April 18, 1986

DMB016

Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated February 4, 1986.

This amendment revises the repair limits for the steam generator tubes under a restrictive set of circumstances as described in your request. The amendment is also only effective until the next refueling outage at which time the steam generator tube repair criteria will be re-evaluated.

This request for amendment was noticed February 28, 1986 (51 FR 7157). On March 10, 1986, Three Mile Island Alert, Inc. (TMIA) requested a hearing on this amendment and petitioned for leave to intervene. On March 27, 1986, TMIA provided comments concerning the staff's proposed no significant hazards consideration determination. A discussion of public comments relevant to this amendment and a Final Determination of No Significant Hazards Consideration are included in the enclosed Safety Evaluation.

A copy of the Notice of Issuance and Final Determination of No Significant Hazards Consideration is enclosed. A repeat of the Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY
JOHN F. STOLZ

John O. Thoma, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 116 to DPR-50
2. Safety Evaluation

cc w/enclosures: See next page

AD:PWR-B
DCrutchfield*
4/11/86

*See previous white for concurrences.

PBD-6	PBD-6	PBD-6			PBD-6	OELD
RIngram	JThoma:cf*	RWeller*	JRajan*	CYCheng*	JStolz*	MWagner*
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and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

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SUBJECT: AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated February 4, 1986.

This amendment revises the repair limits for the steam generator tubes under a restrictive set of circumstances as described in your request. The amendment is also only effective until the next refueling outage at which time the steam generator tube repair criteria will be re-evaluated. However, we interpret the phrase "at which time the repair limit for the primary tube freespan will be such a limit as has been approved by the NRC" to mean that NRC approval will be obtained via a Technical Specification change.

This request for amendment was noticed February 28, 1986 (51 FR 7157). On March 10, 1986, Three Mile Island Alert, Inc. (TMIA) requested a hearing on this amendment and petitioned for leave to intervene. On March 27, 1986, TMIA provided comments concerning the staff's proposed no significant hazards consideration determination. A discussion of public comments relevant to this amendment and a Final Determination of No Significant Hazards Consideration are included in the enclosed Safety Evaluation.

A copy of the Notice of Issuance and Final Determination of No Significant Hazards Consideration is enclosed. A repeat of the Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

John O. Thoma, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. to DPR-50
2. Safety Evaluation

cc w/enclosures: See next page

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RIngram	JThoma:cf	RWeller	JRajan	CYCheng	JStolz	DCrutchfield	MEWagner
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with attached changes on SER

April 25, 1986

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JThoma

DOCKET NO. 50-289

MEMORANDUM FOR: Rules and Procedures Branch
Division of Rules and Records
Office of Administration

FROM: Office of Nuclear Reactor Regulation

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1

One signed original of the *Federal Register* Notice identified below is enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (6) 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 Consideration of 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).
- ☐ Order.
- ☐ Exemption.
- ☐ Notice of Granting Exemption.
- ☐ Environmental Assessment.
- ☐ Notice of Preparation of Environmental Assessment.
- ☒ Other: Notice of Issuance of Amendment to Facility Operating License and Final Determination of no Significant Hazards Consideration

Office of Nuclear Reactor Regulation
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosure:
As stated

Contact: J. Thoma
Phone: 28018

OFFICE ▶	PBD#6						
SURNAME ▶	RIngram;cf						
DATE ▶	4/[illegible]/86						

NRC FORM 318 (10/80) NRCM 0240

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Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety & Licensing Appeal
Board Panel (8)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensees) dated February 4, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 116, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
4-80	4-80
——	4-80a
4-82	4-82

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss of coolant accident requiring actuation of the engineering safeguards, or

4. A major main steam line or feedwater line break.

4.19.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawing or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

6. Repair Limit means the extent of degradation at or beyond which the tube shall be repaired or removed from service because it may become unserviceable prior to the next inspection.

This limit is equal to 40% of the nominal tube wall thickness, except for the primary side tube freespan.

For the primary side tube freespan, the repair limit is either:

a. 50% of the nominal tube wall thickness and defect length of 0.55 inches or less; or

b. 40% of the nominal tube wall thickness and defect length greater than 0.55 inches; or

c. 40% of the nominal tube wall thickness in areas of reduced eddy current sensitivity (upper and lower tubesheet secondary faces and support plate entry and exit locations).

This primary side repair limit applies until Refueling Outage 6R, at which time the repair limit for the primary tube freespan will be 40% of the nominal tube wall thickness or such other limit as has been approved by the NRC by license amendment.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.19.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet, except as permitted by 4.19.2.b.2, above.

The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The Unit is expected to be operated in a manner such that the primary and secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the primary or secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result.

The extent of steam generator tube leakage due to cracking would be limited by the secondary coolant activity Specification 3.1.6.3.

The extent of cracking during plant operation would be limited by the limitation of total steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gpm). Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired or removed from service.

Wastage-type defects are unlikely with proper chemistry treatment of the primary or secondary coolant. However, even if a defect would develop in service, it will be found during scheduled inservice steam generator tube examinations. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Plugging or repair will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness, except for the primary tube freespan.

For the primary side tube freespan, plugging or repair is required for degradation either (a) equal to or greater than 50% of the tube nominal wall thickness if the defect length is less than or equal to 0.55 inches; or (b) equal to or greater than 40% of the tube nominal wall thickness if the defect length is greater than 0.55 inches; or (c) equal to or greater than 40% of the tube nominal wall thickness if the defect is located in an area of reduced eddy current sensitivity (upper and lower tubesheet secondary faces and tube support plate entry and exit locations). The above plugging criteria for the primary side tube freespan apply only until Refueling Outage 6R, at which time the repair limit will be 40% of the nominal tube wall thickness or such other limit as has been approved by the NRC.

Where experience in similar plants with similar water chemistry, as documented by USNRC Bulletins/Notices, indicate critical areas to be inspected, at least 50% of the tubes inspected should be from these critical areas. First sample inspections sample size may be modified subject to NRC review and approval.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

INTRODUCTION

By letter dated February 4, 1986, General Public Utilities Nuclear Corporation (GPUN or the licensee) requested a change in the Technical Specifications (TSs) for Three Mile Island Unit 1 (TMI-1) which would result in a revision to the repair limits for the Once Through Steam Generator (OTSG) tubes. GPUN, in the past, has repaired steam generator tubes based on a general 40% throughwall repair limit. This existing steam generator tube repair limit defines as acceptable a tube with a defect 360° in circumferential extent, unlimited axial extent and depth up to 40% of the tube wall thickness. The licensee has proposed criteria that are based on the extent of both depth and length of the defects.

The proposed criteria apply to primary side (inner surface of the tube wall) defects only. For primary side defects which penetrate greater than 40% throughwall, the proposed repair criteria would allow continued tube operation with a defect of up to 50% throughwall penetration providing its length was less than 0.55 inches. The upper and lower tube sheet secondary faces and tube support entry and exit locations in the primary side, which are areas of reduced eddy current sensitivity, are excluded; the repair limit for defects in these areas as well as those on the outer surface of the tubes (secondary side) remains 40% of the nominal tube wall thickness.

The existing 40% throughwall repair limit is based on an allowance of 10% for eddy current detection uncertainties and an allowance of 10% for corrosion which may occur during inspection intervals. The licensee's proposed plugging limit for the primary side also provides an allowance of 10% for eddy current detection uncertainties but no allowance for corrosion. Tubes with both outer diameter and inner diameter indications at approximately the same elevation will be dispositioned on a case-by-case basis. The proposed plugging criteria will remain in effect until the next scheduled refueling outage (scheduled for approximately December 1986). Based on results of steam generator inspections and the analysis of pulled tubes, the licensee will evaluate what repair criteria should apply on restart from the refueling outage and will make an appropriate application to the NRC for approval.

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On February 28, 1986, the NRC published in the Federal Register a notice entitled "Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing" (51 FR 7157). A request for hearing and a petition for leave to intervene were received on March 10, 1986 on behalf of Three Mile Island Alert, Inc. (TMIA). Subsequently at a prehearing conference held on March 27, 1986, the Atomic Safety and Licensing Board granted the request for a hearing and ruled on admissibility of contentions. By letter dated March 27, 1986, TMIA filed a petition titled "TMIA's Comments on Proposed No Significant Hazard Consideration Finding." Additionally, Congressman Markey, in the course of budget considerations for the NRC, asked a question of the NRC staff concerning the proposed no significant hazards consideration finding. No other comments were received relevant to this amendment request.

DISCUSSION

The licensee has developed an analytical methodology in an attempt to demonstrate that a margin equivalent to the existing 40% plugging criteria can be provided by a tube with a defect greater than 40% of the tube wall thickness and a given continuous length. The analyses include ASME Section III and Section XI fatigue evaluations, and the most severe accident load (Main Steam Line Break Accident, MSLB) analysis conducted during the evaluation of the 1981 tube cracking experience.

Additionally, the licensee demonstrated eddy current testing (ECT) accuracy with the aid of test data obtained from metallurgical samples. GPUN utilized a standard differential eddy current technique to examine the tubing in the TMI OTSGs. In order to establish a more accurate conversion curve for the specific discontinuities present in the TMI-1 OTSGs, the traditional curve was enhanced through the use of supplemental reference points. These supplemental reference points were based on eddy current responses from synthetic defects placed on the inner diameter of inconel tubing representative of the actual OTSG tubing.

The licensee has not considered an additional thickness degradation allowance for environmental corrosion, as suggested in Regulatory Guide 1.121, in formulating the proposed 50% criteria. The primary reason for not considering an additional allowance is that on the primary side corrosive attack is not an ongoing phenomenon and this proposed plugging limit is applicable to defects on the primary side only.

The NRC staff, in an affidavit to the Atomic Safety and Licensing Appeal Board (ASLAB) in 1985, concurred with the licensee relative to the cessation of corrosion on the primary side. From data presented by the licensee, the NRC staff concluded that reasonable assurances exist that operational forces during the hot functional testing performed in 1983, subsequent to the recorded eddy current examinations in 1982, caused grain dropout and grain boundary separation of previously existing IGA, and these grain dropouts enabled ECT to detect this degradation. Thus the new indications were not the result of a new corrosion mechanism and no direct evidence could be found to show that a different type of corrosion mechanism existed. Based on this and other evidence, the ASLAB concluded that the corrosion on the primary side had ceased. ALAB-807, 21 NRC 1195, 1202-05 (1985).

EVALUATION

Structural Integrity

The licensee has used elastic-plastic fracture mechanics, limit load analysis, and linear elastic fracture mechanics to justify the proposed plugging criteria. The primary objective in developing a plugging criterion is to minimize formation of a throughwall crack in the steam generator tube so as to prevent tube leakage between tube inspections. Due to the thin walls of the pressure retaining boundary under consideration and the environment present in steam generators, caution is necessary if fracture mechanics is to be used for this purpose. The NRC staff employed a different methodology to determine the acceptability of the proposed plugging criteria.

From data presented by the licensee, the NRC staff performed independent calculations to determine the minimum acceptable wall thickness for degraded TMI-1 tubes. The acceptance criteria of Regulatory Guide 1.121 and the ASME Code, Section III, were used to establish allowable stress and pressure limits. Minimum wall thicknesses were calculated for a number of different acceptance criteria specified in Regulatory Guide 1.121 using the measured mechanical properties of tubes. The limiting criterion is the requirement that the margin of safety against tube rupture under normal operating conditions should not be less than three. To meet this requirement, a minimum wall thickness of 0.0135 inch or degradation of no more than 60% is allowed. This calculation is independent of the crack length. Additional allowances should be considered for eddy current testing uncertainty in measuring defects and possible tube corrosion occurring between inspections from environmental effects. Each of these considerations is discussed below.

Eddy Current Testing (ECT)

Concerning eddy current testing, the licensee presented data based on tubes removed from service in 1981 and 1982. This data has been extensively analyzed, however, there remain questions as to some of the conclusions drawn regarding minimum eddy current detectability and grain dropout due to subsequent hot functional testing. We believe conclusions concerning these questions can only be confirmed by examination and measurements on tubes that have seen service in the interim period since the tubes were originally removed.

With respect to our concern over the eddy current detectability of intergranular attack (IGA) and possible cracks associated with the IGA, the phase angle change in eddy current reading from 1982 to 1984 results in an apparent decrease in depth of the indications. Since an actual decrease in depth is not feasible, and even after extensive reanalysis, GPUN was unable to conclusively explain
the cause of the phase shift, there is some doubt as to the overall accuracy of the eddy current testing technique to precisely characterize IGA type defects. There is also the question of IGA masking eddy current signals from intergranular cracks that may initiate and grow from the bottom of IGA patches. But these questions are more related to a detection capability of eddy current testing rather than the measurement uncertainties associated with the detected defects. As such, these questions are independent of tube plugging criteria.

The GPUN analysis of ECT accuracy estimate is based on the mean of six data points. The NRC staff position is that since an undercall of a single defect can lead to a tube leak, the more conservative upper-bound of the data or maximum undercall should be used as the criteria. Therefore, a conservative allowance of 10% for eddy current error is considered necessary. This allowance will satisfy the staff's concern on eddy current uncertainties, particularly for the time period proposed by this amendment request. It should also be noted that a 10% allowance for eddy current accuracy is part of the current licensing basis for TMI-1. Thus this staff position does not result in a change in the licensing basis for the plant. However, before a permanent tube repair criteria change is considered, the staff desires confirmatory examination and measurement on tubes which are currently in service.

Corrosion Allowance

Concerning an allowance for tube corrosion between inspections, this amendment request would apply only to defects located on the primary side of the steam generator tubes. GPUN maintains that tight control of sulfur concentrations, maintenance of a specific pH range, and the addition of lithium as an inhibitor are likely to prevent reinitiation of cracking due to a corrosive agent. Therefore, at variance with the provisions of Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes, GPUN has stated that no corrosion allowance is necessary.

Based upon the most recent data supplied by the licensee, the staff continues to believe, as previously stated, that the intergranular attack (IGA) and intergranular stress assisted cracking (IGSAC) on the primary side of the OTSG tubes has been arrested as a result of chemical cleaning and is prevented from reccurring by plant chemistry procedures involving pH control and lithium addition. During the unit layup, GPUN layup specifications were followed. Although some out of specification periods did occur, they were promptly corrected and were not of sufficient magnitude to have caused corrosion. Additional corrosion-preventive conditions were also maintained during layup. During hot functional operations, system chemistry conditions were maintained within specifications such that no further corrosion was expected to occur in the TMI-1 steam generators. The NRC staff determines that, for the time interval involved in this amendment request, no corrosion allowance is necessary for defects located on the primary side of the steam generator tubes.

Summary

The NRC staff's analysis therefore results in a 10% wall thickness allowance to account for eddy current uncertainties and no wall thickness allowance for corrosion on the primary side until the next refueling outage. Thus a plugging criteria of 50% depth of degradation with no limitation on the length of the defects would be acceptable for the primary side. However, the licensee proposed a maximum length for defects located on the primary side with a throughwall penetration greater than 40% but less than 50%. Limiting the length of these defects to relatively small values provides additional assurance for maintaining the tube structural integrity. In addition, the licensee proposes to exclude from the new criteria areas on the primary side of reduced eddy current capability: namely defects located on the inner surface near the upper and lower tube sheet secondary faces and tube support entry and exit locations. For areas on the primary side with reduced eddy current sensitivity and all areas on the secondary side of the tubes, the existing 40% plugging limit is still applicable. The staff concludes that the conditions established by the proposed amendment over the relatively short effective interval of the amendment are bounded by the same safety margins that exist for the 40% tube plugging limit.

Conclusions

Based on NRC staff review of the proposed amendment and the supporting documents submitted by GPUN, it is concluded that the proposed repair limit for the primary side tube freespan is acceptable. This repair limit is either:

a. 50% of the nominal tube wall thickness and defect length of 0.55 inches or less, or

b. 40% of the nominal tube wall thickness and defect length greater than 0.55 inches, or

c. 40% of the nominal tube wall thickness in areas of reduced eddy current sensitivity (upper and lower tube sheet secondary faces and support plate entry and exit locations).

This primary side repair limit applies until refueling outage 6R (approximately December 1986) at which time the repair limit for the primary side freespan will be reevaluated by the NRC staff. The existing 40% plugging limit for the secondary side defects remains unaffected by this change request. Defects on the primary side which may be in close proximity to secondary side defects and whose identification may be difficult will be evaluated on a case-by-case basis.

SIGNIFICANT HAZARDS CONSIDERATION COMMENTS

The amendment application was noticed as a proposed no significant hazards consideration determination on February 28, 1986 (51 FR 7157) and a request for hearing and a petition for leave to intervene were received on March 10, 1986. The filing was on behalf of Three Mile Island Alert, Inc. (TMIA).

The petition did not make specific comments concerning the no significant hazards consideration determination, but it did provide five contentions the petitioners desired to address in hearing. Subsequently at a prehearing conference held on March 27, 1986, the Atomic Safety and Licensing Board granted the request for a hearing and ruled on which contentions would be admitted. By letter dated March 27, 1986, TMIA filed a petition titled "TMIA's Comments on Proposed No Significant Hazard Consideration Finding." (TMIA Comments). Additionally, Congressman Markey, in the course of budget considerations for the NRC, asked a question of the staff concerning the proposed no significant hazards consideration finding. The staff response to Congressman Markey is included in this section. No other comments were received relevant to this amendment request.

TMIA Comments

A synopsis of the TMIA comments and the NRC staff's response follows:

A. The Form of New Tube Degradation Has Not Been Determined (TMIA Comments at 4-10)

The basic thrust throughout TMIA's discussion in this section is that the November 1984 steam generator testing revealed new defects at TMI-1 which were caused by an unknown mechanism. TMIA contends that the mechanism is unknown because no physical tube samples were removed and analyzed. TMIA attempts to point out deficiencies in the analysis to indicate that the licensee cannot be certain that a new mechanism is not causing the new indications. TMIA had previously raised these contentions to the Atomic Safety and Licensing Appeal Board (ASLAB) when the indications were first detected.

The licensee's analysis concludes that the degradation detected by eddy current testing in late 1984 is not a continuation of the old IGSAC, but rather is integranular attack which occurred simultaneously with the IGSAC in 1981. There are questions raised concerning the licensee's analysis such that one cannot conclude with absolute certainty that no new degradation exists. Confirmatory analysis of pulled tube samples will answer some of these questions but not all of them. However, in a January 1985 affidavit to the ASLAB, the NRC staff stated its agreement with the licensee's initial conclusion based upon being reasonably confident that the corrosion mechanism had been identified and arrested. The ASLAB agreed with the licensee's and staff's conclusion. ALAB-807, 21 NRC at 1206-10.

Based on the analysis as a whole, the staff sees no evidence to change its previous agreement with the licensee's results. Although the staff cannot reach its conclusion with absolute certainty, it does have reasonable assurance based on the facts presented to date that the new indications were the result of previously identified corrosion which has stopped. These facts include extensive laboratory analysis conducted by the licensee where a majority of the results support the licensee's conclusion. TMIA has not produced evidence of new corrosion mechanisms which might cause new defects. TMIA did list instances where OTSG chemistry was temporarily out of its normal range; but as indicated in other portions of this Safety Evaluation, the staff does not judge those temporary periods to be of significance.

A specific concern raised by TMIA (TMIA Comments at 7) questioned the effects of radioactive deposits building up on the outer diameter of the steam generator tubes as indicated in Inspection Report 85-27. Inspection Report 85-27 discusses the inability of the plant to achieve 100% rated power, at that time, due to a high steam generator water level. This problem was the result of fouling of heat transfer surfaces on the secondary side of the steam generator. The report does not refer to this buildup as "radioactive deposits" but since there is a measurable, but very small, amount of primary to secondary leakage, there is the possibility of some radioactivity in the deposits. However, where this radioactivity exists, it originates in the primary system and is chemically compatible with the steam generator material. The deposits originate from minute corrosion from components in the secondary system (such as feedwater preheaters). It concentrates in the steam generators because it is heavy enough not to be carried away by the steam. This phenomenon is not unique to TMI-1 or the B&W design. However, there is no data to indicate that this corrosion product buildup created a corrosive environment in the steam generators. In addition, this fouling is on the secondary side, and the amendment request is for defects located on the primary side.

Another specific question raised by TMIA (TMIA Comments at 7) involved the use of Furmanite to repair a leak in a steam generator component. Because of problems in the repair process, some Furmanite was blown into the steam generator. Initially, the licensee could not answer questions posed by the Resident Inspector concerning the effects of Furmanite on the steam generator. However, contrary to the comment made by TMIA, restart of the plant was not permitted until an engineering analysis was completed on the effects of Furmanite. The licensee provided the required engineering analysis. It should be noted that Furmanite is a widely used substance in the industry. Its chemical composition and effects are well known. It does not cause unacceptable chemical or corrosion problems in the steam generator.

TMIA also states (TMIA Comments at 9-10) that there appears to be significant disagreement among the NRC staff's technical experts as to whether this amendment request is "missing a piece" because tube samples were not obtained. As in most technical reviews, the staff begins with many questions to be answered and concerns to be resolved. One of the major considerations involved in this amendment is that it will be effective only for a limited period of time (approximately 1/2 cycle). Thus, many potential long term effects have less significance. Based on this consideration, the staff does not require a tube sample for this amendment. The staff does consider tube samples important for a permanent increase in the repair limit for the OTSG.

B. The Requested License Amendment Relies on Unique Eddy Current Testing Methods for Which There is Little or No Industry Experience to Verify Their Accuracy (TMIA Comments at 10-16)

TMIA questions the ability of eddy current testing (ECT) to adequately detect IGA in the absence of grain dropout. There is no question that current state of the art eddy current testing capability has problems detecting intergranular cracks that may initiate and grow from the bottom of IGA patches. This problem is involved with the basic ability of eddy current testing to initially detect a crack. The level of overall protection is not changed with a 40% or a 50% repair limit. Once a defect is detected, the masking may have some effect on the accuracy of the measurement. However, as indicated in other portions of this Safety Evaluation, sufficient data exist to conclude, on a conservative basis, that eddy current accuracy is 10% in the areas of interest. The staff used a 10% eddy current accuracy in their independent calculations in this Safety Evaluation. The staff is reasonably confident in the adequacy of this 10% value, particularly over the relatively short time interval for which this amendment would apply. Additionally, for the current 40% throughwall TS limit in effect on TMI-1, the ECT uncertainty is 10%. Thus the proposed ECT accuracy does not represent a change in the current licensing basis for the plant.

Before considering a permanent change to the technical specifications, the staff does desire tube samples to be drawn and analyzed. These samples will provide a better understanding of the masking problems caused by IGA, will serve to confirm the conclusion that the cause of the degradation is arrested, and will confirm assumptions made on eddy current accuracy. The licensee has committed to drawing tube samples at the next refueling outage and the staff will consider a permanent technical specification change at that time.

TMIA further questions eddy current testing techniques in general and the licensee's ability to measure inner diameter defects as compared to outer diameter defects. (TMIA Comments at 14-15). The licensee is using current state of the art eddy current techniques as used throughout the industry. These techniques are capable of differentiating between defects located on the inner diameter and outer diameter of the tubes. The traditional curves used in this analysis were designed for the more common outer diameter defects found throughout the industry. The licensee extrapolated a new curve to be more accurate for inner diameter defects using data obtained in laboratory testing. The technique used by the licensee to establish the new curve was verified by the licensee using standard laboratory techniques. The methodology used to establish the new curve is acceptable to the staff even if the curve has little "operational history."

TMIA further questions the ability of the licensee to measure defect length as there will be some uncertainty of length indications. (TMIA Comments at 14-15). The staff agrees that there are questions regarding the accuracy of defect lengths as measured by eddy current testing. Eddy current methodology is not an exact science and there will be a probability distribution surrounding any measurement. However, in the staff's independent calculations, no limit was placed on defect length. Basically the defect was assumed to be of infinite length. As indicated in this Safety Evaluation, the staff's independent calculations regarding the adequacy of an infinite length defect of 50% throughwall penetration is based on the same safety margins for tube integrity as the existing 40% plugging limit. By attempting to limit defect length to a relatively small value which can be measured with some degree of confidence, the licensee is providing a more restrictive limit than was considered in the staff's evaluation.

C. The Requested Amendment Violates Regulatory Guide 1.121 (TMIA Comments at 16-18)

The basic thrust of TMIA's comment is that the amendment violates Regulatory Guide 1.121 in that it does not consider the effects of environmental corrosion or possible wear on the tubes due to operation.

First, it should be noted that Regulatory Guides are not requirements on the licensee. Regulatory Guides provide an acceptable methodology for achieving the requirements found in General Design Criteria (GDC). The licensee may use other methods to meet the GDC requirements.

Regulatory Guide 1.121 provides guidance on steam generator tube plugging criteria. It was written to cover all environments, all tubes and all size defects in a generalized and conservative manner. When a specific application is reviewed, portions of the regulatory guide may not be applicable because of the unique circumstances in the application.

As described in other portions of this Safety Evaluation, the staff concludes that over the time interval involved in the amendment and considering the specifics of the amendment request, environmental corrosion on the primary side of the tubes will be negligible as no active corrosion mechanism has been determined. Thus it is acceptable for this amendment not to consider the effects of environmental corrosion on the primary side.

As far as concerns about possible wear on the tubes due to general operation, the staff notes that the plant was licensed for 40 years of operating life. There is presently no known operational wearing process at TMI-1 which would drastically reduce the tube wall thickness in one half cycle of operation, which is the effective time interval of the amendment.

D. The Proposed Criteria Fail to Comply With General Design Criterion (GDC) 31 (TMIA Comments at 18)

The basic comment from TMIA is that GDC 31 is not met because environmental effects have not been taken into account in the amendment request. As discussed in other portions of this safety evaluation, the environmental effects have been considered over the effective time interval of the amendment (one half cycle). For this limited duration amendment, the staff concludes that GDC 31 is satisfied.

E. The Proposed Criteria Fail to Comply with General Design Criterion (GDC) 32 (TMIA Comments at 19)

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," provides that components of the reactor coolant pressure boundary must be designed to permit 1) inspection and testing of important features and 2) an appropriate material surveillance program for the reactor pressure vessel. TMIA contends leakage from larger cracks will not be detected before rupture occurs and this will violate GDC 32.

The proposed criteria have no effect on present primary to secondary leak rate criteria or leakage detection capability. The leak rate requirements at TMI-1 will detect small amounts of leakage. There is no basis to support the TMIA conclusion that leakage from larger cracks will not be detected before rupture occurs.

F. The Proposed Amendment Contains None of the Assurance Required by the Staff (TMIA Comments at 19-20)

TMIA contends that the staff desired two assurances concerning this amendment that were not met: specifically, a commitment to pull tube samples at the next refueling outage and an undefined acceptability of the March 1986 inspection results.

By letter dated February 19, 1986, the licensee committed to perform laboratory analysis on portions of up to three tubes presently in service at the next refueling outage, if they are capable of pulling tubes then. The licensee must develop unique tube pulling criteria because of previous repair techniques approved in 1984. This commitment is appropriately not a part of the amendment requested.

The "March 1986" inspection outage is currently in progress. The staff has been monitoring results to date and, so far, there is no data substantially beyond what would be expected to see in the inspection. The staff will continue to monitor the inspection results but does not intend to "conditionally" approve an amendment. Based upon the results seen to date, the staff approves the licensee's request. If drastic unexpected results are obtained in the final analysis of the in progress eddy current inspection, the NRC staff will take appropriate independent action.

G. Legal Discussion (TMIA Comments at 20-22)

TMIA provides a discussion as to why the amendment does not legally meet the requirements of 10 CFR 50.92, principally in the area of a significant reduction in margins of safety. As indicated in the Final No Significant Hazards Determination, the amendment does not result in a significant reduction in margins of safety.

Additional Question

While this amendment request was under review, a question concerning this request was raised by Congressman Markey in a letter to the NRC dated March 27, 1986. The question posed was:

The failure of General Public Utilities Nuclear to pull an actual tube from the Three Mile Island Unit 1 steam generators for inspection since new indications have been discovered has reportedly left the staff uncertain as to the exact nature of new tube degradation and whether any current testing is reliable. Because no such actual tests exist, the NRC staff did not approve of licensee's request to change its plugging criteria to up to 70 percent throughwall. In light of these continuing uncertainties, what is the technical basis for NRC's decision for deciding that no significant hazard existed for modifying the plugging criteria to up to 50 percent throughwall. Additionally, has NRC approved plugging criteria of other than 40 percent in any other pressurized water reactor?

THe NRC Staff response stated:

For clarification, the NRC's reasoning for deciding that no significant hazards considerations existed for modifying the steam generator tube repair criteria at Three Mile Island Unit 1 (TMI-1) to allow for operation with up to a 50% throughwall penetration relates to the very restrictive set of conditions which apply for these repair criteria. Specifically, (1) the defect must be located on the primary side of the steam generator tube, (2) the defect location must not be in an area of reduced eddy current detection sensitivity (upper and lower tube sheet secondary faces and support plate entry and exit locations in the primary side), (3) the defect can have no greater than a 50% throughwall penetration, (4) if the defect has greater than 40% throughwall penetration, the defect must be less than 0.55 inches long, and (5) the amendment is only effective until the next refueling outage scheduled for December 1986.

The technical basis for our determination that the proposed 50 percent throughwall plugging criteria amendment involves no significant hazards considerations is that:

(1) Adequate documentation exists to show that tube integrity at TMI-1 is maintained within existing safety margins with a defect of up to 60% throughwall penetration absent any consideration of or allowance for environmental effects or detection technique accuracy. This analysis is independent of defect length.

(2) On the primary side of the steam generator tubes, the staff has reasonable assurance that the active corrosion mechanism has been arrested. Thus for the limited time interval involved in the 50 percent throughwall amendment, we anticipate no further tube wall thickness degradation from environmental effects. (However before a permanent

long-term change to the plugging criteria is considered, we believe it prudent to obtain tube samples for analysis to confirm once and for all that the active corrosion mechanism has been permanently arrested.)

(3) The staff has reasonable data to conclude that the eddy current detection accuracy in the areas where the 50 percent throughwall criterion is proposed for use is within 10%. Thus, a defect indicated by eddy current testing to be 50 percent throughwall may actually be as much as 60 percent throughwall. (The staff does desire confirmatory analysis of pulled tubes to more accurately determine eddy current accuracy for consideration of long-term changes to tube plugging criteria proposed by licensee.) Therefore, a 10% wall thickness should be allowed to account for eddy current accuracy.

(4) Utilizing the above findings, the staff has reasonable data to conclude that for TMI-1 steam generators, a defect which meets the limitations of the proposed 50 percent throughwall amendment (i.e., defect is on primary side of tube; defect is not in area of reduced eddy current detection sensitivity; defect is no more than 50 percent throughwall and, if defect is greater than 40 percent throughwall, its length is less than 0.55 inches) is bounded by current safety margins applicable to the present 40% throughwall and unlimited defect length criterion. Being thus bounded by the present licensed plugging criterion, the proposed 50 percent throughwall criterion would not involve a significant increase in the probability or consequences of an accident previously evaluated, would not create the possibility of a new or different kind of accident from any accident previously evaluated, would not involve a significant reduction in a margin of safety and, therefore, would not involve significant hazards considerations.

In addition, the licensee has committed to obtaining tube samples for confirmatory analysis at the next refueling outage (December 1986) - this analysis will be used to support a permanent technical specification change. It should be noted that the staff had previously noticed for a prior hearing the licensee's earlier amendment request which would allow operation with up to a 70% throughwall defect. In that instance the staff's inability to find no significant hazards consideration was based on inadequate safety margins with the use of that plugging limit and the issue of a tube sample analysis did not enter into the staff's determination.

The generic 40% plugging criteria is a conservative value based on a tube defect of unlimited size. The NRC has approved higher tube plugging limits on pressurized water reactors where the particular corrosion mechanism is known to result in pits rather than cracks or where the tube is thicker than the normal size. Indian Point 3 operated with a 63% plugging limit for part of a cycle and with a 50% plugging limit for almost an entire cycle. Indian Point 3 currently has a 40% plugging limit. Haddam Neck has a permanent 50% plugging limit. These plugging limits at Indian Point 3 and Haddam Neck were granted after detailed staff review. For the amendment proposed for Three Mile Island Unit 1, the licensee's request applies to the primary side of the tubes only. Since the corrosion mechanism has ceased on the primary side for the present, a higher plugging limit is acceptable until the next refueling outage.

FINAL NO SIGNIFICANT HAZARDS DETERMINATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's regulations, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The event of concern for this amendment is a steam generator tube rupture. The proposed criteria provide assurance of OTSG tube wall integrity under normal operating and faulted conditions. In particular, the proposed amendment has been verified to satisfy the recommendations of Regulatory Guide 1.121 in that it contains a margin of safety against ductile failure equal to 3.0 times normal loads. The staff concludes that the conditions established by the proposed amendment are bounded by the same safety margins that exist for the current approved 40% tube plugging limit. Thus, use of the proposed criteria does not involve a significant increase in the probability of occurrence of a steam generator tube rupture event or any other accident previously evaluated. In addition, since an ample margin against ductile failure is maintained, the consequences of previously evaluated accidents are not significantly increased by this amendment.

(2) Use of the proposed criteria would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Steam generator tube plugging criteria have an effect on steam generator tube rupture events and main steam line break events. Both events were considered in the acceptance criteria for steam generator tube plugging. Use of the proposed criteria has no bearing on other accidents and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Use of the proposed criteria would not involve a significant reduction in a margin of safety.

The margin of safety for the proposed revised criteria is in accordance with the licensing basis for the existing repair limit. The limiting margin of safety previously approved by NRC is not affected or reduced. The margin separating the proposed revised criteria from the analytical results for normal operating and faulted conditions is in accordance with the guidelines of Regulatory Guide 1.121 and is not significantly reduced by this amendment. Thus the staff concludes that use of the proposed criteria does not involve a significant reduction in a margin of safety.

Based on our review of the licensee's submittal, as described in our above evaluation, we have made a final determination that the amendment request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) this amendment does not involve significant hazards considerations, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 18, 1986

Principal Contributors: J. Rajan, J. Thoma

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION, ET AL.

DOCKET NO. 50-289

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 116 to Facility Operating License No. DPR-50 issued to GPU Nuclear Corporation (the licensee), which revised the Technical Specifications for operation of Three Mile Island Nuclear Station, Unit 1 (the facility), located in Dauphin County, Pennsylvania. The amendment is effective as of the date of issuance.

This amendment revises the repair limits for the steam generator tubes under a very restrictive set of circumstances as described in the request. Basically, for certain defects located on the primary side of the tubes, the amendment changes the mandatory repair limit from 40% to 50% throughwall penetration providing the defect is less than 0.55 inches long. The amendment is also only effective until the next refueling outage at which time the steam generator tube repair criteria will be re-evaluated.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

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Notice of Consideration of Issuance of Amendment and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER (51 FR 7157) on February 28, 1986. A request for a hearing was filed on March 10, 1986, by Three Mile Island Alert, Inc.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any persons, in advance of the holding and completion of any required hearing, where it is determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any hearing will be held after issuance.

The Commission has determined that this amendment satisfies the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for this amendment.

For further details with respect to this action see (1) the application for amendment dated February 4, 1986, (2) Amendment No. 116 to Facility Operating License No. DPR-50 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 18th day of April 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B