. 16,556 (Colitz).

(3) The DHR System has sufficient capacity to remove 100% of the decay heat immediately following a controlled plant shutdown. The DHR System is not capable of removing the maximum decay heat present at the instant following a reactor trip. Following a reactor trip, there is an interval of approximately 160 seconds before the core decay heat level drops to the DHR System capacity. During this time, decay heat removal is accomplished by the use of other plant systems (i.e., steam generators, HPI, LPI). See also Keaten et al., ff. Tr. 16,552, at 6, 9.

Question IV.D: What is the extent of the environmental qualification of the PORV block valve and its controls?

LICENSEE RESPONSE

The PORV block valve operator (Limitorque series SMB-00) and the operator motor are qualified for a LOCA environment of 100% humidity, $2 \times 10^8$ Rads (gamma integrated dose), and 90 psig.7/ See also Tr. 8800-8801, 8994-8998 (Urquhart, Correa). Similarly, the Class 1E power supply and control subcomponents are qualified to survive the adverse environments associated with a LOCA, feedwater line break or main steamline break.

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7/ The pressure and temperature parameters decrease over time: 329°F, 40 psig for hours 3 to 5; 272°F, 20 psig for hours 5 to 24; 251°F, 17 psig for days 2 through 6.

Question IV.F: Describe the method for cooling the plant to RHR initiation conditions by feed and bleed cooling using only safety-grade equipment.

LICENSEE RESPONSE

"Feed and bleed cooling" refers to the process in which (1) water is added to the reactor coolant system (RCS) to maintain sufficient liquid inventory to cool the core, and (2) steam, water, or a two phase mixture is released from the RCS to maintain RCS pressure within design limits. The process adequately cools the core and prevents RCS overpressure when the energy removal rate exceeds the core decay heat level. The equipment used to add water in this mode is the high pressure injection pumps, piping, valves and associated circuitry. It is fully safety-grade and is capable of supplying water at an adequate rate to maintain core cooling through the pressure range of interest. Fully safety-grade equipment can be used to maintain the "bleed" cooling at high system pressure. This is accomplished with the pressurizer code safety valves relieving steam, water, or a two phase mixture to the reactor containment building. Feed and bleed cooling could be maintained indefinitely in this mode by recirculation of water from the reactor containment building sump through safety-grade support systems. See generally Keaten and Jones, ff. Tr. 4588, at 7-8, 11-12.

Feed and bleed cooling could be used to cool the plant while depressurizing it to the conditions required for initiation of the decay heat removal system (equivalent to RHR) by using the power operated relief valve (PORV) mounted on the pressurizer. Tr. 16,575 (M. Ross). This operation is covered by the TMI-1 emergency procedures. The PORV, however, although fulfilling some requirements of safety-grade equipment, is not fully safety-grade. Id. (Keaten); see also Correa et al., ff. Tr. 8746, at 7-8.

When decay heat levels are sufficiently low, the newly installed pressurizer vent line could perform the "bleed" function in the same fashion as the PORV. Tr. 16,575-76 (Keaten). This vent path meets the safety-grade criteria identified in NUREG-0737, Item II.B.1, High Point Vents.

Question IV.I: During RHR system operation, how is overpressure protection provided?

LICENSEE RESPONSE

Overpressure protection of the reactor coolant system (RCS) during DHR System operation is provided by plant design, operating procedures and Technical Specification 3.1.12 requiring PORV operability. The plant operating procedures and Technical Specifications require that sources of pressure that could cause an overpressure condition be disabled or physically isolated from the RCS during DHR System operation. Further, operation of the PORV8/ to relieve pressure will protect the RCS from overpressure conditions during DHR System operation. See also Tr. 8756 (Jones).

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8/ The setpoint for PORV actuation is required, by Technical Specification 3.1.12, to be lowered to 485 psig when system temperature is below 275°F.

JG 04 1982

GPU Nuclear

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August 2, 1982
5211-82-153

Office of Nuclear Reactor Regulation
Attn: John F. Stolz
Operating Reactors Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Main Steamline Rupture Detection System Design Changes

In its Partial Initial Decision (PID) on design (See PID 1060-1064) the Atomic Safety and Licensing Board (ASLB) required that GPUN investigate design changes to the Main Steamline Rupture Detection System (MSLRDS). The changes are to prevent unnecessary isolation of feedwater under single failure conditions. A description and evaluation of the changes to the MSLRDS is attached. This is submitted for NRC approval as requested by the ASLB (PID 1064).

Sincerely,

H. D. Hukill

H. D. Hukill
Director, TMI-1

HDH:CWS:vjf

Attachment

cc: R. C. Haynes
R. Jacobs

"ATTACHMENT A"

ATTACHMENT 1

Main Steamline Rupture Detection System Design Changes

## I. INTRODUCTION

The Main Steamline Rupture Detection System (MSLRDS) is actuated on low steam generator pressure (below approximately 600 psig) and automatically closes the Emergency Feedwater (EFW) and Main Feedwater (MFW) control valves to isolate feed flow to the depressurized steam generator. If subsequently pressure rises above 600 psig in a steam generator the EFW associated with that steam generator is restored. This MSLRDS action prevents overpressurization of containment from steamline breaks in containment. The ASLB was concerned that the MSLRDS would block all feedwater, including EFW, to the steam generators in certain scenarios when it should not be blocked.

## II. SOLUTION

The proposed solution to the above concern consists of the addition of cavitating venturis and the deletion of the MSLRDS signal to the Emergency Feedwater System. Low OTSG pressure, which actuates the MSLRDS, can result from either a severe overcooling or a main steamline break event. The original design required operator action to bypass MSLRDS to prevent a loss of heat sink if a low OTSG pressure condition developed and single failure then blocked EFW. The addition of the cavitating venturis to the EFW System and removal of the MSLRDS from the EFW valves eliminates operator action to provide EFW to the intact OTSG in the event of a single failure. Since the venturis also limit EFW flow, the MSLRDS is no longer required for EFW and need not be up graded to safety grade (PID 1037e) since it is eliminated.

## III. SAFETY EVALUATION

Deletion of the MSLRDS from the EFW valves does not affect any of the FSAR acceptance criteria. The basis for this judgment is as follows:

The MSLRDS was installed to prevent overpressurization of the containment due to a Main Streamline Break (MSLB). Removal of the MSLRDS from the EFW valves will make TMI-1 feedwater isolation functionally the same as TMI-2 in its response to a MSLB. The TMI-2 MSLB analysis was reviewed and approved by the NRC (See TMI-2 FSAR, Chap. 15, Appendix B). The TMI-2 analysis is bounding for TMI-1 for the following reasons:

a) The TMI-1 venturis limit total flow to a lower flow rate than the TMI-2 venturis (1150 GPM vs. 1250 GPM), and

b) TMI-1 cannot have a double OTSG blowdown in containment (limiting pressurization accident for TMI-2) because the main steam isolation valves are stop check valves for TMI-1.

Deletion of the MSLRDS from the EFW valves does not increase the probability of occurrence of a steamline break accident. The consequences of the accident, as analyzed in the TMI-2 FSAR, have not been increased, Reactor Building overpressurization does not occur and the required heat removal capability to prevent fuel damage is provided. Specifically, fuel damage will not result, off-site doses will not be increased, and steam generator tube integrity will not be compromised. The conclusions are confirmed in the Restart Repor Section 8.3.9 which references the TMI-2 FSAR, Chapter 15, Appendix B. EFW f is continued throughout the referenced analysis. Addition of cavitating ventur_ to the EFW system limits the maximum EFW flow at TMI-1 and assures that the referenced TMI-2 analysis is bounding for TMI-1. Furthermore, the systems, setpoints and/or plant conditions that are utilized in the referenced analysis are applicable to both TMI-1 and TMI-2. (The NRC was also advised of the TMI-1 design modification in Met-Ed response to IE Bulletin 80-04 May 9, 1980 TLL 228).

The referenced TMI-2 analysis assumed 1%ΔK/K shutdown margin and demonstrated that the core does not return to criticality and that the fuel rods do not violate a DNBR of 1.0. Other assumptions made in the referenced analysis are more severe than those allowed by TMI-1 Tech. Specs., most notably power level (2772 MW), and RCS flow (100%). The design peaking factor of 1.78 used in TMI-2 analysis exceeds the current design peaking factor for TMI-1. The referenced steamline break analysis also demonstrated acceptable offsite doses and showed that OTSG tube stresses resulting from the accident are acceptable. Tube stress conditions were evaluated in BAW-1588. The results of this evaluation bound the TMI-1 EFW system design with the MSLRDS signal deleted from the EFW valves.

Other considerations and/or questions:

> Overfilling of the OTSG is an issue which has been raised and is documented in the Restart Report, Supplement 1, Part 2, Question 2. The analysis presented in the TMI-1 FSAR did not take credit for EFW isolation via the MSLRDS signal. The EFW flow rate assumed was 1500 GPM to one (1) OTSG at 600 PSIG (the MSLRDS set point), this assumed flow is 2-½ time the flow rate available to one (1) OTSG from the TMI-1 EFW system with cavitating venturis installed.
>
> Filling of the OTSG from the 50% operating range took 6.6 minutes using these assumptions. Therefore, the operator would have (with the venturis installed and a fully opened control valve) approximately 16 minutes to terminate an overfill condition due to EFW flow. The revised design therefore allows sufficient time for the operator to terminate EFW.

As discussed above, deletion of the MSLRDS signal to the EFW valves does not introduce any accident or malfunctions not previously evaluated, nor does it increase the likelihood of occurrence or consequences of any accident analyzed in the TMI-1 FSAR.

In conclusion, this modification does not introduce any accident or malfunctions not previously evaluated, nor does it increase the likelihood of occurrence or consequences of any accident as analyzed in the TMI-1 FSAR. No safety margins will be reduced as a result of the modification. Furthermore, the revised design improves the reliability of the EFW System to deliver flow to the intact OTSG and will not create a containment overpressurization or OTSG overfill condition.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

| | | |
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| In the Matter of | ) | |
| | ) | |
| METROPOLITAN EDISON COMPANY | ) | Docket No. 50-289 |
| | ) | (Restart) |
| (Three Mile Island Nuclear | ) | |
| Station, Unit No. 1) | ) | |

## CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Response to the Atomic Safety and Licensing Appeal Board's Order of July 14, 1982" were served this 12th day of August, 1982, by hand delivery upon the parties identified by one asterisk and by deposit in the U.S. mail, first class, postage prepaid, to the other parties on the attached Service List.

Thomas A. Baxter

Thomas A. Baxter, P.C.

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