

May 29, 1990

Docket No. 50-289

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

Dear Mr. Hukill:

SUBJECT: THREE MILE ISLAND NUCLEAR STATION UNIT 1 - ISSUANCE OF AMENDMENT
(TAC NO. 76160)

The Commission has issued the enclosed Amendment No. 153 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, in response to your letter dated March 12, 1990.

The amendment changes the Technical Specifications related to inservice inspections of steam generator tubes following a primary-to-secondary leak.

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,

/s/

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 153 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

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NAME	:SNOYTS	:RHernan:RWA	:JStolz	:L. Dewey		:	:
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GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated March 12, 1990 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.

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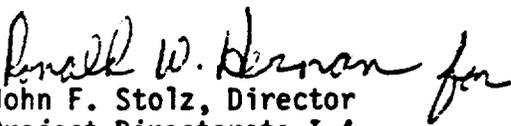
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 153, 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


John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove

4-78
4-79
4-80
4-81
4-82

Insert

4-78
4-79
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4-82

each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
 1. All nonplugged tubes that previously had detectable wall penetrations (>20%).
 2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.
 3. A tube inspection (pursuant to Specification 4.19.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
 4. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.
 - (1) Group A-1: Tubes in rows 73 through 79 adjacent to the open inspection lane, and tubes between and on lines drawn from tube 66-1 to tube 75-15 and from 86-1 to 77-15.
 - (2) Group A-2: Tubes having a drilled opening in the 15th support plate.
- b. The tubes selected as the second and third samples (if required by Table 4.19.2) during each inservice inspection may be subjected to a partial tube inspection provided:
 1. The tubes selected for these second and third samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.
 2. The inspection includes those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category

Inspection Results

C-1

Less than 5% of the total tubes inspected in a steam generator are degraded tubes and none of the inspected tubes are defective.

- 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 must exhibit significant (>10%) further wall penetrations 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.
- 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:

*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

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 below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.
2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.
3. Degraded Tube means a tube containing imperfections $\geq 20\%$ of the nominal wall thickness caused by degradation.
4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.
5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

4.19.4 Acceptance Criteria (Continued)

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.

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.

- 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. Following the completion of each inservice inspection of steam generator tubes, the number of tubes repaired or removed from service in each steam generator shall be reported to the NRC within 15 days.
- b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection.

This 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. 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 followup 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 RGS will be maintained.

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

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

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

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

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

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.

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



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 153 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

INTRODUCTION:

By letter of March 12, 1990, GPU Nuclear Corporation (GPUN) requested a change in the Three Mile Island Nuclear Station, Unit 1 Technical Specifications relating to steam generator surveillance.

The change was requested to modify the TMI-1 Technical Specifications for unscheduled steam generator tube inspection requirements after a primary-to-secondary leak in excess of the limits of Specification. The proposed change specifies that: (1) when a leaking tube is located in Group A-1 ("lane wedge" area) all tubes in this group in only the affected steam generator need be inspected (current Technical Specifications are not explicit in this regard) to include those portions of the tubes where the leak was found, and if the results of the inspection fall into the C-3 Category, additional inspections will be performed in the same group in the other steam generator; and (2) when the leaking tube is not in Group A-1, an inspection will be performed on the affected steam generator in accordance with the Technical Specification.

BACKGROUND

On March 6, 1990 at 0912, TMI-1 began a plant shutdown because of a primary to secondary leak in the once through steam generator (OTSG) which occurred shortly after a refueling outage. Following cooldown, the A OTSG was opened and a bubble test performed on March 8. The test identified tube 1 in row 77 (designated A77-1) as the leaking tube. This tube is in the "lane wedge" region of the OTSG and had been Eddy Current examined in January 1990 as part of the 8R refueling inservice inspection program. The 8R inspection identified no recordable indication of degradation on tube A77-1. Post-leak Eddy Current inspection performed on March 9 identified that A77-1 had a through wall defect at the point where the tube exits the bottom of the upper tube sheet.

The failure of the tube was identified as a circumferentially oriented 360° crack. This is believed to be the result of environmentally assisted high cycle fatigue (HCF). This is based on eddy current testing (ECT) data, visual examination of the tube, and on a comparison of the failure with prior industry experience with HCF cracking of OTSG tubes. The determination that "lane wedge" area tubes are susceptible to this failure mechanism is based on four tube samples from the Oconee Nuclear Station removed and analyzed during the period from 1976 to 1982.

The ECT performed for the inservice inspection program on tube A77-1 during January 1990 yielded no recordable indications of degradation. B&W industry experience confirms that this type of failure occurs rapidly and therefore evidence of the condition may exist only shortly before leakage would be experienced. Tube inspection techniques do not effectively identify HCF precursor conditions unless they are performed just prior (e.g. hours) before tube failure. Mitigating actions in response to tube leakage are provided by plant normal and emergency procedures.

GPUN completed an inspection of all tubes in the "lane wedge" area of the A OTSG. No new imperfections of these tubes were identified which differed from the prior outage inspection results. It should be noted that one tube defect was found during examination of the "lane wedge" area following the tube leak. The defect was a shallow inside diameter pit at 41% through wall based on a less than 1 volt one coil indication on a 8x1 absolute ECT probe. A review of the ECT data confirmed that this defect existed during the last inspection. According to the licensee, it was not previously judged as defective because of its very low signal level and shallow phase angle.

GPUN considered the tube failure to be caused by HCF, an industry identified problem. Since additional eddy current inspection in the "lane wedge" area of the A OTSG had essentially duplicated the results of inspections performed during the previous outage inservice inspection, it was unnecessary to expand the present ECT beyond the "lane wedge."

A Waiver of Compliance was issued by the NRC on March 14, 1990, that permitted resumption of plant operation without completing Technical Specification required random ECT tube examinations of the affected OTSG. Rather, a focused ECT program was performed which inspected all unplugged "lane wedge" area tubes, which OTSG industry experience has demonstrated are prone to the HCF failure mechanism. This ECT was performed down to the 14th support plate which includes all of the HCF failure-prone tube portions. This inspection resulted in the two indications noted above. A drip test of the entire OTSG was performed to provide additional confidence in the integrity of the tubes, and showed no problems. A post-repair bubble test was also performed.

EVALUATION

TMI-1 Technical Specification Section 4.19.3.c.1 currently specifies that additional unscheduled inservice inspections shall be performed on each steam generator during shutdown following a primary-to-secondary tube leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of the Technical Specification.

The proposed change to limit the unscheduled inservice inspection to the leaking steam generator following primary-to-secondary leakage through the steam generator tubes which exceeded Technical Specification limits will reduce personnel radiation exposure associated with the inspections without compromising the objective of these inspections. If the leaking tube is located in the "lane wedge" area and the results of the unscheduled inspection of the affected steam generator fall into the C-3 category, additional inspections will be performed in the same tube group in the other steam generator. If the leaking tube is not located in the "lane wedge" area the unscheduled inspection will be performed on the affected steam generator only, in accordance with existing Technical Specification.

OTSG industry experience has shown that the "lane wedge" area has been experiencing corrosion assisted fatigue and fretting wear. This area is more susceptible to damage due to the proximity to the open lane which allows higher moisture carryover and highest cross flow since the steam changes direction from vertical to horizontal to exit the steam generators. Performing Technical Specification limited tube inspection in the area where leaks are found will identify potential additional tubes which may be experiencing similar degradation and enabling appropriate corrective action to be taken to prevent further tube leakage.

The licensee has stated that this approach is consistent with EPRI recommendations in the PWR Inspection Guidelines, which means that the inspections would normally be performed with a 8x1 coil.

SUMMARY

Industry experience indicates that the failure of tubes in the area of the failed tube are due to HCF. This mechanism is a rapid failure mechanism and precursors may exist only briefly before failure occurs. There is no method to predict failures with such a rapid development. Leakage monitoring may be one effective means of detection for slowly developing leaks. Repeating random ECT of the A OTSG outside the "lane wedge" area would provide no additional technical information relevant to this failure mechanism. We therefore conclude, based on the above, that the proposed Technical Specification change should be granted.

STAFF RECOMMENDATIONS

In view of the fact that other B&W plants that have experienced fatigue failures at the upper tube sheet have implemented preventive tube sleeving programs, we recommend that GPUN consider such a program unless it can be shown that the probability of additional fatigue cracks leading to a steam generator tube rupture is very low. However, we believe that probably the only way this can be shown is to demonstrate that augmented leak detection and leak rate monitoring methods can detect incipient leaks due to fatigue crack initiation before the rapidly propagating crack has grown around the circumference of the tube leading to a potential tube rupture.

ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and changes surveillance requirements. We have determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released off site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

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

Principal Contributor: Herb Conrad

Dated: May 29, 1990

DATED: May 29, 1990

TMI-1 AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. DPR-50

DISTRIBUTION

Docket File

NRC & Local PDR

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