GPU NUCLEAR

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May 26, 1999

1920-99-20140

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 281
Revised Seismic Verification Methodology for TMI-1 Auxiliary Steam System Piping

In accordance with 10 CFR 50.4(b)(1) and 10 CFR 50.59(c), please find enclosed License Amendment Request No. 281.

This license amendment request proposes a revision to the TMI-1 Updated Final Safety Analysis Report (UFSAR) Chapter 5 and 14 to permit the use of the Conservative Deterministic Failure Margin (CDFM) methodology for seismic analysis of the portions of the auxiliary steam line located in the Auxiliary, Control and Fuel Handling Buildings at TMI Unit 1. The revised seismic evaluation method for the auxiliary steam system piping utilizes the CDFM methodology developed by Electric Power Research Institute. The CDFM methodology demonstrates that the auxiliary steam system piping will maintain integrity during a seismic event and thus prevent adverse impact on safety related equipment.

A001

The CDFM methodology is an analytical technique developed to predict the behavior of pipe and pipe supports when subjected to a seismic event. The methodology was developed by a group of experts in the field of seismic analysis based on observations from strong motion earthquake sites. Piping systems which meet the requirements of the CDFM methodology have performed well and maintained integrity during earthquakes much larger than the TMI-1 SSE. In addition, the CDFM methodology has been used at a number of nuclear plants (e.g., Edwin I Hatch Nuclear Plant-Unit 2, Limerick Generating Station-Unit 2 and Duane Arnold Energy Center) to analyze piping systems to address the main steam isolation valve leakage rate and control room habitability issues. These plants have received safety evaluation reports (SERs) from the US

Nuclear Regulatory Commission for use of the CDFM methodology. These SERs conclude that with the available safety margins demonstrated by the results of the analysis, the employment of the CDFM methodology to demonstrate the functional capability of the system is reasonable and acceptable. Therefore, GPUN proposes changes to the TMI-1 UFSAR to permit use of the CDFM methodology for analysis of the portions of the auxiliary steam piping located in the Auxiliary, Fuel Handling and Control Buildings.

Using the standards in 10 CFR 50.92, GPU Nuclear has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91 (a) (1). Also enclosed is the Certification of Service for this request certifying service to the chief executives of the township and county in which the facility is located, as well as the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection.

Sincerely,

James W. Langenbach

James W. Langenbach
Vice President and Director, TMI

/YN

Encl. (1) TMI-1 License Amendment Request No. 281 Safety Evaluation, No Significant Hazards Consideration
(2) Proposed UFSAR Revised Pages
(3) Certificate of Service for TMI-1 License Amendment Request No. 281
(4) EQE International Report No. 240046-R-001, "Seismic Verification Review of Auxiliary Steam System Piping, Three Mile Island Unit 1" Dated June 5, 1998

cc: Administrator, Region I
TMI Senior Resident Inspector
TMI-1 Senior Project Manager
File No. 97062

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
AND
PENNSYLVANIA ELECTRIC COMPANY
THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Licenses Amendment Request No. 281

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF DAUPHIN)

This License Amendment Request is submitted in support of Licensee's request to change the Updated Final Safety Analysis Report (UFSAR) for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for the UFSAR are also included.

GPU NUCLEAR INC.

BY: James W Langenbach
Vice President and Director, TMI

Sworn and Subscribed to before me
this 26th day of May, 1999.

Suzanne C. Miklosik
Notary Public

Notarial Seal
Suzanne C. Miklosik, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Nov. 22, 1999
Member, Pennsylvania Association of Notaries

Enclosure 1

**TMI-1 License Amendment Request No. 281 Safety Evaluation
No Significant Hazards Consideration**

I. License Amendment Request

GPU Nuclear requests that TMI-1 Updated Final Safety Analysis Report (UFSAR) Chapter 5 and 14 be revised to include the use of the Conservative Deterministic Failure Margin (CDFM) methodology for analysis of the portions of the auxiliary steam system piping in the Auxiliary, Control and Fuel Handling Buildings at TMI Unit 1.

Revised TMI UFSAR Chapter 5 and 14 pages indicating the proposed changes are attached. TMI-1 UFSAR Sections 5.4.4, 5.8 and Appendix 14A will be updated to include CDFM methodology as described below, upon approval. The proposed changes to the TMI-1 UFSAR will be implemented following NRC approval.

II. Reason for Change

The Three Mile Island Unit 1 Updated FSAR (TMI-1 UFSAR) states that the auxiliary steam piping is analyzed in accordance with the USAS B31.1 Code for Power Piping as described in the TMI-1 UFSAR. The purpose of this analysis is to demonstrate that the auxiliary steam piping will not rupture and create a harsh environment with the potential to adversely impact the function of safety related equipment located in the vicinity. The auxiliary steam pipe performs no direct function related to safe plant operation or safe shutdown. The pipe does not contain significant amounts of radiation and is operated at a low pressure (nominal 11 psig)[1]. The pipe is not required to remain 100% leak-tight and is not required to provide steam to any system following a seismic event.

The USAS B31.1 code is developed to demonstrate that piping designed in accordance with the code will maintain leak-tightness and will have stresses in the elastic range during a plant safe shutdown earthquake (SSE). This code is typically used for nuclear safety related piping systems and piping systems that form part of the reactor pressure boundary. The use of the USAS B31.1 code to demonstrate that a pipe will not rupture during a seismic event is extremely conservative. In the case of the auxiliary steam pipe at TMI, the design of the piping system does not conform to the requirements of the USAS B31.1 code. Thus, it would appear that the pipe would not survive a seismic event. However, as indicated in the attached report developed by EQE International (Reference 1), the auxiliary piping has been shown to be capable of surviving the TMI-1 SSE with considerable margin. The assessment of the auxiliary steam pipe is based on the Conservative Deterministic Failure Margin (CDFM) methodology described in EPRI Report NP-6041-SL (Reference 2).

The CDFM methodology is an analytical approach developed to provide a basis for evaluating pipe and pipe supports based on experience gained from observing the behavior of piping systems subjected to actual earthquakes. The methodology is based on the use of expert engineering experience to identify configurations and conditions that are potentially vulnerable to seismic events. The methodology then provides analytical tools to be used to determine if these potential vulnerabilities do in fact adversely impact the ability of the pipe or supports to survive the SSE without a pipe rupture. The

[1] This piping is protected by relief valves set to lift at 16 psig.

methodology permits the engineer to specifically focus on configurations that have been shown by experience and test data to potentially be vulnerable to seismic events. This allows a design that contains less conservatism than traditional piping analyses but still has sufficient margin to withstand a specified seismic event without rupture or significant leakage.

The CDFM methodology was developed by a panel of experts in the fields of seismic analysis and seismic margin assessment. The methodology is based on actual earthquake experience data, generic equipment qualification and fragility test data and the results of analysis of piping systems subjected to earthquakes. The methodology was developed by inspection and analysis of piping that survived earthquakes much larger than the TMI-1 SSE without rupture or significant leakage. Based on the experience data, the CDFM methodology has been shown to be adequate to predict the performance of piping systems subjected to actual earthquakes. The CDFM methodology has been used at a number of nuclear plants (e. g. Edwin I Hatch Nuclear Plant-Unit 2, Limerick Generating Station-Unit 2 and Duane Arnold Energy Center) to analyze piping systems to address the main steam isolation valve leakage rate and control room habitability issues. These plants have received safety evaluation reports (SERs) from the U. S. Nuclear Regulatory Commission. These SERs conclude that with the available safety margins demonstrated by the results of the analysis, the employment of the CDFM methodology to demonstrate the functional capability of the system is reasonable and acceptable.

The CDFM approach has been used to analyze the auxiliary steam piping at TMI Unit 1 because it provides the most realistic assessment of the ability of the piping to withstand a seismic event equal to the TMI-1 SSE without rupture or significant leakage. As described in the attached report prepared by EQE International, the auxiliary steam pipe at TMI Unit 1 complies with the provisions of the CDFM methodology with significant margin. Therefore, the analysis shows that the pipe will not rupture and adversely impact the function of any safety-related equipment located in the vicinity of the pipe. This methodology will be used by GPUN only for the analysis of the portions of the auxiliary steam system piping located in the Auxiliary, Fuel handling and Control Buildings at TMI Unit 1.

The proposed changes to the TMI-1 UFSAR will be implemented following NRC approval.

The CDFM methodology has been used at a number of other nuclear plants (e.g., Edwin I Hatch Nuclear Plant-Unit 2, Limerick Generating Station-Unit 2 and Duane Arnold Energy Center) to analyze piping systems to address the main steam isolation valve leakage rate and control room habitability issues. These plants have received safety evaluation reports (SERs) from the US Nuclear Regulatory Commission. These SERs conclude that with the available safety margins demonstrated by the results of the analysis, the employment of the CDFM methodology to demonstrate the functional capability of the system is reasonable and acceptable.

III. Safety Evaluation Justifying Change

Although the auxiliary steam line is a low pressure system, Appendix 14A of the TMI-1 UFSAR currently states that it is a high energy line and that all high energy lines are analyzed in accordance with the requirements of the USAS B31.1.0 code for power piping. However, using USAS B31.1.0 to design a pipe when the only requirement is that the pipe maintains integrity during a seismic event is extremely conservative. Since the pipe design does not comply with the requirements of the USAS B31.1.0 code, the auxiliary steam line was recently analyzed in accordance with the CDFM methodology as described in the attached report produced by EQE International (Reference 1). The results of this analysis show that the pipe will maintain integrity during a seismic event and that the design complies with the requirements of the CDFM methodology with significant margin.

The auxiliary steam system and the portions of the auxiliary steam piping located in the Auxiliary, Control and Fuel Handling Buildings do not perform any safe shutdown function and are not required for safe operation of the plant. However, a breach of this high-energy line could result in a release of steam to some portions of the Auxiliary, Control or Fuel Handling Buildings. Since equipment in these areas has not been evaluated for the effects of the resulting harsh environment from a pipe rupture, an analysis of the seismic adequacy of this line is required.

The attached EQE International report (Reference 1) describes in detail the methodology used to evaluate the piping system and presents a summary of the results of the analysis. The analysis was performed in accordance with the CDFM methodology as described in the report. The CDFM methodology was developed by a group of experts in the field of seismic analysis and is based on the observed behavior of a significant number of piping installations subjected to actual strong motion earthquakes. Details of the development and application of the methodology are contained in EPRI Report NP-6041SL (Reference 2).

The analysis performed by EQE International and described in Reference 1 includes a detailed walkdown of the auxiliary steam system piping configuration and an analysis of portions of the pipe and supports judged to be most likely to be vulnerable to a seismic event. As described in the report, the auxiliary steam pipe satisfies the design requirements of the CDFM methodology with significant margin. Based on the results of this analysis, the auxiliary steam system piping will not rupture during or after a safe shutdown earthquake.

IV Environmental Consideration

GPU Nuclear has determined that this change to the TMI-1 UFSAR Chapter 5 and 14 to include the use of the Conservative Deterministic Failure Margin (CDFM) methodology for seismic analysis of the auxiliary steam pipe in the Auxiliary, Control and Fuel Handling Buildings at TMI Unit 1, does not involve an unreviewed environmental safety question. The analysis performed by EQE International shows that the pipe will maintain integrity with sufficient margin to prevent adverse impact on any nuclear safety related equipment.

V. No Significant Hazards Consideration

GPU Nuclear has determined that this license amendment, pursuant to 10 CFR 50.92 (c), involves no significant hazard consideration since:

(1) The proposed amendment, use of CDFM methodology for the analysis of the auxiliary steam system piping, would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The analysis of the auxiliary steam pipe using the CDFM methodology demonstrates that the pipe will maintain integrity sufficient to prevent adverse impact on safety related equipment during a safe shutdown earthquake (SSE). The methodology is based on actual earthquake experience data and has been shown to be adequate to demonstrate that piping systems will maintain integrity. The CDFM methodology was developed by experts in the field of seismic analysis and is based on actual earthquake experience and the results of dynamic tests with large seismic accelerations. The methodology provides a conservative mechanism for analytically predicting performance during actual earthquakes, and thus its application would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) The proposed amendment, use of CDFM methodology for the analysis of the auxiliary steam system piping, would not create the possibility of a new or different kind of accident from any accident previously evaluated.

No changes to plant systems, structures or components are proposed and no changes to methods of operation are involved.

(3) The proposed amendment, use of CDFM methodology for the analysis of the auxiliary steam system piping, would not involve a significant reduction in a margin of safety.

No changes are proposed to operating limits or safety system settings, or to accident analysis acceptance criteria. The CDFM methodology provides a conservative mechanism for analytically predicting system performance during actual earthquakes. Its application to the auxiliary steam system piping would not involve a significant reduction in a margin of safety.

VI. Implementation

GPU Nuclear requests that the amendment authorizing this change become effective immediately upon issuance.

VII References

(1) Report No. 240046-R-001, "Seismic Verification Review of Auxiliary Steam System Piping, Three Mile Island Unit 1" prepared by EQE International, dated June 5, 1998.

(2) EPRI NP-6041SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" prepared by Electric Power Research Institute (EPRI), dated July 1991

Enclosure 2

Proposed UFSAR Revised Pages
License Amendment Request No. 281

5.4-9
5.4-14
5.4-14a
5.8-5
14A-3
14A-6
14A-14

(The proposed changes are shown by margin bar.)

5.4.3.2.5 Tornado Missiles

Equipment within the Auxiliary Building vital to the safe shutdown of the station is included in the tornado protection criteria.

There is no equipment located in the Turbine Building that is required for safe shutdown of the plant. The emergency feedwater pumps are located in the Intermediate Building and have been protected by a structure designed for the tornado loads as stated in Section 5.4.3.2.2.

5.4.4 PIPING DESIGN CRITERIA

5.4.4.1 Seismic Class I, II And III Piping System Design

The piping design criteria for the primary loop piping and related pipe lines is discussed in Chapter 4. The following design criteria cover the other pipe lines which are not described in Chapter 4.

Nuclear class piping systems have been fabricated, erected, and inspected with the intent of satisfying the applicable sections of the Code for Nuclear Power Piping, USAS B31.7 February 1968 Draft with Errata through June 1968 (issued for trial use and comment).

The basic guideline for the design of piping has been the Code for Pressure Piping, USAS B31.1.O-1967, and those portions of Code Cases N7, N9, and N10 that pertain to design criteria. Pertinent sections from this Code apply to Class I and II piping described hereafter:

a. Code Paragraph 101.5.3, Earthquake

   "The effect of earthquakes, where applicable, shall be considered in the design of piping, piping supports, and restraints, using data for the site as a guide in assessing the forces involved. However, earthquake need not be considered as acting concurrently with wind."

b. Code Paragraph 101.5.4, Vibration

   "Piping shall be arranged and supported with consideration of vibration."

c. Code Paragraph 121.2.5, Sway Braces

   "Sway braces or vibration dampeners shall be used to control the movement of piping due to vibration."

d. Assure that spring-type hangers are correctly adjusted for cold conditions.

e. Assure that seismic restraints are located as indicated on the appropriate drawings and should not interfere with expected pipe motion during heatup and operation.

2) During the cold functional testing of systems, station personnel and/or assigned test personnel observe system piping and equipment for vibration and pipe motion upon initiation of system operation and record vibration data for rotating equipment and report any observed vibration or motion of piping and equipment.

3) During system heatup, hot functional testing, initial system, hot operation system, plant personnel, and/or assigned test personnel inspect and observe systems for the following:

a. Assure that hangers and supports do not interfere with system equipment and piping expansion.

b. Assure that seismic restraints do not interfere with equipment and piping expansion.

c. Observe spring-type hangers for proper hot loading.

d. Observe system equipment and piping for vibration and report observations to the appropriate responsible person.

During startup testing, as power level and thus, steam, feedwater, and heater drain flows increase, system is inspected and observed for equipment and piping vibration.

During surveillance testing of systems or equipment, station personnel observe system equipment and piping for motion upon initiation of the system and for vibration after system initiation, report their findings to the shift foreman.

During plant operation the nuclear plant auxiliary operators observe equipment and piping of operating systems during their plant shift tours, reporting any vibrations noted to the shift foreman on duty.

Inspections of equipment and piping supports, hangers, and seismic restraints, in accordance with the ASME Section XI, to assure that they are functioning properly and that spring-type hangers are assuming the proper loading. Maintenance procedures require that hangers or supports which require blocking or removal during the specific maintenance operation are returned to the proper condition upon completion of the job.

5.4.4.2 Auxiliary Steam System Piping Design

The auxiliary steam system does not perform any safety related function and is not required for normal plant operation or for safe plant shutdown. However, the portions of the piping located in the Auxiliary, Fuel Handling and Control Buildings must maintain pressure boundary to prevent potential damage to safety related components and systems located within these structures. Therefore, these portions of the auxiliary steam system piping were analyzed for combined operating and seismic conditions to ensure that a rupture of the pipe will not occur. The pipe is designed for normal operating and occasional loads in accordance with USAS B31.1.0 code and, in addition, the portions described above are analyzed for seismic loads in accordance with the conservative deterministic failure margin (CDFM) methodology described in EPRI NP-6041-SL (Reference 66). Details of the methodology and the results of the CDFM analysis are contained in report 240046-R-001 (Reference 67). Important aspects of the CDFM methodology as applied to the analysis of the auxiliary steam system piping are summarized below:

a. The CDFM methodology consists of an explicit analysis of portions of the subject piping system combined with a detailed walkdown of the entire system to identify potential vulnerabilities.

b. The seismic inputs to the analysis using the CDFM methodology are obtained from 5% damped amplified response spectra in place of the damping specified for Seismic Class I piping analysis. The analysis includes loads resulting from operating conditions and the SSE.

c. The amplified in-structure response spectra used for the evaluation of the auxiliary steam piping is the spectra developed for the resolution of USI A-46 as described in EQE report 42105-R-001 (Reference 68). These spectra are more conservative than the median centered amplified response spectra permitted by the CDFM approach.

d. The CDFM approach uses ASME service level D allowables (e.g. 3.0 Sh) for evaluating pipe and pipe components for load combinations that include the SSE. Ultimate strength values are used for the evaluation of support members and steel and concrete structures for SSE conditions. The 20% reduction in calculated stresses permitted by the CDFM methodology to account for inelastic energy absorption is not utilized.

e. Expansion anchor bolt allowables are based on a factor of safety equal to three on ultimate capacity.

65. GPUN TDR-1212, "TMI Response to Generic Letter 96-06".

66. EPRI NP-6041SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Electric Power Research Institute. Revision 1, July 1991.

67. Report 240046-R-001, "Seismic Verification Review of Auxiliary Steam System Piping", EQE International. Revision 1, June 5, 1998.

68. Report 50097-R-001, "Conservative Design In-Structure Response Spectra for Resolution of Unresolved Safety Issue A-46 for the Three Mile Island Nuclear Generating Station, Unit 1", EQE International. Revision 1, July 1993.

3.0 GENERAL DISCUSSION

3.1 DESIGN

A rupture of the high energy piping is considered highly unlikely due to the low seismic and operating stress levels. All these systems have been conservatively designed and all the systems except auxiliary steam to the Auxiliary Building have been analyzed in accordance with USAS B31.1.0, Code For Power Piping. The auxiliary steam system to the Auxiliary Building has been designed and analyzed as described in Section 5.4.4.2. This includes all portions of the auxiliary steam system located in the Control, Fuel handling and Auxiliary Buildings. Results of these analyses show that the maximum stress levels from combined operating and seismic conditions are well below those limits designated as potential pipe rupture stress levels.

Piping systems are designed to USAS B31.1.0. In addition, portions of the auxiliary steam system piping are analyzed in accordance with the CDFM methodology as described in Section 5.4.4.2. Quality assurance was applied to USAS B31.1.0 requirements for the non nuclear piping and to USAS B31.7 requirements for nuclear piping. (Nuclear piping is defined as piping that normally contains radioactivity.) The analysis of the auxiliary steam piping is based on the configuration and conditions of the piping system at the time of the walkdown and evaluation. To ensure that the existing condition is maintained, the quality classification of the auxiliary steam piping system has been upgraded to "Regulatory Required with QA".

On non-nuclear piping, welders qualified to ASME Section IX requirements were used. Piping was hydrostatically tested and subjected to visual inspection. In addition, the following steps were taken:

a. The main steam piping welds 4 inches and over were 100 percent radiographed from steam generators to the turbine generator.

b. The main feedwater piping welds 4 inches and over were 100 percent radiographed from pumps to steam generator.

c. The emergency feedwater piping welds were 100 percent radiographed from the steam generators up to the first isolation valve (which is in the Intermediate Building).

d. The steam supply (to the emergency feedwater pump turbine) piping welds were 100 percent radiographed.

All the NDT required by USAS B31.7 was applied to nuclear piping systems, i.e., decay heat, makeup and purification, sampling, and so forth.

3.2 QUALITY ASSURANCE

The design and construction phase Quality Assurance Program was a three-level program. The first level of the program was performed by the equipment manufacturer or site contractor, the second level by Met- Ed's main contractor (i.e., B&W--NPGD, GAI, or UE&C, as appropriate), and the third level by Met-Ed itself and/or its agent, MPR Associates.

In addition, crack breaks were posulated at adverse locations and assumed to be one half the pipe diameter in length and one half the pipe wall thickness in width.

The specific thrust versus time curves used in designing the restraints defined in this supplement are shown on Figures 14A-14 and 14A-15.

4.1 CONTROL ROOM AND CONTROL BUILDING

The Control Building equipment, electrical power and control, chilled water system, and ductwork systems are contained within the structure of the Control Building. In this isolated location they would not experience adverse effects from any high energy pipe break. Access to the Control Building structure is either through the Turbine Building or the Fuel Handling Building. Outside air to the Control Building is ducted to the Control Room from a remote underground intake terminal and would not be adversely affected by a high-energy pipe break. The Turbine Building would experience momentary overpressure if the break occurred in this area, but this would be dissipated through numerous wall and roof openings. Steam leakage from the turbine hall to the Control Room or Control Building during this period is minimized as it is forced to travel through the west Turbine Building wall, through multiple doors in series, before entering the Control Building areas. Also, the doors have automatic closers. Any steam leakage into the corridor space outside the Control Room or the Control Room space will be condensed and dissipated by the ventilation systems, and no significant ambient changes would be anticipated in these areas.

Investigation indicates that there are no high-energy lines larger than 1 inch other than the auxiliary steam pipe in or near the Control Building, and thus, postulated pipe whip and steam jet impingement are not able to damage the Control Room. Due to the low operating pressure of the auxiliary steam system, rupture of this line is not a consideration.

High-energy sample lines (under 1 inch) are discussed in Section 4.3.

4.2 INTERMEDIATE AND TURBINE BUILDINGS

a. The Intermediate and Turbine Buildings contain all of the lines over 1 inch with internal fluid exceeding both 200F and 275 psig.

b. Electrical equipment is required to function subsequent to a High Energy Break (HELB) inside or outside of containment and must meet environmental qualification requirement.

   The qualification conditions considered include postaccident pressure and temperature conditions. The result is shown in Table 14A-7. Electrical equipment which is required to function following a postulated HELB is located at the 295 ft elevation of the Intermediate Building.

5.0 SUMMARY AND CONCLUSIONS

The results of this design review are summarized as follows:

a. A rupture of the high energy piping systems is considered highly unlikely. The systems other than auxiliary steam piping have been conservatively designed in accordance with the criteria in USAS B31.1.0, Code for Power Piping. Materials, fabrication, and quality assurance requirements of the Code have been utilized. In addition, the main steam piping has been subject to 100 percent radiography of welds from the steam generators to the turbine stop valves, and the feedwater piping has been subject to 100 percent radiography from the steam generators to the feedwater pumps. Quality assurance provisions of USAS B31.7, Code for Nuclear Piping, have been implemented for nuclear systems. The auxiliary steam piping system has been shown to be adequate for combined effects of operating conditions and the SSE. Quality assurance provisions are implemented to ensure configuration of the auxiliary steam piping system is maintained.

b. The Atomic Energy Commission criteria had been implemented in identifying postulated break locations in high energy piping systems.

c. All of the equipment required for shutdown is protected from the postulated ruptures by virtue of location (remote from high energy lines), restraints, inspection, or barriers.

d. The Control Room will remain habitable and operable following a postulated high energy pipe break due to its remote location from such breaks.

e. The plant can be brought to cold shutdown conditions by utilizing either feedwater system or the Emergency Core Cooling System and Reactor Building cooling.

f. This analysis shows that the borated water storage tank serves as an adequate interim heat sink from the standpoint of core covering, Reactor Building pressure considerations and achieving cold shutdown.

g. Further, the analysis shows that the core remains covered, the Reactor Building is protected against excessive pressure, the plant can be taken to cold shutdown, and containment integrity can be maintained.

Enclosure 3

Certificate of Service for TMI-1
License Amendment Request No. 281

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF GPU NUCLEAR INC.	DOCKET NO. 50-289 LICENSE NO. DPR-50

CERTIFICATE OF SERVICE

This is to certify that a copy of License Amendment Request No. 281 to the Facility Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Darryl LeHew, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, PA 17057

Ms. Sally S. Klein, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Front & Market Streets
Harrisburg, PA 17101

Director, Bureau of Radiation Protection
PA Dept. of Environmental Resources
Rachael Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469
Attn: Mr. Stan Maingi

GPU NUCLEAR INC.

BY: James W. Langenbach
Vice President and Director, TMI

DATE: 5/26/99

Enclosure 4

EQE International Report No. 240046-R-001,
"Seismic Verification Review of Auxiliary Steam System Piping,
Three Mile Island Unit 1"
Dated June 5, 1998
License Amendment Request No. 281

EQE INTERNATIONAL

Engineering
Risk ▪ Safety ▪ Design

June 28, 1999

NE-99-063

Mr. Yosh Nagai
GPU Nuclear Corp.
1 Upper Pond Road
Parsippany, NJ 07054

Subject: NRC Reproduction of EQE Report

Dear Sir:

The following confirms telephone conversations between Mr. Ken Whitmore (GPU) and Mr. Jim White (EQE) concerning the copyright page for EQE Report 240046-R-001, "Seismic Verification Review of Auxiliary Steam Piping, Three Mile Island Unit 1," Revision 1.

GPU and the NRC are granted permission to reproduce this report as required for their use.

Please let me know if you have any questions or require any additional information.

Very truly yours,

James L. White

James L. White
Project Manager

JLW:hlh

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