March 13, 1997

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENTS RE: ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBING (TAC NO. M95894)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 215 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 19, 1996, and supplemented September 19, 1996, and December 20, 1996.

The amendments revise the TS to allow a permanent extension of the interim steam generator tube voltage-based repair criteria for steam generator tubes used in Cycles 13, 14 and 15 at the Donald C. Cook Nuclear Power Plant, Unit 1. The inspection and analysis methodology of the voltage-based plugging criteria approved for Cycle 15 was modified to incorporate the staff recommendations stated in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," issued August 3, 1995.

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

Sincerely,

Original signed by:

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315
Enclosures: 1. Amendment No. 215 to DPR-58
2. Safety Evaluation
cc w/encl: See next page

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

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Docket Nos. 50-315

Enclosures: 1. Amendment No. 215 to DPR-58
2. Safety Evaluation

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DATED: March 13, 1997

AMENDMENT NO. 215 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK-UNIT 1

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 215
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated June 19, 1996, and supplemented September 19, 1996, and December 20, 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-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 215, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John B. Hickman, 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:

ATTACHMENT TO LICENSE AMENDMENT NO. 215

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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

REMOVE	INSERT
3/4 4-8	3/4 4-8
3/4 4-11	3/4 4-11
-	3/4 4-11a
-	3/4 4-11b
3/4 4-12	3/4 4-12
3/4 4-16	3/4 4-16
B 3/4 4-2a	B 3/4 4-2a
-	B 3/4 4-2b
-	B 3/4 4-2c
B 3/4 4-3	B 3/4 4-3

SURVEILLANCE REQUIREMENTS (continued)

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.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 left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. In addition to the sample required in 4.4.5.2.b.1 through 3, all tubes which have had the F* criteria applied will be inspected in the roll expanded region. The roll expanded region of these tubes may be excluded from the requirements of 4.4.5.2.b.1.

d. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for the samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

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

f. Inspection of sleeves will follow the initial sample selection (1st sample) and sample expansion requirements of Table 4.4-2.

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 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 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 are degraded tubes or more than 1% of the inspected tubes are defective.

SURVEILLANCE REQUIREMENTS (continued)

9. Sleeving a tube is permitted with tube support plate sleeves and with tubesheet sleeves. Tube support plate sleeves are centered about the tube support plate intersection. Tubesheet sleeves start at the primary fluid tubesheet face and extend to the free span region of tube above the tubesheet.

10. Tube Support Plate Repair Limit is used for the disposition of an alloy 600 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 intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

    a. Steam generator tubes, whose degradation is attributed to ouside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

    b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 4.4.5.4.a.10.c below.

    c. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit[1] may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit[1] will be plugged or repaired.

    d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.10.a, 4.4.5.4.10.b, and 4.4.5.4.10.c.

1 The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05.

SURVEILLANCE REQUIREMENTS (continued)

The mid-cycle repair limits are determined from the following equations:

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

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

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit bassed on V_{MURL} and time into cycle

Δt = Length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (the time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)[2]

Implementation of these mid-cycle repair limits should follow same approach as in TS 4.4.5.4.10.a, 4.4.5.4.10.b, and 4.4.5.4.10.c.

2 The NDE is the value provided by the NRC in GL 95-05.

SURVEILLANCE REQUIREMENTS (continued)

11. F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.11 inches (not including eddy current uncertainty).

12. F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

13. Tube Repair refers to sleeving as described by the reports listed in 4.4.5.4.c which are used to maintain a tube in service or return a tube to service. Tubes with degradation indications of less than the plugging limit may be preventively sleeved at the Owner's discretion. This includes removal of plugs that were installed as a corrective or preventive measure. A tube inspection per 4.4.5.4.a.8 is required prior to returning previously plugged tubes to service. Further restrictions regarding identified indications and their proximity to the joint areas of various sleeving processes may be applicable.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging or sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

c. Steam generator tube repairs may be made in accordance with the methods described in either WCAP-12623, WCAP-13088 (Rev. 3), or CEN-313-P.

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice 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 and sleeves inspected.

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

3. Identification of tubes plugged or sleeved.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the Commission 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 of 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 Commission and provide an assessment of the safety significance of the occurrence.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 600 gallons per day total primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. Seal line resistance greater than or equal to 2.27E-1 ft/gpm^2 and,

f. The leakage from each Reactor Coolant System Pressure Isolation Valves specified in Table 3.4-0 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System average pressure within 20 psi of the nominal full pressure value.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any reactor coolant system pressure isolation valve(s) leakage greater than the above limit, declare the leaking valve inoperable and isolate the high pressure portion of the affected system from the low pressure portion by the use of a combination of at least two closed valves, one of which may be the OPERABLE check valve and the other a closed de-energized motor operated valve. Verify the isolated condition of the closed de-energized motor operated valve at least once per 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Specification 3.4.6.2.e is applicable with average pressure within 20 psi of the nominal full pressure value.

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY

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 a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. 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 plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system. The allowable primary-to-secondary leak rate is 150 gallons per day per steam generator. Axial or circumferentially oriented cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Leakage in excess of this limit will require plant shutdown and an inspection, during which the leaking tubes will be located and plugged or repaired. A steam generator while undergoing crevice flushing in Mode 4 is available for decay heat removal and is operable/operating upon reinstatement of auxiliary or main feed flow control and steam control.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the repair limit which is defined in Specification 4.4.5.4.a. 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.

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY (Continued)

The voltage-based repair limits of these surveillance requirements (SR) implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) 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 SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no NDE detectable cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of these SRs requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then 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 degradation 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 degradation 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 in SR 4.4.5.4.10.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. 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 (EOC) 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 SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should provided per the GL section 6.b (c) criteria.

Tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates are plugged or repaired by the criteria of 4.4.5.4.a.10.

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission 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.

Degraded steam generator tubes may be repaired by the installation of sleeves which span the section of degraded steam generator tubing. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with Draft Regulatory Guide 1.121 (August 1976). In addition, a combined allowance of 20 percent of wall thickness is assumed for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

The following sleeve designs have been found acceptable by the NRC staff:

1. Westinghouse Mechanical Sleeves (WCAP-12623)
2. Combustion Engineering Leak Tight Sleeves (CEN-313-P)
3. Westinghouse Laser Welded Sleeves (WCAP-13088, Rev. 3)

Descriptions of other future sleeve designs shall be submitted to the NRC for review and approval in accordance with 10 CFR 50.90 prior to their use in the repair of degraded steam generator tubes. The submittals related to other sleeve designs shall be made at least 90 days prior to use.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitations provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The limitation on seal line resistance ensures that the seal line resistance is greater than or equal to the resistance assumed in the minimum safeguards LOCA analysis. This analysis assumes that all of the flow that is diverted from the boron injection line to the seal injection line is unavailable for core cooling.

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) will minimize the potential for a large leakage event during steam line break under LOCA conditions. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 8.4 gpm which will ensure the calculated offsite doses will remain within 10 percent of the 10 CFR 100 requirements and that control room habitability continues to meet GDC-19. Leakage in the intact loops is limited to 150 gpd. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 8.4 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 8.4 gpm.

Also, the 150 gpd leakage limit incorporated into this specification is more restrictive than the standard operating leakage limit and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend ouside the thickness of the the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected and the plant shut down in a timely manner.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 215 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letters dated June 19, 1996, September 19, 1996, and December 20, 1996, American Electric Power (the licensee) submitted a request to amend the Donald C. Cook Nuclear Plant, Unit 1 (Cook-1) Technical Specifications (TS). The proposed amendment allows the use of a 2.0 volt steam generator (SG) tube support plate (TSP) repair criteria. The voltage-based SG tube repair criteria allow tubes with axially oriented outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the TSPs to remain in service based on the magnitude of the bobbin coil voltage response.

On August 3, 1995, the NRC issued Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," which outlined generic criteria for licensees considering implementation of alternate repair criteria. The licensee's proposed amendment request follows the guidance provided in GL 95-05 as discussed below.

The September 19, 1996, and December 20, 1996, supplements provided additional information that did not change the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

SG tube flaw acceptance criteria (i.e., plugging limits) are specified in the plant TS. The traditional strategy for achieving adequate structural and leakage integrity of the tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Development of minimum wall thickness requirements to satisfy RG 1.121 was governed by analyses assuming a uniform thinning of the tube wall. This assumed degradation mode is inherently conservative for most other forms of SG tube degradation. Conservative repair limits may lead to plugging tubes with adequate structural and leakage integrity for further service.

The staff developed generic criteria for voltage-based limits for ODSCC confined within the thickness of the TSPs. The staff published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in a draft GL titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the *Federal Register* on August 12, 1994 (59 FR 41520). On August 3, 1995, the staff issued GL 95-05 that took into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data made available from European nuclear power plants.

The guidance of GL 95-05 does not set depth-based limits on predominantly axially oriented ODSCC at TSP locations; rather it relies on empirically derived correlations between a nondestructive inspection parameter, the bobbin coil voltage, and tube burst pressure and leak rate. The staff recognizes that although the total tube integrity margins may be reduced following application of voltage-based repair criteria, the guidance in GL 95-05 ensures structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of 10 CFR Part 50 and the guideline values in 10 CFR Part 100. Since the voltage-based repair criteria do not incorporate a minimum tube wall thickness requirement, there is the possibility for tubes with through-wall cracks to remain in service. Because of the increased likelihood of such flaws, the staff included provisions for augmented SG tube inspections and more restrictive operational leakage limits.

The licensee used interim plugging criteria for the last three cycles, with each cycle's application reviewed and approved by the staff [References 1-3]. The licensee's proposed amendment requests a permanent change to the Cook-1 TS to incorporate voltage-based repair criteria per the guidance of GL 95-05. The guidance specifies, in part that (1) the repair criteria are only applicable to predominantly axially oriented ODSCC located within the bounds of the TSPs; (2) licensees perform an evaluation to confirm the SG tubes will retain adequate structural and leakage integrity until the next scheduled inspection; (3) licensees adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes must be periodically removed from the SG to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit be reduced; (6) licensees implement an operational leakage monitoring program; and (7) specific reporting requirements shall be incorporated into the plant TS.

3.0 EVALUATION

The licensee will follow the requested actions of GL 95-05 for implementing the proposed voltage-based plugging criteria. As permitted by GL 95-05, the licensee proposes to (1) use an alternative method to Section 3.c.3 of Attachment 1 to GL 95-05 to address bobbin coil probe wear and (2) include a provision to use alternative inspection techniques in proposed TS 4.4.5.4.10.c.

3.1 Tube Repair Limits

The proposed voltage criteria pertain specifically to predominantly axially oriented ODSCC flaws, and the proposed criteria (1) permit indications confined to within the thickness of the TSPs with bobbin voltages less than or equal to 2.0 volts to remain in service; (2) permit indications confined to within the thickness of the TSPs with bobbin voltages greater than 2.0 volts but less than or equal to the upper voltage limit to remain in service if an RPC probe or acceptable alternative inspection does not detect degradation; and (3) require indications confined to within the thickness of the TSPs with bobbin voltages greater than the upper voltage limit be plugged or repaired.

The licensee's proposed repair limits are based on the use of a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data. The licensee will use the burst pressure versus bobbin voltage correlation containing all applicable data consistent with the guidance in GL 95-05.

The proposed lower voltage limit of 2.0 volts is consistent with the recommended value specified in GL 95-05 for 7/8 inch SG tubing. The upper voltage limit is based on the lower 95 percent prediction interval of the burst pressure/bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95 percent confidence level. This voltage is further reduced to account for uncertainty in the nondestructive examination (NDE) technique and flaw growth over the next operating cycle. Because licensees periodically update the burst pressure/bobbin voltage database with pulled tube data, the upper voltage limit may vary as additional data are included in the correlation.

3.2 Inspection Issues

In the September 19, 1996 submittal, the licensee proposed to use an alternative to the probe wear reinspection requirements of GL 95-05. The industry approach, developed through the Nuclear Energy Institute (NEI), is such that if any of the probe wear standard signal amplitudes prior to probe replacement exceed the +/- 15 percent limit, all tubes having indications with voltage responses measured at 75 percent or greater of the lower repair limit must be reinspected with a bobbin probe satisfying the +/- 15 percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The NRC staff completed a review of the proposed alternative method and concluded the approach is acceptable because it provides adequate flaw detection sensitivity. [Reference 4]. Therefore, the licensee's proposal to follow the industry approach to address bobbin coil probe wear is acceptable.

In the laboratory and field studies supporting the alternative probe wear criteria, the correlation of worn probe voltages with new probe voltages shows that for all significant voltage levels, the worn probe voltages are never less than 25% of the new probe voltage [Reference 5]. However, in a recent licensee 90 Day Report submittal, a comparison made between the worn probe voltage and the new probe voltage resulted in two data points where the worn probe voltage was substantially less than 25% of the new probe voltage. The licensee evaluated these indications and concluded the criteria to retest tubes with worn probe voltages above 75% of the repair limit is adequate and generally conservative due to the average trend for worn probe volts to exceed new probe voltages. Comparison of the actual and projected end-of-cycle voltages did not show anything unusual attributable to the alternate probe wear criteria. The staff concludes the aforementioned probe wear results do not indicate an immediate need to modify the NEI alternative probe wear criteria. However, the staff will continue to monitor the 90 Day reports of licensees using this approach to probe wear.

The proposed amendment includes a modification to TS 4.4.5.4.10.c. The change allows indications of potential degradation with bobbin coil signal voltages between the lower and upper repair limit to remain in service if they are not confirmed by either an RPC probe or an acceptable alternative inspection technique. The licensee stated in the September 19, 1996 letter that all acceptable alternative inspection techniques will be qualified using similar criteria used for qualifying RPC probes. The NRC staff encourages licensees to use the most sensitive inspection techniques available. As such, the staff concludes the proposed modification of TS 4.4.5.4.10.c to allow the use of alternative inspection techniques is acceptable.

3.3 Structural and Leakage Integrity Assessments

The NRC staff guidance for implementation of voltage repair criteria ensures SG tubes will retain adequate structural integrity during the full range of normal, transient and postulated accident conditions with adequate allowance for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. Tube structural limits based on RG 1.121 criteria require maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. Because the GL 95-05 criteria address tubes affected with ODSCC confined to within the thickness of the TSP during normal operation, the staff concluded the structural constraint provided by the TSP ensures all tubes to which the voltage-based criteria apply will retain a margin of 3 with respect to burst under normal operating conditions, consistent with the criteria of RG 1.121. For a postulated main steam line break (MSLB) accident, however, the TSP may displace axially during blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the TSP. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated MSLB conditions.

In order to confirm the SG tubes will retain adequate structural and leakage integrity until the next scheduled inspection, GL 95-05 describes the methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected SG during a postulated MSLB event. To complete these assessments, the licensee proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated December 1996. The conditional burst probability and accident leak rate calculations should incorporate all available pulled tube data. The licensee referenced the support databases approved by the staff in Reference 6. The datasets contain all applicable data supporting the empirical models for use of these voltage-based repair criteria.

3.3.1 Conditional Probability of Burst

The licensee proposes to follow the methodology described in Revision 1 of WCAP-14277 for performing a probabilistic analysis to quantify the potential for SG tube ruptures given an MSLB event. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per cycle in accordance with GL 95-05. This threshold value provides assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for SGs contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates ODSCC confined to within the thickness of the TSP could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation assumed and evaluated as acceptable in NUREG-0844. The NRC staff concludes the licensee's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable.

3.3.2 Accident Leakage

The licensee proposes to follow the methodology described in Revision 1 of WCAP-14277 for calculating the SG tube leakage from the faulted SG during a postulated MSLB event. The model consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model). The NRC staff concludes the licensee's proposed methodology for calculating the tube leakage is consistent with the guidance in GL 95-05 and is acceptable.

3.3.3 Primary-to-Secondary Leakage During Normal Operation

An important implication of using voltage-based SG tube repair criteria is tubes may have or may develop through-wall or near through-wall cracks during an operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents.

The staff concludes adequate leakage integrity during normal operation is reasonably assured by the TS limits on allowable primary-to-secondary leakage. GL 95-05 specifies the operational leakage limits of the plant TSs should be reduced to 150 gallons per day (gpd). Cook-1 TSs currently limit the primary-to-secondary leakage through one SG to 150 gpd. This requirement is consistent with the guidance in GL 95-05 and is therefore acceptable.

3.4 Degradation Monitoring

To confirm the nature of the degradation occurring at the TSP elevations, tubes are periodically removed from the SGs for destructive analysis. Tube pulls can confirm that the nature of the degradation observed at these locations is predominantly axially oriented ODSCC, provide data for assessing the reliability of the inspection methods, and supplement the existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 contains guidance stating licensees should remove at least two tube specimens with the objective of retrieving as many intersections as practical (minimum of four intersections) during the plant SG inspection outage preceding initial application of the voltage-based repair criteria. On an ongoing basis, additional tube specimen removals (minimum of two intersections) should be obtained at the first refueling outage following 34 effective full power months of operation or at the maximum interval of three refueling outages after the previous tube pull. Alternatively, the licensee may participate in an industry-sponsored tube pull program endorsed by the staff as described in GL 95-05.

Upon initial implementation of interim voltage-based alternate repair criteria in 1992 (EOC-12), the licensee removed 3 tubes for burst and leak rate testing and metallographic examination from the Cook-1 SGs. The metallurgical examination confirmed the dominant degradation mechanism for the indications at the TSP elevations was axially oriented ODSCC. The staff concludes the tubes removed from the Cook-1 SGs satisfy the initial tube pull criteria in GL 95-05. The licensee plans to remove at least one tube in the 1997 outage (EOC-15), and future tube pulls will be scheduled in accordance with GL 95-05.

3.5 Technical Specification Changes

The licensee proposed modifications to TS 4.4.5.2.e which remove the cycle 15 limitation on implementation of the steam generator tube/tube support plate plugging criteria and to TS 3.4.6.2.c. which remove the cycle 15 limitation on primary-to-secondary leakage limits through the steam generators. These changes allow use of the 2.0 volt tube repair limits on a continuing basis, are consistent with the prior discussions and are therefore acceptable.

The licensee proposed modifications to TS 4.4.5.4.10., TS 4.4.5.4.10.a. and TS 4.4.5.4.10.b. which provide more specificity than the previous TS which were granted on a cycle-by-cycle basis. The proposed TS are consistent with the proposed repair criteria discussed above and are therefore acceptable.

Proposed changes to TS 4.4.5.4.10.c. were addressed in section 3.2 of this SE.

A new TS 4.4.5.4.10.d. was proposed which provides specific repair limits if an unscheduled mid-cycle inspection is performed. The proposed limits are consistent with and based on GL 95-05 and are therefore acceptable.

Proposed modifications to 4.4.5.5.d. provide for reporting requirements, including the reporting of an unacceptable calculated conditional burst probability, which are consistent with the previous changes and are therefore acceptable.

Bases changes providing supporting discussion for the associated TS changes were also proposed and are acceptable.

4.0 SUMMARY

The licensee submitted an application for a TS amendment to permit the use of voltage-based SG tube repair criteria at Cook-1. The staff reviewed the proposed amendment and concluded the methods proposed by the licensee are consistent with the guidance in GL 95-05. The staff concludes adequate structural and leakage integrity can be ensured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied. The staff's approval of the proposed voltage-based repair criteria is based in part on the licensee being able to successfully demonstrate after each inspection outage the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable per the guidance in GL 95-05.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (61 FR 40022). The amendments also change record-keeping or reporting requirements. Accordingly, the amendments meet 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 amendments.

7.0 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

References:

1. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to License Amendment No. 166 to Facility Operating License DPR-58, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant, Unit No. 1, Docket No. 50-315, dated July 29, 1992.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-58, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant - Unit 1, Docket No. 50-315, dated March 15, 1994.

3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 200 to Facility Operating License No. DPR-58, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant - Unit No. 1, Docket No. 50-315, dated September 13, 1995.

4. Letter from B. Sheron (NRC) to A. Marion (NEI), dated March 18, 1996.

5. Letter from A. Marion (NEI) to B. Sheron (NRC), "Eddy Current Probe Replacement Criteria for Use in ODSCC Alternate Repair Criteria," dated January 23, 1996.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 198 to Facility Operating License No. DPR-66, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, dated April 1, 1996.

Principal Contributor: Stephanie M. Coffin, NRR/DE/EMCB

Date: March 13, 1997

March 13, 1997

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENTS RE: ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBING (TAC NO. M95894)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 215 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 19, 1996, and supplemented September 19, 1996, and December 20, 1996.

The amendments revise the TS to allow a permanent extension of the interim steam generator tube voltage-based repair criteria for steam generator tubes used in Cycles 13, 14 and 15 at the Donald C. Cook Nuclear Power Plant, Unit 1. The inspection and analysis methodology of the voltage-based plugging criteria approved for Cycle 15 was modified to incorporate the staff recommendations stated in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," issued August 3, 1995.

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

Sincerely,

Original signed by:

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315
Enclosures: 1. Amendment No. 215 to DPR-58
2. Safety Evaluation
cc w/encl: See next page

DISTRIBUTION: See attached list

DOCUMENT NAME: G:\DCCOOK\CO95894.AMD * see previous concurrence

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