50-289

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 1999

Mr. James W. Langenbach, Vice President
and Director, TMI
GPU Nuclear, Inc.
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ONCE THROUGH STEAM GENERATOR INSPECTION CRITERIA FOR THE CYCLE 13 REFUELING EXAMINATIONS (TAC NO. MA4234)

Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No. 209 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) in response to your application dated November 25, 1998, as supplemented February 12, 1999.

The amendment approves the requested surveillance Technical Specifications (TSs) related to the once through steam generator inservice inspections to be completed during the 13R refueling outage in fall 1999. Related TS bases changes are also included. The staff notes that your February 12, 1999, letter withdrew the portion of your request to extend the reporting period from 90 days to 12 months.

The staff notes that this request was for one cycle only even though a similar request had been reviewed for the 12R refueling outage. The staff would not, without a compelling reason, again review another one-cycle request for these inspection surveillance criteria.

The staff further notes that this request, with others, was identified as required to be completed in order to support the sale and license transfer of TMI-1 to AmerGen. Neither the December 3, 1998, application for that request, nor the November 25, 1998, application related to this request provided a basis for a higher priority accelerated review to support the April 15, 1999, proposed transfer date. The staff has consented to support your requested schedule; however, the information stated above may be used as a Plant Issues Matrix (PIM) entry by Region I when that document is forwarded to you following completion of its semi-annual performance review of your facility.

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210056

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. A Notice of Withdrawal is also enclosed.

Sincerely,

/s/

Timothy Colburn, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 209 to DPR-50
2. Safety Evaluation
3. Notice of Withdrawal

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Sincerely,

Timothy H. Colburn

Timothy Colburn, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
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Docket No. 50-289

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Three Mile Island Nuclear Station, Unit No. 1

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DATED: April 13, 1999

AMENDMENT NO. 209 TO FACILITY OPERATING LICENSE NO. DPR-50 THREE MILE ISLAND

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METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR, INC.

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear, Inc., et al. (the licensee) dated, November 25, 1998, as supplemented February 12, 1999, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 209 , are hereby incorporated in the license. GPU Nuclear, Inc. shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: April 13, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 209

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 areas of change.

Remove	Insert
4-79	4-79
4-80	4-80
4-81	4-81
4-82	4-82
4-83	4-83

4.19.2 Specification (Continued)

C-2 One or more tubes, but not more than 1% of the total tubes inspected in a steam generator are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected in a steam generator are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES: (1) In all inspections, previously degraded tubes whose degradation has not been spanned by a sleeve must exhibit significant increase in the applicable degradation size measurement (> **0.24** volt bobbin coil amplitude increase for inside diameter IGA indications or > 10% further wall penetration for all other degradation) to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.19.2.a.4, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.19.3 Inspection Frequencies

The required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first (baseline) inspection was performed after 6 effective full power months but within 24 calendar months of initial criticality. The subsequent inservice inspections shall be performed not more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group of tubes* encompassing not less than 18 calendar months all fall into the C-1 category or demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.19-2 at 40 month intervals for a given group of tubes* fall into Category C-3 the inspection frequency for that group shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.19.3.a; the interval may then be extended to a maximum of once per 40 months.

* A group of tubes means:

(a) All tubes inspected pursuant to 4.19.2.a.4, or

(b) All tubes in a steam generator less those inspected pursuant to 4.19.2.a.4

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.19-2 during the shutdown subsequent to any of the following conditions:

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

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

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

d. After primary-to-secondary tube leakage (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.6.3, an inspection of the affected steam generator will be performed in accordance with the following criteria:

1. If the leak is above the 14th tube support plate in a Group as defined in Section 4.19.2.a.4(1) all of the tubes in this Group in the affected steam generator will be inspected above the 14th tube support plate. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

2. If the leaking tube is not as defined in Section 4.19.3.d.1, then an inspection will be performed on the affected steam generator(s) in accordance with Table 4.19-2.

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 less than degraded tube criteria specified in a.3 below may be considered 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 :

(a) an inside diameter (I.D.) IGA indication with a bobbin coil indication $\geq$ **0.2** volt or $\geq$ 0.13 inches axial extent or $\geq$ 0.26 inches circumferential extent (for **operation through** Cycle **13** only), or

(b) imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

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

4.19.4 Acceptance Criteria (Continued)

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. For **operation through** Cycle **13** only, inside diameter IGA indications shall be repaired or removed from service if they exceed an axial extent of 0.25 inches, or a circumferential extent of 0.52 inches, or a through wall degradation dimension of ≥ 40% if assigned.

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.

9. Inside Diameter Inter-Granular Attack (IGA) Indication means a bobbin coil indication initiating on the inside diameter surface and confirmed by diagnostic ECT to have a volumetric morphology characteristic of IGA.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, of all tubes exceeding the repair limit and all tubes containing throughwall cracks) required by Table 4.19-2.

4.19.5 Reports

a. After the completion of each inservice inspection of steam generator tubes, prior to exceeding a reactor coolant system (RCS) temperature of 250 °F, the NRC shall be notified of the following:

   1) The number of tubes repaired or removed from service in each steam generator,

   2) An assessment of growth of inside diameter IGA degradation, and

   3) Results of in-situ pressure testing, if performed.

4.19.5 Reports (Continued)

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 90 days following completion of the inspection and repairs **(main generator breaker closure)**. The report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Location, bobbin coil amplitude, and axial and circumferential extent (if determined) for each inside diameter IGA indication, and

   4. Identification of tubes repaired or removed from service.

c. Results of steam generator tube inspections which fall into Category C-3 require notification in accordance with 10 CFR 50.72 prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence in accordance with 10 CFR 50.73.

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained.

The program for inservice inspection of steam generator tubes is based on modification of Regulatory Guide 1.83, Revision 1. In-service 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 the secondary coolant. However, even if a defect would develop in service, it will be found during scheduled in-service steam generator tube examinations. For tubes with ID IGA indications, additional conservatism is being applied **through** Cycle **13** operation, to evaluate circumferential and axial dimensions for determining final disposition of the tube. For ID IGA indications through wall dimension will continue to be assigned to those indications where amplitude response permits measuring through wall dimension. 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.

Removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness. For **operation through** Cycle **13** only, tubes with I.D. initiated intergranular degradation may remain in service without % T.W. sizing if the degradation morphology has been characterized as not crack-like by diagnostic eddy current inspection and the degradation is of limited circumferential and axial length to ensure tube structural integrity. Additionally, serviceability for accident leakage under the limiting postulated Main Steam Line Break (MSLB) accident will be evaluated by determining that this I.D. initiated degradation mechanism is inactive (e.g. comparison of the **13**R Outage examination results with the results from past outages does not show growth greater than expected ECT repeatability variations) and by successful **13**R in-situ pressure testing of a sample of these degraded tubes to evaluate their accident leakage potential **when in-situ pressure tests are performed.**

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.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 on the first sample inspection (See Table 4.19.2), these results will be reported to NRC pursuant to the requirements of Specification 4.19.5.c. Such cases will be considered by the NRC on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.

Note: The eddy current examination voltages referred to in this section (section 4.19) are based on a normalization procedure that sets the bobbin coil prime frequency peak-to-peak response from the four 20% through-wall holes of an ASME calibration standard to 4 volts.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR, INC.

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated November 25, 1998, as supplemented on February 12, 1999, GPU Nuclear, Inc., (the licensee) submitted a request to change the technical specifications (TS) at the Three Mile Island Nuclear Station, Unit 1 (TMI-1). The TS change request proposed the continuation of an alternate repair criteria, previously approved through Cycle 12, to address intergranular attack (IGA) degradation identified on the inside diameter (ID) of the TMI-1 once-through steam generator (OTSG) tubes. The repair criteria would be used in the Cycle 13 Refueling (13R) examinations and would be in effect until the end of operating cycle 13. The TS change request also proposed changes to related TS eddy current testing (ECT) voltage limits and a clarification that the TS reporting period for steam generator inservice inspection reports begins with the closure of the plant's main electrical generator breaker. The November 25, 1998, letter requested extension of the steam generator inspection reporting period from 90 days to 12 months. The February 12, 1999, supplemental letter withdrew that request. The supplemental letter did not affect the initial "proposed no significant hazards consideration" determination.

2.0 BACKGROUND

In November 1981, while performing reactor coolant system hydrostatic testing with the reactor shut down, primary-to-secondary system leakage was detected in both OTSGs. Subsequently, eddy current examinations revealed many defective tubes. Metallographic examination of portions of removed tubes confirmed that the tube degradation initiated from the primary side of the tubes in the form of circumferential stress-assisted intergranular cracks. The active chemical impurity causing the corrosion was sulfur in reduced forms, which had been inadvertently introduced into the reactor coolant system. The vast majority (approximately 95 percent) of the defects occurred within the top 2 to 3 inches of the 24-inch thick upper tubesheet (UTS). The corrosion attacks most rapidly at the air/water interface and during lay up. The air/water interface was located in the UTS during a significant portion of the post-hot-functional shutdown period. To repair the defective OTSG tubes within the UTS, the licensee applied a kinetic (explosive) expansion repair technique. The staff previously reviewed and approved the licensee's repair of the OTSG tubing in NUREG-1019, "Safety Evaluation Report Related to Steam Generator Tube Repair and Return to Operation -- Three Mile Island Nuclear Station, Unit No. 1," dated November 1983.

The kinetic expansion repair technique applied in the early 1980's addressed the existence of defects located in tubes in the UTS. However, a limited population of tubes in the TMI-1 OTSGs contained degradation located below the UTS secondary face that could not be repaired by the kinetic expansion technique. Because of the uncertainty in sizing the depth of ID IGA degradation that was not previously repaired, the NRC and the licensee agreed during a meeting held in Rockville, Maryland, on July 15, 1997, that the tube repair criteria in the TMI-1 TS should be amended to address tubes identified with this mode of degradation.

Subsequently, the licensee proposed TS repair criteria to be used through the Cycle 12 operating cycle, which addressed ID IGA below the UTS. The repair criteria supplemented the existing TS depth-based tube repair limit (i.e., 40 percent through-wall) with ECT voltage-based and length-based repair limits. The length-based repair limits do not account for flaw growth over the cycle because the licensee has concluded the ID IGA flaws are inactive. In addition to the proposed repair limits, the licensee proposed modifications to the reporting requirements for steam generator tube inspections to include the submission of a summary of steam generator inspections and results to the NRC within 90 days following completion of the inspections and repair. The proposed TS repair criteria were reviewed and approved in a staff safety evaluation dated October 16, 1997 (Reference 1).

The licensee has proposed the continuation of the TS repair criteria, previously approved through Cycle 12 as documented in Reference 1, through the end of Cycle 13. The licensee has also proposed two other related changes. The staff's evaluation of this proposal considers leakage and structural integrity margins, growth rate analysis of ID IGA indications, and eddy current technique capability of reliably implementing the length-based repair criteria.

3.0 EVALUATION

3.1 Extension Through Cycle 13

The licensee has proposed continuation of the alternate repair criteria [TS 4.19.4.a.3(a), TS 4.19.4.a.6, TS 4.19.5.b and TS Bases], previously approved through Cycle 12, to address ID IGA identified in TMI-1 steam generator tubes. The repair criteria would be extended to be in effect during 13R steam generator examinations and through the end of operating cycle 13. In addition to the submittals made as part of this license amendment request, the staff reviewed several other documents containing relevant information. These are listed as References 2 through 4. The staff reviewed leakage integrity, structural integrity, growth rate analysis and inspection capability to assess the effect of extending the repair criteria through Cycle 13.

3.1.1 Leakage Integrity

In order to demonstrate leakage integrity margins, the licensee completed in-situ pressure testing, during 12R, of steam generator tubes with the bounding ID IGA indications to demonstrate a low leakage potential for tubes containing this mode of degradation. In addition, supplemental laboratory leak tests were performed on pulled tubes in order to effectively simulate both peak hoop and axial stresses which might be applied to tubes containing volumetric flaws during a postulated main steamline break (the in-situ test equipment used during 12R was not able to fully simulate these stresses on steam generator tubes). None of the indications tested in-situ or in the laboratory leaked at pressures simulating up to three times the normal steam generator tube operating delta pressure. Therefore, the licensee concluded that no increase in accident-induced leakage would be expected due to the presence of ID IGA flaws.

During 13R, the licensee plans to select tubes for leak testing if any have indications of degradation which appear more limiting with respect to leakage integrity than those which were tested during 12R, thereby continuing to assess the potential for leakage from all degraded tubes. Assuming the limiting tubes retain leakage integrity throughout the test, it is reasonable to conclude that other, less degraded tubes in the population would also have sufficient integrity to withstand accident-induced loads without failure. The licensee indicated that supplemental laboratory leak testing will not be necessary during 13R because the in-situ pressure test equipment that will be used has been modified since 12R and can effectively simulate both peak hoop and axial stresses. On these bases, the staff concludes the licensee has demonstrated adequate leakage integrity margins in the presence of ID IGA flaws.

3.1.2 Structural Integrity

The primary method of ensuring structural integrity of ID IGA flaws is the application of the TS length-based repair criteria which were determined based on structural analyses of flawed steam generator tubing assuming through-wall defects of the limiting length. In order to demonstrate continued structural integrity, the licensee completed in-situ burst testing, during 12R, of steam generator tubes with bounding ID IGA indications. In addition, supplemental laboratory burst tests were performed on pulled tubes in order to effectively simulate both peak hoop and axial stresses which might be applied to tubes containing volumetric flaws during a postulated main steam line break (the in-situ test equipment used during 12R was not able to fully simulate these stresses on steam generator tubes). None of the indications tested in-situ or in the laboratory burst at pressures simulating up to three times the normal steam generator tube operating delta pressure. In fact, the burst pressures of the tube sections containing ID IGA that were tested in the laboratory were above 10,000 psig. This indicates that substantial structural margin exists for these flaws since the burst pressure for tubes with no defects averages 11,216 psig.

During 13R, the licensee plans to select tubes for burst testing if any have indications of degradation which appear more limiting with respect to structural integrity than those which were qualified by tests during 12R, thereby continuing to assess the structural integrity of tubes with ID IGA flaws. The licensee indicated that supplemental laboratory burst testing will not be necessary during 13R because the in-situ pressure test equipment that will be used has been modified since 12R and can effectively simulate both peak hoop and axial stresses. On these bases, the staff concludes the licensee has demonstrated, and will continue to demonstrate, adequate structural integrity margins in the presence of ID IGA flaws.

3.1.3 Growth Rate Analysis

The licensee has previously demonstrated that ID IGA degradation is not an active degradation mechanism at TMI-1. This was demonstrated through comparisons of nondestructive examination data from several outages. This is a significant aspect of the technical basis supporting this amendment request because the length-based repair criteria were developed based on the assumption of zero flaw growth. Therefore, the staff has a heightened sensitivity to the results of the licensee's growth rate analysis. The licensee's 12R growth rate analysis was based on a review of eddy current data and metallurgical and chemical analysis performed on the 12R pulled tubes and is discussed in the following paragraphs.

The licensee performed assessments of volumetric ID IGA growth rates based on changes in bobbin coil voltage, motorized rotating pancake coil (MRPC)-indicated axial length and MRPC-indicated circumferential length. When the 12R bobbin coil voltages were compared to previous voltage responses at the same location, a +0.1 volt average increase (0.26 volt standard deviation) was observed. These voltage changes are comparable with voltage variations from seven previous outages and indicated no statistically significant growth. MRPC-indicated axial and circumferential length measurements were made for all ID IGA indications detected during 12R. These values were compared to measurements from examinations performed during the previous two outages (where available). The licensee determined there was essentially no change in axial or circumferential dimensions for the values compared. The dimensions used for this comparison were obtained using the same type of eddy current probe for all three outages.

The licensee described the ECT equipment and calibration techniques used in 11R and 12R and planned for 13R. The licensee discussed differences between outages and the expected effects on the ECT data and growth rate analysis, and concluded the differences would not adversely affect flaw growth studies because procedures were implemented to assure consistency. Two new practices were implemented in 12R which provide enhanced consistency: bobbin coil probe wear is now being monitored for examinations where ID IGA is being evaluated; and a Babcock and Wilcox Owners Group (BWOG) Mother ASME standard is being utilized for calibrations (versus individual calibration standards) in order to eliminate subtle voltage differences between calibration standards.

The licensee performed chemical and metallurgical analyses on the 12R pulled tube to supplement the growth rate analysis. The following procedures were utilized specifically to investigate the micro chemistry of the volumetric ID IGA (pit-like) degradation found in the pulled tube: X-ray Photoelectron Spectroscopy, Scanning Auger Microprobe and Scanning Electron Microscopy with Energy Dispersive Spectroscopy. The strongest evidence derived from these analyses that supports the historical nondestructive examination record of inactivity of the ID degradation is the absence of aggressive chemical species and the absence of metallographic evidence of any new or different corrosion mechanisms. These analyses did not reveal unusual surface chemistry or reduced forms of sulfur, such as that which caused the original ID damage. In addition, the analyses did not identify other contaminants in amounts that have been linked to steam generator tube pitting-type degradation. Microscopic studies of the IGA pits confirmed their volumetric pit-like geometry, which had been characterized during the field ECT exams. The ID IGA defect characterization of the 1997 tubes was similar to that of the ID IGA defects of tubes pulled in 1986. If this IGA were continuing to propagate, as was observed in the 1981 degradation, it would be expected to ultimately develop into stress corrosion cracks (SCC). There was no evidence of ID-initiated SCC at the ID IGA flaws examined in the laboratory by metallographic grinding or microscopy.

Based on eddy current data comparisons, measures taken to validate comparisons of ECT data between outages, and chemical and metallurgical analyses, the staff concluded the licensee's growth rate analysis supports implementation of the ID IGA repair criteria for another cycle.

3.1.4 Eddy Current Inspection Capability

During 12R, a steam generator tube was pulled that contained several volumetric ID IGA flaws. The purpose of the tube pull was, in part, to demonstrate the inspection capability by confirming the ability of rotating pancake coil eddy current to conservatively estimate axial and circumferential extents of flaws in the field. The licensee compared axial and circumferential lengths based on eddy current tests in the field, to axial and circumferential lengths based on destructive examination results. The licensee found that in all cases the eddy current measurements were conservative relative to actual lengths. Based on this, the licensee concluded that eddy current probes (i.e., MRPC probes) are able to conservatively assess the axial and circumferential extents of TMI-1 ID IGA flaws. On this basis, the staff concludes the eddy current technique (MRPC) is capable of adequately implementing the TS length-based repair criteria.

3.1.5 Conclusions

Based on the four criteria discussed above (leakage integrity, structural integrity, growth rate analysis, and inspection capability), the staff concludes that extension of the repair criteria through the end of Cycle 13 is acceptable and will not affect public health and safety.

3.2 Voltage Limits

The licensee has proposed to revise two voltage limits [TS 4.19.4.a.2 Note 1 and TS 4.19.4.a.3(a)] which apply to the ID IGA repair criteria. This revision is a result of a proposed change in voltage normalization for bobbin probe examinations. The licensee has used a 10-volt normalization for the steam generator tube bobbin coil eddy current examinations since the tube damage that occurred in 1981. Other nuclear plants have also used this normalization; however, a greater number of plants have adopted a 4-volt normalization. The licensee has proposed to convert to the 4-volt normalization, which is becoming standard in the industry. The licensee indicated that this change will allow a more direct comparison between TMI-1 eddy current data and that of industry. This change will affect TS sections in which an eddy current voltage is specified. In effect, the measured voltages representative of the eddy current signals under the proposed 4-volt normalization will be 40 percent of the measured voltages under the 10-volt normalization. This voltage normalization change is applicable for the bobbin probe examinations only, and does not affect signal quality or phase angle analysis of the bobbin probe eddy current signals, rotating probe eddy current, or other aspects of eddy current signal analysis. The licensee has also stated they will adjust all previous voltage measurements obtained under the 10-volt normalization to a voltage measurement that would have been obtained under a 4-volt normalization in order to make comparisons of 13R voltages to prior outage voltages equivalent. On these bases, the staff concludes the proposed TS change to voltage limits is acceptable.

3.3 Clarify Event Signifying End of Outage's Steam Generator Inspection and Repair Work

The existing TS for TMI-1 state that the licensee will submit to the NRC complete results of the steam generator tube inservice inspection within 90 days following completion of the inspection and repairs. The licensee has proposed to modify this requirement (TS 4.19.5.b) by clarifying that inspection and repairs are considered complete when the main generator breaker is closed. This modification will not impact the staff's ability to adequately assess any steam generator tube integrity issues stemming from the inspection in a reasonable period of time. Therefore, the staff concludes the proposed change to specify that a complete inservice inspection report is due 90 days from main generator breaker closure (i.e., restart) is acceptable.

4.0 FUTURE CONSIDERATIONS

The licensee stated they submitted this TS change request as a request for a one-cycle change primarily because the upcoming outage (13R) will be the second of two consecutive outages in which a very large number of ID IGA flaws will have been examined using the same eddy current testing techniques. 12R provided a 100 percent bobbin probe examination with follow-up MRPC examinations of all of the ID IGA flaws identified by the bobbin probe. In previous outages, less than 100 percent examination of all ID IGA flaws by MRPC was performed. 13R will provide the same scope as 12R, using the same eddy current acquisition equipment that was used in 12R. The results from these two outages, taken together, will provide a significant amount of eddy current flaw length data with which to analyze the growth rate of the ID IGA flaws. The licensee expects this data should further substantiate the insignificant growth rate of ID IGA flaw lengths already seen.

Additionally, the licensee stated that several regulatory and industry initiatives are currently being considered for implementation. The industry and NRC staff are actively working together to achieve agreement on new steam generator tube integrity requirements, which the licensee believes may require revision of current TS requirements related to steam generators. It is expected that these initiatives and the required supporting programs and criteria will be in a more mature status before the next TMI-1 outage (14R), and, therefore, it is a more appropriate time for the licensee to submit a request for a permanent amendment.

The staff strongly encourages the licensee to pursue the development of a qualified eddy current technique which can reliably depth size ID IGA in accordance with the original 40 percent tube repair limit. If this path were pursued, further TS amendments would not be required to address this mode of degradation. As documented in Reference 5, the Nuclear Energy Institute (NEI) indicated that each licensee would evaluate its existing steam generator program and, where necessary, revise and strengthen program attributes to meet the intent of the guidance provided in NEI 97-06, Steam Generator Program Guidelines, no later than the first refueling outage starting after January 1, 1999. NEI 97-06 states that each utility shall follow the inspection guidelines contained in the latest revision of the EPRI PWR Steam Generator Examination Guidelines (Reference 6). Reference 6 contains guidelines for developing eddy current techniques which are qualified for detecting and sizing steam generator tube degradation.

Regardless of which path is pursued, NRC staff previously identified in Reference 1 areas of weakness in the licensee's ID IGA growth rate study. A number of variables were identified which were not specifically addressed in the growth rate studies and are as follows: (1) bobbin probe wear, (2) calibration practices and standards, (3) differences in data acquisition hardware, and (4) data analyst uncertainty. Based on submittals made for the Cycle 13 TS change request, it appears the licensee has addressed some of these factors for the current ID IGA repair criteria. The licensee should ensure that these are completely addressed in the future.

5.0 DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

In order to incorporate the proposed changes to permit continuation of the alternate repair criteria for ID IGA below the UTS, the licensee has proposed the following changes to the technical specifications.

a) Proposed change to TS 4.19.2 Note (1)

The threshold voltage increase at which a tube whose ID IGA indication is counted in the degraded tube population is revised from 0.6 volts to 0.24 volts. This revision is a result of the proposed change in voltage normalization for bobbin probe examinations.

b) Proposed change to TS 4.19.4.a.3(a)

The threshold voltage at which tubes whose ID IGA bobbin coil indication is considered degraded is revised from 0.5 volts to 0.2 volts. This revision is a result of the proposed change in voltage normalization for bobbin probe examinations.

The definition of a degraded tube is also revised so that the criteria pertinent to ID IGA indications are extended from the "12R Outage" and "Cycle 12 operation" to the "13R Outage" and "Cycle 13 operation" respectively.

c) Proposed change to TS 4.19.4.a.6

The definition of "repair limit" is revised to that the criteria pertinent to ID IGA indications are extended beyond Cycle 12 operation to the end of Cycle 13 operation.

d) Proposed change to TS 4.19.5.b

The parenthetical phrase "(main generator breaker closure)" is added to define this event as the start date of the reporting period for submittal of the complete results of the 13R steam generator tube inspections.

e) Proposed change to TS Bases section

Several references to "Outage 12R" are revised to indicate "Outage 13R" and references to "Cycle 12" operation are revised to indicate "Cycle 13" operation.

Discussion on in-situ pressure testing is modified to reflect that in-situ pressure testing <u>may be</u> performed during 13R.

A Note is added to the basis to reflect that TS voltage values related to ID IGA indications are based on a 4-volt normalization procedure.

Based on its review of the licensee's proposal, the staff has determined the proposed changes to the TMI-1 TS will continue to provide adequate assurances of steam generator tube integrity. The extension of the ID IGA repair criteria through the end of cycle 13 is acceptable based on the licensee's leakage and structural integrity analyses, growth rate analysis and eddy current inspection capability analysis. In addition, the changes to TS voltage values and the clarification regarding when steam generator inspection result reports are to be submitted to the NRC do not impact steam generator tube integrity or the staff's ability to assess steam generator tube integrity on a timely basis. Therefore, the staff has determined that the proposed modifications will not impact the public health and safety.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

8.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The amendment also changes reporting or recordkeeping requirements. The NRC staff has 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 69342). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

9.0 CONCLUSION

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

10.0 REFERENCES

1) Safety Evaluation by the Office of Nuclear Reactor Regulation Related to License Amendment No. 206 to Facility Operating License DPR-50, GPU Nuclear Corporation, Three Mile Island Nuclear Station, Unit No. 1, Docket No. 50-289, dated October 16, 1997.

2) Letter from J.W. Langenbach (TMI) to NRC, "TMI-1, NRC Notification on Completion of 12R Outage Steam Generator Examinations," dated December 17, 1997.

3) Letter from J.W. Langenbach (TMI) to NRC, "TMI-1, Cycle 12 Refueling (12R) Outage Once Through Steam Generator (OTSG) Tube Inspection Report with ASME NIS Data Reports for Inservice Inspections (ISI)," dated January 12, 1998.

4) Letter from J.W. Langenbach (TMI) to NRC, "TMI-1, Results from Cycle 12 Refueling (12R) Outage Pulled Tube Examinations," dated May 19, 1998.

5) Letter from Ralph E. Beedle (NEI) to L. Joseph Callan (NRC), dated December 16, 1997.

6) PWR Steam Generator Examination Guidelines, EPRI Report TR-107569 (Rev. 5, September 1997).

Principal Contributor: C. Beardslee

Date: April 13, 1999

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR, INC., ET AL.

DOCKET NO. 50-289

NOTICE OF WITHDRAWAL OF APPLICATION FOR

AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of GPU Nuclear, Inc., et al., (the licensee) to withdraw its November 25, 1998, application as supplemented by letter dated February 12, 1999, for proposed amendment to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, located in Dauphin County, Pa.

The proposed amendment would have, in part, extended the Technical Specification (TS) reporting interval in TS 4.19.5 from 90 days to 12 months.

The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the FEDERAL REGISTER on December 16, 1998 (63 FR 69342). However, by letter dated February 12, 1999, the licensee withdrew the proposed change request.

For further details with respect to this action, see the application for amendment dated November 25, 1998, as supplemented February 12, 1999, which withdrew that portion of the application for license amendment. The above documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Law/Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, P.O. Box 1601, Harrisburg, PA 17105.

Dated at Rockville, Maryland, this 13th day of April 1999.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn

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Project Directorate I
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Office of Nuclear Reactor Regulation