September 11, 1996

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Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. M94810)

Dear Mr. Marchi:

The Commission has issued the enclosed Amendment No. 126 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated February 19, 1996, as supplemented on July 3 and August 26, 1996.

The amendment revises KNPP TS Section 4.2 and its associated basis by allowing the application of a voltage-based repair limit for the steam generator tube support plate intersections experiencing outside diameter stress corrosion cracking. The repair criteria are based on guidance provided in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995, and on associated industry guidance.

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

Sincerely,

Original signed by:

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

100021

Docket No. 50-305

Enclosures: 1. Amendment No. 126 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

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DATE	9/10/96		9/10/96		9/9/96	

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

September 11, 1996

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

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2. Safety Evaluation

cc w/encls: See next page

Mr. M. L. Marchi
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

cc:

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Chairman
Town of Carlton
Route 1
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Mr. Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

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

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated February 19, 1996, as supplemented on July 3 and August 26, 1996, 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-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 126 , are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Laufer

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: September 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 126

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS ii	TS ii
TS 4.2-3	TS 4.2-3
TS 4.2-5	TS 4.2-5
TS 4.2-6	TS 4.2-6
TS 4.2-7	TS 4.2-7
TS 4.2-8	TS 4.2-8
TS B4.2-4	TS B4.2-4
---------	TS B4.2-5

Section	Title	Page

Amendment No. ~~64, 91, 95, 100, 103, 106, 110, 118, 119,~~ 126

1. Steam Generator Sample Selection and Inspection

The in-service inspection may be limited to one steam generator on a rotating schedule encompassing the number of tubes determined in TS 4.2.b.2.a provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. Steam Generator Tube Sample Selection and Inspection

The tubes selected for each in-service inspection shall:

a. Include at least 3% of the total number of nonrepaired tubes, in both steam generators, and 3% of the total number of repaired tubes in both steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted below and in TS 4.2.b.2.b.

Indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future REFUELING outages.

b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

Implementation of the steam generator tube/tube support plate repair criteria requires a 100% bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

d. The second and third sample inspections during each in-service inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tubesheet array and on those portions of the tubes where tubes with imperfections were previously found.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of TS 3.1.d and TS 3.4.a.1.C or

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

   3. A loss-of-coolant accident requiring actuation of the engineering safeguards, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr, or

   4. A main steam line or feedwater line break, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next outage of sufficient duration following 3 months of power operation since the change.

4. <u>Plugging Limit Criteria</u>

The following criteria apply independently to tube and sleeve wall degradation except as specified in TS 4.2.b.5 for the tube support plate intersections for which voltage-based plugging criteria are applied.

a. Any tube which, upon inspection, exhibits tube wall degradation of 50% or more shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Tube repair shall be in accordance with the methods described in WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)" or CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves."

b. Any Westinghouse mechanical sleeve which, upon inspection, exhibits wall degradation of 31% or more shall be plugged prior to returning the steam generator to service. Figure TS 4.2-1 illustrates the application of tube, sleeve, and tube/sleeve joint plugging limit criteria.

c. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation of 40% or more shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld region.

5. Tube Support Plate Plugging Limit

The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersection, the repair limit is based on maintaining steam generator tube serviceability as described below:

a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage $\leq$ 2.0 volts will be allowed to remain in service.

b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in TS 4.2.b.5.c below.

c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but $\leq$ the upper voltage repair limit, may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage > the upper voltage repair limit will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following repair limits apply instead of TS 4.2.b.5.a, b and c. The mid-cycle repair limits are determined from the following equation:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Amendment No. ~~52, 73, 76, 93, 95, 103, 118,~~ 126

Where:

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle
V_{SL} = structural limit voltage
NDE = 95% cumulative probability allowance for NDE uncertainty
Gr = average growth rate per cycle length
CL = cycle length (time between scheduled inspections)
Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
V_{URL} = upper voltage repair limit

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.2.b.5.a, b and c.

NOTE: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

6. Reports

a. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degradation.

3. Identification of tubes plugged.

4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10 CFR 50.72(b)(2)(i). A written follow up report shall be submitted to the Commission consistent with Specification 4.2.b.6.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

There are three types of Combustion Engineering leak tight sleeves. The first type, the straight tubesheet sleeve, spans the degraded area of the parent tube in the tubesheet crevice region. The sleeve is welded to the parent tube near each end. The second type of sleeve is the peripheral tubesheet sleeve. The sleeve is initially curved as part of the manufacturing process and straightened as part of the installation process. The third type of sleeve, the tube support plate sleeve, spans the degraded area of the tube support plate and is installed up to the sixth support plate. This sleeve is welded to the parent tube near each end of the sleeve.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

Technical Specification 4.2.b.5

The repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 limits are not exceeded.

The voltage-based repair limits of TS 4.2.b.5 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generators. Additionally, the repair criteria apply only to indications where the degradation mechanism is predominantly axial ODSCC with no indications extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of TS 4.2.b.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{GR} - V_{NDE}$$

Where V_{GR} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

The mid-cycle equation should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.

Technical Specification 4.2.b.6

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(ii).

TS 4.2.b.6.d implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the steam generators to service. For TS 4.2.b.6.d.3 and 4, indications are applicable only where alternate plugging criteria is being applied. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the steam generators to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing GL Sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per GL Section 6.b(c) criteria.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated February 19, 1996, as supplemented on July 3 and August 26, 1996, Wisconsin Public Service Corporation (WPSC), the licensee, requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would revise KNPP TS Section 4.2 and its associated basis by allowing the application of a voltage-based repair limit for the steam generator tubes. The proposed alternative repair criteria would allow tubes having outside diameter stress corrosion cracking (ODSCC) oriented in the axial direction and confined within the tube support plates to remain in service if the tube inspection and associated results satisfy certain staff recommended criteria and limitations. The staff criteria are outlined in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995.

The July 3 and August 26, 1996, submittals provided clarifying information that did not alter the proposed no significant hazards determination published in the April 10, 1996, Federal Register.

2.0 BACKGROUND

GL 95-05 outlined criteria for licensees considering implementation of a voltage-based alternate repair criteria. The traditional regulatory framework for monitoring steam generator tube integrity establishes requirements for a minimum wall thickness in accordance with Regulatory Guide 1.121. The plant TS require that any tubes having degradation which exceeds 40 percent through the tube wall will be repaired. The 40 percent limit is conservative for highly localized flaws such as pits, short cracks and in particular ODSCC that occurs at the tube support plates.

The voltage-based alternate repair criteria do not set limits on the depth of ODSCC to ensure tube integrity; instead it relies on correlating the eddy current voltage amplitude from a bobbin coil probe with the specific measurement of tube burst pressure and leak rate. This approach takes no credit for the tube support plates in preventing and/or reducing the likelihood of a tube from bursting and/or leaking during postulated accident conditions. It assumes that the degradation affecting the steam generator tubes at the tube support plate elevation is in the free span region of the tubes.

GL 95-05 specifies, in part, that (1) the repair criteria is only applicable to axially oriented ODSCC located within the bounds of the tube support plates; (2) licensees should perform an evaluation to confirm that the steam generator tubes will retain adequate structural and leakage integrity until the next scheduled inspection; (3) licensees should adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes should be periodically removed from the steam generators to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit should be reduced; (6) specific reporting requirements should be followed; and (7) the licensees' proposed TS should follow the model TS in Attachment 2 of the GL.

3.0 EVALUATION

Kewaunee uses two Westinghouse model 51 steam generators. The tubes were fabricated using mill annealed alloy 600 material. There are 3388 tubes per each steam generator.

The staff approved an application for the interim alternate repair criteria for Kewaunee in license amendment number 118 dated April 17, 1995. The interim criteria will expire at the end of the current operating cycle (Cycle 21) in October 1996. The proposed permanent alternate repair criteria will replace the interim criteria and will eliminate the need for periodic license amendments for the repair criteria addressed by GL 95-05.

The licensee has committed to follow the guidance in GL 95-05 for its proposed alternate repair criteria. In addition, the licensee has proposed TS that are consistent with the model TS in GL 95-05. Clarifications regarding the use of GL 95-05 are discussed below.

Section 1.b.1 of Attachment 1 to GL 95-05 specifies that the repair criteria do not apply to tube-to-tube support plate intersections where the tube with degradation may potentially collapse or deform as a result of the combined postulated loss-of-coolant accident and safe shutdown earthquake loadings. Licensees should perform or reference an analysis that identifies which intersections are to be excluded. The licensee submitted an analysis, WCAP-12985, Revision 2, as a part of its application for the interim alternate repair criteria on May 28, 1993. As a result of the licensee's analysis, no tubes will be excluded from using the voltage-based repair criteria.

Section 2.0 of Attachment 1 to GL 95-05 specifies criteria for an acceptable evaluation to confirm that the tubes will maintain adequate structural and leakage integrity until the next scheduled inspection. The evaluation includes a conditional burst probability calculation and a total leak rate calculation from the affected steam generators during a postulated main steam line break. The licensee stated that it will follow the methodology described in the Westinghouse report, "SLB Leak Rate and Tube Burst Probability Analysis

Methods for ODSCC at TSP Intersections," WCAP-14277. The report prescribes deterministic and probabilistic methods. The staff has approved the probabilistic method in WCAP-14277 in calculating the leak rate and tube burst probability because it is consistent with the acceptance criteria outlined in GL 95-05. Westinghouse stated that the deterministic method in WCAP-14277 is for screening purposes which the staff has not approved. The probabilistic methodology should be used for all regulatory commitments. To assure the adequacy of these methods, the staff will periodically verify the results of these calculations and assess the effectiveness of the methodologies as indicated in GL 95-05.

Section 2.b.3(2) of Attachment 1 to GL 95-05 recommends that licensees use all applicable data (e.g., burst pressure, probability of leakage, and conditional leak rate databases) consistent with the latest revision of the industry database as approved by the NRC in their tube integrity evaluations (e.g., calculation of tube repair limits, conditional burst probability, and total leakage under postulated accident conditions). The licensee stated that the latest NRC approved database, using the NRC approved data exclusion criteria, will be applied to the voltage correlations used in calculating the leak rate, burst probability and upper voltage repair limit. The licensee committed to use the NRC approved industry database, "EPRI ARC Database for 3/4-inch and 7/8-inch Diameter Tubes and Updated ARC Correlation for 7/8-inch Diameter Tubes," which was submitted By the Tennessee Valley Authority for Sequoyah Unit 2 on March 4, 1996.

Nuclear Energy Institute (NEI) is developing a protocol for updating the steam generator degradation database. The staff will review the adequacy of the updating process and the associated database. Pending the implementation of an NRC approved process for updating a generic industry database for steam generator tube degradation, the licensee has committed to provide the NRC with the database it intends to use prior to each refueling outage.

Sections 3.c.2 and 3.c.3 of Attachment 1 to GL 95-05 specify guidance in regard to probe wear and variability. In addition to guidance in both sections, the licensee committed to the guidance in the two letters from NEI to NRC dated January 23 and February 23, 1996, and the two letters from NRC to NEI dated February 9 and March 18, 1996. The licensee will verify that both the primary and mix frequencies of the probe will satisfy the ±10 percent variability requirement.

Section 4 of Attachment 1 to GL 95-05 specifies guidance in regard to tube removal and examination and testing. The licensee removed three tubes with a total of five intersections from the steam generators in 1993 for initial application of the voltage-based repair limit. A minimum of two tube support plate intersections will be removed from the steam generators during the 1996 refueling outage. The licensee stated that subsequent tube removal will either be in accordance with the frequency specified in GL 95-05, or an alternative industry program endorsed by the NRC.

On the basis of the information submitted, the staff concludes that the licensee's proposed TS revision to permanently incorporate the alternate repair criteria for the steam generator tubes satisfies the intent of GL 95-05 and will provide adequate assurance of steam generator tube integrity. Therefore, the proposed amendment is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 15999). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

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

Principal Contributor: J. Tsao

Date: September 11, 1996