October 16, 1997

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

SUBJECT: THREE MILE ISLAND - ISSUANCE OF AMENDMENT RE: CHANGES TO THE TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENTS FOR ONCE-THROUGH STEAM GENERATOR INSERVICE INSPECTION FOR CYCLE 12 OPERATION (TAC NO. M99392)

Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No. 206 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 August 12, 1997, as supplemented August 28, September 15, October 3, 9, and 10, 1997.

The amendment changes the technical specifications that authorize the use of alternate tube repair criteria for once-through steam generator inservice inspection for Cycle 12 operation and makes other changes to surveillance requirements including reporting requirements.

This completes our effort on this issue and we are, therefore, closing out TAC No. M99392.

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

Sincerely,
Original signed by:

Bart C. Buckley, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures:
1. Amendment No. 206 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION: See attached page

DOCUMENT NAME: G:\BUCKLEY\M99392.AMD *See previous concurrence

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Three Mile Island Nuclear Station, Unit No. 1

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DATED: October 16, 1997

AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-50 THREE MILE ISLAND

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

METROPOLITAN EDISON COMPANY

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GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated August 12, 1997, as supplemented August 28, September 15, October 3, 9, and 10, 1997, 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. 206 , 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, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronald B. Eaton, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 16, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 206

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 (> 0.6 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.

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*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

4.19.3 Inspection Frequencies (Continued)

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 $\geq$ 0.5 volt or $\geq$ 0.13 inches axial extent or $\geq$ 0.26 inches circumferential extent (for 12R outage examinations and Cycle 12 operation 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 Outage 12R examinations and Cycle 12 operation 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**. 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.

Bases (Continued)

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 during the 12R Outage, for Cycle 12 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 the 12R Outage examinations and Cycle 12 operation 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 12R Outage examination results with the results from past outages does not show growth greater than expected ECT repeatability variations) and by successful 12R in-situ pressure testing of a sample of these degraded tubes to evaluate their accident leakage potential.**

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated August 12, 1997, as supplemented on August 28, September 15, October 3, 9, and October 10, 1997, GPU Nuclear, the licensee for Three Mile Island Nuclear Station, Unit 1 (TMI-1), submitted proposed changes to the Technical Specifications (TS). The TS change request proposed an alternate repair criteria 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 12 Refueling (12R) examinations and would be in effect until the end of the next operating cycle. The proposed changes impose restrictions on the axial and circumferential length of ID IGA degradation. The licensee would also complete in-situ pressure testing of a sample of tubes to demonstrate acceptable leakage integrity for tubes left in service with indications of ID IGA. The supplemental letters did not affect the initial 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 (ID) 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%) 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.

3.0 PROPOSED ALTERNATE TUBE REPAIR CRITERIA

The existing depth-based tube repair limit (40 percent through-wall) in the TMI-1 TS attempt to ensure adequate tube integrity through the end of the next operating cycle. However, the ability to accurately size the depth of service-induced IGA degradation with eddy current inspection techniques is complicated by a number of variables. Therefore, accurately dispositioning tubes with identified ID IGA degradation per the 40 percent depth repair limit is difficult using today's inspection methods. However, currently available eddy current examination technology does possess the capability to estimate the length of such degradation. The licensee's basis for assessing the structural integrity of tubes with ID IGA indications is founded on proposed length-based repair limits.

Although eddy current inspection techniques can measure the length of ID IGA degradation, these methods currently are not qualified to adequately assess tube leakage integrity margins. The licensee has proposed to use in-situ pressure testing to demonstrate that tubes would not leak under accident conditions. Based on previous inspection data, the licensee has concluded that the ID IGA degradation is dormant and will not continue to degrade over the next cycle of operation. This conclusion, considered in conjunction with the in-situ pressure test results, will be used to demonstrate adequate leakage integrity margins for the tubes with ID IGA indications through the end of the next operating cycle. Details of the licensee's proposed repair criteria are included in the following sections.

3.1 Inservice Inspection of Steam Generator Tubes

The inservice inspection scope for the TMI-1 Cycle 12 refueling outage includes an examination of 100 percent of the steam generator tubes full length with a bobbin coil eddy current probe. Tube examinations with this probe should identify all indications of ID IGA degradation that could potentially degrade the tube structural and leakage integrity margins. The bobbin probe, however, cannot assess the morphology or size of detected indications. Because the proposed repair criteria apply only to ID IGA degradation, the licensee will inspect all indications detected with a bobbin coil probe using a Plus Point rotating probe. The rotating probe examinations can confirm that the morphology of a bobbin indication is volumetric which is indicative of IGA. In addition, the data acquired in the Plus Point probe inspections can be used to assess the axial and circumferential length of confirmed tube degradation, and whether it initiated from the ID or outside diameter (OD) of the tube.

IGA degradation is characterized as a mode of degradation that is volumetric rather than crack-like in nature. That is, the degradation affects a small volume of tube material and typically has dimensions that extend axially, circumferentially, and radially (depth) in the tube. The ID IGA degradation should exhibit a morphology that extends both along the tube axis and around the circumference. Crack-like indications, however, extend primarily along only two tube directions (i.e., radial/axial, radial/circumferential). Because rotating probes are sensitive to degradation extending in both the axial and circumferential directions, these probes are capable of providing data to allow determination of whether an indication is crack-like or volumetric. In addition, they possess the capability to size the length of steam generator tube degradation. The licensee will inspect all indications detected by bobbin coil with a Plus Point probe. This will enable the licensee to confirm the mode of the degradation and determine the axial and circumferential length of the indication, if applicable.

Eddy current inspection techniques are capable of distinguishing the origin (i.e., ID or OD) of reflectors included in calibration standards and have been used successfully in the past to determine the initiating surface for steam generator tube degradation. The examination of each indication with the bobbin coil and Plus Point probes will provide complementary data to the analysts that will indicate whether the degradation is ID or OD in nature. The licensee will consider those signals with phase angles less than 30 degrees for the bobbin coil probe data to be indicative of ID degradation. Indications with signal phase angles less than approximately 26 degrees as measured with a Plus Point probe will be considered ID. Only indications which display a phase rotation in the ID flaw plane are considered ID indications. The TMI-1 inspection guidelines require all bobbin coil indications with phase angles outside the ID phase plane (i.e., ≥ 30%) to be assigned an NQI (i.e., non-quantifiable indication) code and thereby preclude the indication from the scope of this proposed repair criteria. In addition, the analysis of rotating probe data requires analysts to verify that the indications are ID in origin.

3.2 Structural Integrity Assessment

The proposed modifications to the TMI-1 TS include changes to the steam generator tube repair limits defined in TS 4.19.4.6, Repair Limit. The existing requirements specify that tubes shall be repaired or removed from service when degradation exists within the tube that is equal to or greater than 40 percent of the nominal tube wall thickness. In order to address the acceptability of tubes with confirmed ID IGA degradation, the licensee has proposed to impose additional limits on the dimensional extent of tube indications. Specifically, the circumferential and axial length of ID IGA tube degradation is limited to less than 0.52 and 0.25 inches, respectively. The length-based repair limits for the ID IGA degradation are based on structural analyses of flawed steam generator tubing assuming it contains through-wall defects of the limiting length. The licensee has concluded that tubes with defects at the proposed repair limit would retain their structural integrity under normal operating and postulated accident loadings.

TS 4.19.4.a.3.a currently defines a degraded tube as one with imperfections that exceed 20 percent in depth. A degradation depth of 20 percent is one-half of the depth-based repair limit of 40 percent through-wall. In order to remain consistent with this observation, the licensee has proposed to define degraded tubes as those with indications of ID IGA with lengths greater than or equal to 0.13 inches axial extent or 0.26 inches circumferential extent. In addition, a tube will be considered degraded if the bobbin coil voltage amplitude for an ID IGA indication is equal to or greater than 0.5 volts. These criteria are imposed in addition to the existing depth-based definition for degraded tubes.

The licensee will also complete in-situ pressure testing of a sample of steam generator tubes containing ID IGA indications. The sample of tubes selected for testing will include the most significantly degraded tubes (i.e., lowest expected burst pressure) as determined by an assessment of eddy current signal characteristics. The licensee will consider indication voltage, dimensional lengths, and phase angle depth measurements to choose potential candidates for testing. If the tested tubes retain structural and leakage integrity throughout the test, there is additional assurance that tubes with less significant degradation will have adequate margins for tube integrity.

Because the peak accident-induced loads for steam generator tubing are largely a result of thermally induced stresses rather than internal tube pressure, the in-situ pressure test device is unable to simulate the postulated axial tube loads. Such loads could challenge the structural or leakage integrity of tubes containing circumferentially oriented degradation of significant length or depth. However, because of the limitations in the in-situ testing device, the test cannot directly demonstrate the structural integrity margins for tubes containing indications of longer circumferential length. The in-situ pressure test device can effectively simulate the peak hoop stress applied to tubes containing axial indications. As discussed later, the in-situ pressure testing is proposed primarily to assess the leakage integrity of tubes with ID IGA degradation, but it also will verify the structural integrity margins of tubes with axial indications of significant length.

3.3 Demonstration of Leakage Integrity Margins

The existing depth-based repair criteria are established to ensure steam generator tubes have adequate structural and leakage integrity with appropriate margins of safety under normal operating and postulated accident conditions. The licensee will apply the depth-based repair limits to ID IGA degradation in addition to the proposed length-based limits during the 12R examinations. Because of the difficulty in accurately assessing indication depth, a majority of the tubes with these indications will be dispositioned and returned to service using only the proposed length-based repair limits. Such an approach may permit tubes containing degradation with actual depths greater than the 40 percent depth to remain in service. Under high differential pressures, this degradation could become a leak path for the reactor coolant to the steam generator secondary side. As part of the repair criteria to address the ID IGA degradation, the licensee will complete in-situ pressure testing of steam generator tubes with ID IGA indications to demonstrate a low leakage potential for tubes containing this mode of degradation.

In-situ pressure testing subjects degraded tubes to conditions that are conservative with respect to internal pressure loadings postulated to occur under accident conditions. Internal pressure within the tube during the test induces axial and circumferential stresses within the tube wall. The purpose of the testing is to assess whether the degraded tubes exposed to these elevated stresses are capable of withstanding the test conditions while retaining leakage and structural integrity. The test pressure is adjusted to account for the temperature dependence of material strength and other factors that may not be simulated in the test. If a tube leaks during testing, the leak rate can be quantified provided it is not in excess of the capabilities of the test equipment.

As mentioned in Section 3.2, the in-situ pressure testing does not effectively simulate bounding axial tube loads expected to occur under accident conditions. Therefore, the testing will not fully load the tube to the level necessary to evaluate the leakage integrity of tubes with circumferential ID IGA degradation. However, the licensee has proposed to evaluate the potential for leakage from circumferentially oriented indications through a combination of in-situ testing and analytical calculations as well as through confirmatory testing of tubes with simulated ID IGA degradation.

The limitation in the in-situ test device stems from the fact that the axial loads imparted into the tube during the test are a function of the test pressure. In order to achieve larger axial loads during a test, the in-situ test pressure would have to be increased proportionally. The direct relationship between the test pressure and axial tube load during in-situ pressure testing confound the ability to adequately simulate OTSG accident loadings. OTSGs are unique in relation to recirculating steam generators in that the tube axial loading during an accident is the result of both the mismatch in thermal expansion between the tube bundle and the steam generator shell and the internal tube pressure. Although the internal pressure loads are significant, these loads are not responsible for the majority of the axial tube loads. Because thermal and pressure loads in OTSGs are postulated to act somewhat independently during an accident, the in-situ pressure test device cannot effectively simulate the axial tube loading without a corresponding increase in the test pressure. In order to induce comparable axial tube loads during in-situ pressure testing, the internal pressure of the tube would have to be increased to levels beyond the capacity of the tube and the test device.

To address the limitations of in-situ pressure testing, the licensee has proposed to bound the potential leakage from circumferential indications using the results from in-situ pressure testing and deterministic calculations using the PICEP computer code. The licensee will test the limiting circumferential indications identified during the steam generator tube examinations. The measured leakage, actual test parameters (e.g., pressures), tube material properties, and other PICEP parameters are then used in determining the circumferential flaw length that was completely through-wall and responsible for the leakage during the test. The licensee then will calculate the expected cumulative 2-hour leakage using conservative postulated accident loadings. This bounding leakage then will be applied to the population of ID

IGA indications identified in tubes that were returned to service using the proposed repair criteria that are considered susceptible to potential accident-induced leakage.

The licensee has also completed testing of two tubes with artificial ID IGA degradation. The testing conservatively simulated postulated accident loads on the tube with the objective of quantifying the primary-to-secondary leak rate. The simulated tubes were exposed to high temperatures under applied axial loading and internal pressures. No leakage was measured during either of the tests. The degradation introduced into the tube samples prior to testing is similar to that in the TMI-1 OTSG tubing. In addition, the dimensions of the degraded areas in one of the tube samples significantly bounded the size of the indications identified in pressure boundary of the TMI-1 OTSG tubing. According to the licensee, the results from these tests considered in conjunction with the in-situ pressure test results and associated analytical calculations demonstrate that the TMI-1 OTSG tubing will maintain the required leakage integrity under postulated accident conditions.

Using the eddy current inspection results, the licensee will select several tubes for testing with indications of degradation that appear limiting from a leakage integrity standpoint. Although the inspection cannot accurately assess the depth of the degradation, other available data such as defect length, eddy current amplitude response, or phase angle depth measurements may be used to identify a subset of degraded tubes with indications that appear to bound the remaining tubes in the population. By testing the integrity of a sample of tubes, the licensee can assess the potential for leakage from all the degraded tubes. Assuming the limiting tubes retain full structural and leakage integrity throughout the test, it is reasonable to conclude that the other tubes in the population would also have sufficient integrity to withstand accident-induced loads without failure.

Steam generator tube flaws may progress in length or depth during operation and degrade the margins for tube integrity below acceptable limits. Therefore, the licensee's proposal to use in-situ pressure testing in the 12R outage is only capable of demonstrating that the population of tubes with ID IGA indications has adequate leakage integrity at the time of the test. If it can be demonstrated that the expected flaw growth rate for the ID IGA degradation is negligible, then the in-situ pressure testing will provide assurance that the affected tubes will have sufficient margins for structural and leakage integrity beyond the outage in which the testing occurred. A discussion on the expected growth rate for ID IGA indications is included in the following section.

3.4 Analysis of Growth Rate for ID IGA Degradation

The ID IGA degradation present in the TMI-1 steam generators is unique with respect to other known damage modes affecting pressurized-water reactor steam generator tubes. Since stress corrosion cracking involves the interplay between stresses in the material, the existence of a corrosive environment, and a material susceptible to this type of cracking, it has generally not been identified in the locations where degradation has been found in the TMI-1 tubes. The degradation at TMI-1 was introduced as a result of a chemistry excursion that occurred during plant shutdown years ago. Sodium thiosulfate in the reactor coolant caused significant cracking in many of the steam generator tubes. Many of these tubes were subsequently repaired and returned to service. However, some tubes currently in service contain indications of ID IGA that are the result of the past chemistry event.

The licensee has reviewed eddy current inspection data of the tubes with known ID IGA indications to assess degradation growth rates. The signal characteristics (e.g., voltage amplitude, phase angle) have remained approximately constant for the known population of degraded tubes for several cycles of operation indicating no change in the degradation. In addition, considering that the root cause of the degradation was the inadvertent introduction of sodium thiosulfate into the plant during shutdown and that these chemicals have been removed from the coolant, the driving mechanism for this mode of degradation is no longer present. Therefore, the licensee has concluded that the ID IGA indications are not growing in size or depth.

3.5 Classification of Inspection Results

The current TMI-1 TS require the classification of inspection results based on the number of degraded or defective steam generator tubes identified during inservice inspections per TS 4.19.2. The results are classified as either C-1, C-2, or C3, and depending on the classification, the licensee may be required to complete additional inspections during an outage. TS 4.19.2 exempts previously degraded tubes from the inspection results classification except those tubes with indications that have exhibited significant growth from the previous examinations. The term significant is defined within the context of this TS requirement as any indication that increases in depth by greater than 10 percent through-wall. Since the licensee's current proposed TS change is based on criteria that are other than depth-based, a modification is proposed to TS 4.19.2 to define significant as an increase in voltage for ID IGA indications of greater than 0.6 volts. This value was determined based on a statistical review of the outage-to-outage variation in the voltage for ID IGA indications. Historically, a significant fraction of the ID IGA indications have exhibited voltage changes less than this value. Therefore, any ID IGA indications that have an increase in voltage greater than this proposed threshold are likely to have grown from the previous inspection.

4.0 STAFF EVALUATION

The proposal to measure the length of ID IGA degradation in both the axial and circumferential directions and remove from service those tubes exceeding the proposed dimensional repair limits will ensure that all tubes with confirmed indications have adequate structural integrity at the beginning of Cycle 12. In accordance with the guidance provided in NRC Regulatory Guide 1.121, steam generator tube repair limits generally incorporate an allowance for degradation growth over the next cycle of operation. The proposed dimensional limits, however, do not include a growth allowance because the licensee has concluded that the ID IGA degradation is currently inactive. Growth assessments that considered changes in the voltage of ID IGA indications over recent fuel cycles have concluded that there has been no increase in the mean voltage.

Although the growth rate study that examined the ID IGA degradation in the TMI-1 steam generators concluded that there was no apparent growth, the NRC staff identified a number of variables not specifically addressed in these studies. These variables could affect the bobbin voltage response and inhibit the ability to detect a voltage increase for some of the indications caused by actual growth. Because the licensee's study primarily relied upon an evaluation of the changes in indication voltage between inspections, such effects would make an accurate assessment of limited growth difficult. The sources of variability identified by the NRC staff include: (1) bobbin probe wear, (2) calibration practices and standards, (3) differences in data acquisition hardware, and (4) data analyst uncertainty. Despite the potential errors in the growth rate study from these factors, the licensee concluded that the IGA degradation mechanism is currently not active.

Since the bobbin coil voltage amplitude is sensitive to the size of volumetric indications, the bobbin probe data should be an effective screening tool for the detection of growth for the IGA indications at TMI-1. However, the outage-to-outage variation in voltage for these indications may vary between successive inspections. The NRC staff believes the variations observed in the licensee's growth rate study are primarily caused by one or more of the sources of error discussed previously Such variations make it difficult to accurately assess growth. However, the analysis did not reveal a well defined increase in the mean voltage for the population of indications studied. This indicates that the growth of the IGA defects is below the level of scatter in the voltage measurements due to nondestructive evaluation (NDE) uncertainty. In the absence of a definitive assessment of NDE uncertainty, the NRC staff concludes that the population of IGA indications may be experiencing limited growth during operation. However, since the potential growth for the next cycle would be expected to proceed at a rate similar to prior operational cycles, the staff concludes that tubes left in service as a result of using the proposed repair criteria should maintain similar structural and leakage integrity margins through the end of the next operating cycle. In addition, the 0.6 volt threshold defined for the inspection results classification in TS 4.19.2 will provide assurance that any indications which are actually growing are identified during the course of the OTSG tube examinations.

The length of a volumetric indication can be measured from a rotating probe examination using currently available eddy current data analysis software. Because of the limitations in the detection capabilities for eddy current inspection methods, there is the potential that some degradation could have axial or circumferential lengths greater than the proposed repair limits. However, the repair limits were determined assuming the tube degradation extended 100 percent through-wall. Plus Point inspection probes have demonstrated the capability to detect IGA tube degradation with partial through-wall depths much less than 100 percent. Therefore, the length of degradation that is not detected during the inspection, and thus not included in the length measurements, would not significantly diminish the structural margin of the tube.

The staff notes that, in general, measurements of IGA flaw length are typically conservative because of a combination of eddy current field effect and the degradation morphology. An eddy current coil interrogates a volume of tube material that is larger than the physical dimensions of the coil. Because a coil's electromagnetic field extends beyond the coil dimensions, inspection probes have the ability to detect degradation before and after the coil passes over the actual tube degradation. Therefore, this has the effect of extending the bounds of tube degradation as measured from the data. The ability to detect degradation before passing a coil over the affected tube areas also depends, in part, on the geometry of the degradation. The three-dimensional morphology of IGA tube degradation facilitates its detection by eddy current methods. The conservatism in measuring IGA degradation length has been demonstrated for less detectable forms of IGA degradation experienced at other pressurized-water reactor (PWR) facilities.

The staff recognizes that there is a degree of uncertainty associated with calling an indication ID or OD. However, the use of data from two inspection probes will minimize the potential for applying the proposed repair criteria to OD-initiated degradation. This should provide adequate assurance that the length-based repair limits are applied only to ID-initiated degradation. The staff also notes that the licensee has not reported the presence of localized OD IGA in the locations of the TMI-1 steam generators with known ID IGA indications. Therefore, it is probable that all bobbin coil indications characterized as originating from the ID are associated with the tube damage that occurred in 1981.

The staff has previously reviewed similar repair criteria for OTSG tubing and concluded that the axial length repair limit of 0.25 inches was acceptable. The staff's evaluation is documented in Amendment 154 to the Crystal River Unit 3 Technical Specifications dated April 30, 1996. The licensee's proposed repair limit for circumferentially oriented degradation differs from that evaluated in the staff assessment of the Crystal River Unit 3 TS amendment; however, the staff has reviewed the basis for the proposed repair limit as documented in previous submittals related to the licensee's effort to restore defective tubes to service after the ID IGA degradation was identified in the 1981 timeframe. The staff's evaluation approving the original repairs is included in NUREG-1019. The staff reviewed the information submitted in support of NUREG-1019 and completed independent calculations that confirm that the proposed circumferential length repair limit will ensure that tubes with ID IGA indications will have adequate structural integrity under accident-induced loads. On this basis, the staff concludes that the proposed circumferential repair limits are acceptable.

The use of in-situ pressure testing will effectively demonstrate the leakage integrity of tubes with ID IGA degradation. Because the growth of ID IGA degradation in the TMI-1 steam generator tubes is expected to be minimal over the next cycle of operation, the in-situ pressure test results will provide assurance that the leakage integrity margins for tubes with axial ID IGA indications are maintained through the end of the next cycle of operation. Although the in-situ pressure testing device cannot effectively simulate the axial tube loads to directly assess the leakage integrity of circumferentially oriented indications, the licensee has proposed to bound the expected leakage from tubes with these indications using a combination of in-situ pressure test results and analytical calculations. The testing completed on tubes with simulated ID IGA degradation also provides confirmation that the tubes returned to service using the proposed repair criteria should have acceptable margins for leakage integrity.

The staff also notes that tubes with volumetric IGA degradation in OTSGs at other PWR facilities typically have significant margins for structural and leakage integrity. Burst tests of tubes removed from service with IGA indications have shown significant margins for leakage integrity under postulated accident conditions. In addition, the licensee for TMI-1 has not attributed any measurable operational leakage in the OTSGs to the presence of ID IGA degradation in the tubing. The licensee reported that the tubes with ID IGA indications that have been in-situ pressure tested did not leak during the tests. A comparison of eddy current inspection data collected before and after the testing indicated that the degradation does not appear to have grown in length or depth as a result of the test. This indicates that the tubes that were in-situ pressure tested have leakage integrity margins beyond the loading conditions encountered in the tests.

The licensee has indicated that it will select tubes for in-situ pressure testing with indications that appear limiting from a leakage integrity standpoint. Although the eddy current examinations cannot accurately determine the depth of ID IGA indications, the staff considers that the tube selection factors proposed for selecting the limiting tubes are a strong indication of the tubes with the most limiting indications from a leakage integrity perspective. The in-situ pressure test results will demonstrate the ability of the tubes with axial indications to resist leakage under postulated accident conditions. The licensee will also demonstrate that the maximum postulated leakage from circumferentially oriented indications is bounded by the allowable offsite dose leakage. Therefore, the staff concludes that the licensee's proposal to assess the potential leakage from tubes with ID IGA indications through in-situ pressure testing and analytical calculations is acceptable for this one cycle amendment.

The licensee has proposed to modify TS 4.19.5, "Report," to require that the NRC be notified of its assessment of ID IGA degradation and the results of in-situ pressure testing. In addition, the licensee will be required to submit additional information regarding the ID IGA indications identified in the inservice inspections. These changes will enable the NRC staff to assess the significance of the inservice inspection findings in a timely manner. Therefore, should the staff identify any concerns stemming from the information provided by the licensee, it could consider appropriate actions. The staff concludes that the proposal to change TS 4.19.5 is an improvement to the existing reporting requirements on the basis that the licensee will be required to provide the NRC with information that will enable the staff to identify potential issues related to steam generator tube integrity.

5.0 OTHER CONSIDERATIONS FOR THE PROPOSED REPAIR CRITERIA

During a meeting between the licensee and the NRC on November 26, 1996, to discuss TMI-1 steam generator degradation issues, the licensee indicated that the NRC had previously reviewed and approved of using a bobbin coil probe to size the depth of ID IGA degradation as documented in NUREG-1019. Because the previous staff review was completed approximately 15 years prior to the licensee's current proposed alternate tube repair criteria, the staff has reviewed the original safety evaluation from the NRC as well as supporting documentation. The review focused on the qualification of the eddy current examination methods used to inspect the ID IGA degradation. The NRC's conclusion in NUREG-1019 states "that the eddy-current techniques developed and qualified for inspection of the OTSG tubing demonstrated the ability to reliably detect and size, with a high degree of sensitivity, the defects that were present in the tubing." Based on a reexamination of the original qualification of the eddy current methods, the staff has concluded that the statement in NUREG-1019 did not apply to the ability to adequately size the depth of ID IGA degradation. Rather, the reference to sizing in the safety evaluation applies to the ability to size the circumferential extent of indications using other inspection probes. This conclusion is based on the scope of the licensee's original qualification, as well as the conclusions and recommendations that were included in the docketed information.

In order to address the inability to accurately size the depth of ID IGA degradation using a qualified inspection technique, the licensee has proposed to apply the proposed length-based steam generator tube repair criteria. These criteria will ensure that a mode of degradation that cannot be effectively dispositioned by using the existing repair limit in the TS has equivalent margins for tube structural and leakage integrity for operation until the Cycle 13 Refueling Outage. Should the licensee intend to apply the depth-based repair limits in the next inservice inspection of the TMI-1 steam generators, additional qualification may be necessary to demonstrate that the inspection technique is capable of assessing the depth of ID IGA degradation with a level of accuracy commensurate with the bases for the repair limit.

The proposed OTSG tube repair criteria are in effect for only one cycle of operation. Because of the short duration and limited applicability of the alternate repair criteria and staff findings documented in Draft NUREG-1570, the licensee was not requested by the NRC to complete analyses to identify potential beyond-design-basis challenges to the OTSG tubes. If such accident challenges were identified and if these challenges were determined to be severe enough and expected to occur at too high a frequency, the licensee would have to address the potential failure of the tubes due to high temperature mechanisms such as creep or erosion and ablation. The staff is currently developing a revised regulatory framework for addressing steam generator tube degradation issues that may require licensees to address severe accident issues for alternate repair criteria. Preliminary, generic assessments by the staff documented in NUREG-1570 of the potential frequency of severe accident challenges indicate that OTSGs are less susceptible to high temperature and pressure scenarios than recirculating steam generators. However, should the licensee request to make permanent the proposed alternate repair criteria for tubes with ID IGA indications for future inspections, a plant-specific assessment of postulated severe accident issues may be necessary.

6.0 PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

In order to implement the proposed steam generator tube alternate repair criteria for Cycle 12 operation, the licensee has proposed the following changes to the TMI-1 TS.

1. Proposed modification to Note (1) of TS 4.19.2, "Specification"

   Note (1) is amended to specify that a greater than 0.6 volt increase in the bobbin coil amplitude for ID IGA degradation is considered significant. Tubes with voltage increases above this value are included in the determination of the inspection results category.

2. Proposed modification to TS 4.19.4.a.1, "Imperfection"

   The definition of imperfection is modified to state that indications with responses less than the criteria established in TS 4.19.4.a.3, "Degraded Tube," may be considered imperfections.

3. Proposed modification to TS 4.19.4.a.3, "Degraded Tube"

   A tube with an ID IGA indication exhibiting a bobbin coil amplitude response greater than or equal to 0.5 volts or having an axial length of at least 0.13 inches or a circumferential length greater than or equal to 0.26 inches is considered degraded. These criteria are only in effect for the 12R outage examinations and Cycle 12 operation.

4. Proposed modification of TS 4.19.4.a.6, "Repair Limit"

   The Repair Limit for steam generator tubes is amended to define the limits for tubes with ID IGA indications. For the 12R Outage examinations and Cycle 12 operation, tubes with ID IGA indications that exceed measurements in the axial or circumferential direction of 0.25 and 0.52 inches, respectively, shall be repaired or removed from service. Additionally, tubes containing indications with a measured depth greater than or equal to 40-percent through-wall shall also be repaired or removed from service.

5. Proposed new TS 4.19.4.a.9, "Inside Diameter Inter-Granular Attack (IGA) Indication"

   A definition of an ID IGA Indication is specified as a bobbin coil indication on the ID surface and confirmed by diagnostic eddy current inspection techniques to have a volumetric morphology characteristic of IGA.

6. Proposed modification of TS 4.19.5, "Reports"

   The reporting requirements are modified to specify that the licensee must notify the NRC prior to exceeding a reactor coolant system temperature of 250°F of: (1) the assessment of the growth of ID IGA degradation and (2) the results of in-situ pressure testing, if performed. In addition, within 90 days from the completion of inspection and repairs, the licensee must notify the NRC of the location, bobbin coil amplitude, and axial and circumferential extent (if determined) for each ID IGA indication.

7. Proposed modification of the Bases Section

   The Bases Section for TS 4.19, "OTSG Tube Inservice Inspection" is modified to discuss the inclusion of the proposed repair criteria for ID IGA indications for the 12R Outage.

7.0 SUMMARY

The licensee has proposed to impose length-based repair limits for ID-initiated IGA indications detected in the TMI-1 steam generator tubes. These limits will ensure that the structural integrity of the tubes is adequate for loads under normal operating and postulated accident conditions. The licensee has completed testing of artificial tube flaws that demonstrates the resistance of ID IGA degradation to leakage, and it has proposed to complete in-situ testing to assess the leakage integrity of tubes with limiting ID IGA indications. Successful in-situ pressure testing (i.e., without failure) completed on a limited sample of tubes will also provide additional validation of tube structural integrity margins. The staff has reviewed the proposed length-based limits for defining degraded tubes and specifying a repair limit as proposed in TS 4.19.4.a.3 and TS 4.19.4.a.6, respectively. On the basis provided in Section 4.0 of this Safety Evaluation, the staff concludes that these changes are acceptable for Cycle 12 operation. Based on the staff's evaluation documented in Section 4.0 of this Safety Evaluation, the proposed changes to the TS that are not limited to Cycle 12 operation as noted in Section 4.0 of this Safety Evaluation (i.e., TS 4.19.2, 4.19.4.a.1, 4.19.4.a.9, and 4.19.5) clarify steam generator tube degradation and reporting requirements consistent with proposed alternate repair criteria, and are thus acceptable.

8.0 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.

9.0 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 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 (62 FR 45458). The amendment also modifies reporting or recordkeeping requirements. 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.

10.0 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.

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