September 13, 1995

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, OH 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: AN INTERIM STEAM GENERATOR TUBE SUPPORT PLATE PLUGGING CRITERION FOR FUEL CYCLE 15 (TAC NO. M91582)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 200 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 3, 1995, as supplemented April 25, 1995.

The amendment modifies the technical specifications to extend the interim steam generator tube plugging criteria used in Cycle 14 to the next operating cycle (Cycle 15).

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-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 200 to DPR-58
2. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\WPDOCS\DCCOOK\C091582.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	LA:PD31	E	PM:PD31	E	OGC		D:PD31	
NAME	CJamerson		JHickman:				JHannon	
DATE	9/7/95		9/7/95		9/12/95		9/12/95	

OFFICIAL RECORD COPY

DF01

Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, Michigan 49106

Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
1 Cook Place
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington, DC 20037

Mayor, City of Bridgman
Post Office Box 366
Bridgman, Michigan 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, Michigan 48909

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 48909

Mr. S. Brewer
American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43215

DATED: September 13, 1995

AMENDMENT NO. 200 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK-UNIT 1

Docket File
PUBLIC
PDIII-1 Reading
E. Adensam (e-mail)
J. Hannon
C. Jamerson
J. Hickman (2)
OGC-WF
G. Hill (2)
C. Grimes, O-11F23
P. Rush
ACRS (4)
W. Kropp, RIII
SEDB

cc: Plant Service list

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. 200
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 February 3, 1995, as supplemented April 25, 1995, 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. 200 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 13, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 200

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 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
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 4-2b
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

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

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for 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. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All tubes that previously had detectable wall penetrations (greater than or equal to 20%) that have not been plugged or repaired by sleeving in the affected area.

* This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions for Operation for Specification 3.4.1.3 are maintained.

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 for one fuel cycle (Cycle 15) 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.

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)

Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency 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 4.4.5.3.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 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

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

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings 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 or sleeve.

3. Degraded Tube or Sleeve means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. Percent Degradation means the amount of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit.

6. Repair/Plugging Limit means the imperfection depth at or beyond which the tube or sleeved tube shall be repaired or removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40 percent or more of the nominal tube wall thickness shall be plugged or repaired prior to returning the steam generator to service. This definition does not apply to the portion of the tube in the tubesheet below the F* distance for F* tubes. Any sleeve which, upon inspection, exhibits wall degradation of 29 percent or more of the nominal wall thickness shall be plugged prior to returning the steam generator to service. In addition, any sleeve exhibiting any measurable wall loss in sleeve expansion transition or weld zones shall be plugged. This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 4.4.5.4.a.10 for the plugging limit applicable to these intersections.

7. Unserviceable describes the condition of a tube or sleeve 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 4.4.5.3.c, above.

8. Inspection determines the condition of the steam generator tube or sleeve from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube in which the tube support plate elevation interim plugging limit has been applied, the inspection will include all the hot leg intersections and all cold leg intersections down to, at least, the level of the last crack indication.

9. Sleeving a tube is permitted only in areas where the sleeve spans the tubesheet area and whose lower joint is at the primary fluid tubesheet face.

SURVEILLANCE REQUIREMENTS (continued)

10. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:

    a. Degradation attributed to ODSCC within the bounds of the tube support plate with bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.

    b. Degradation attributed to ODSCC 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. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

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.

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 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 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 staff prior to returning the steam generators to service should any of the following conditions arise:

   1. If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing "Standard Review Plan - NUREG-0800" assumptions) during the previous operating cycle.

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

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

   4. If the calculated conditional burst probability, as calculated per WCAP-14277, exceeds 1×10^{-2}, notify the NRC 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 for Fuel Cycle 15,

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 for one fuel cycle (Cycle 15). 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.

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)

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) for Fuel Cycle 15 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 12.6 gpm which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. 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 12.6 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 12.6 gpm.

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.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Cook Nuclear Plant site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

Offsite doses following a main steam line break are limited to 10 percent of the 10 CFR 100 guideline. The restriction is based on a Cook Nuclear Plant site-specific radiological evaluation that assumes a post-accident primary-to-secondary leak rate of 120 gpm in the faulted loop and a primary coolant specific activity concentration corresponding to 1% fuel defects (approximately 4.6 microCuries/gram dose equivalent I-131), rather than a specific activity of 1.0 microCuries dose equivalent I-131.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

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

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

1.0 INTRODUCTION

By letter dated February 3, 1995, as supplemented April 25, 1995, the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed amendment would revise, in part, TSs 4.4.5.2, 4.4.5.3, 4.4.5.4, 4.4.5.5, and 3.4.6.2 for D.C. Cook Unit 1, for Cycle 15 operation to permit the use of voltage-based steam generator tube repair criteria. The voltage-based steam generator tube repair criteria allows axially oriented outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates to remain in service based on the magnitude of the eddy current voltage response.

The NRC staff has reviewed the submittals noted above and additional information obtained in a phone call with the licensee on July 28, 1995. The following is the staff's evaluation of the licensee's proposed TS amendment.

2.0 BACKGROUND

The staff has previously approved similar requests from the licensee to apply voltage-based tube repair criteria at D.C. Cook Unit 1. Implementation of voltage-based tube repair criteria for fuel Cycle 13 was approved as documented in a letter to the licensee dated July 29, 1992, "Donald C. Cook Nuclear Plant, Unit 1 - Amendment No. 166 to Facility Operating License No. DPR-58." Similarly, implementation of voltage-based tube repair criteria for fuel Cycle 14 was approved as documented in an amendment to the license dated March 15, 1994, "Donald C. Cook Nuclear Plant, Unit No. 1 - Issuance of Amendment RE: Incorporation of 2.0 Volt Steam Generator Tube Support Plate Interim Plugging Criteria For Cycle 14." In the previous voltage-based tube repair amendments listed above, the staff concluded that the tube repair limits and leakage limits would ensure adequate structural and leakage integrity for indications accepted for continued service at D.C. Cook Unit 1, consistent with applicable regulatory requirements. This evaluation addresses comparable tube repair criteria for operating Cycle 15; however, in this amendment, the licensee has proposed to increase the voltage limits from 2.0/3.6 volts to 2.0/5.6 volts.

The staff has been developing generic criteria for voltage-based limits for ODSCC confined within the thickness of the tube support plates. The staff has 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 generic letter (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, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," that took into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data which have been made available from European nuclear power plants. The licensee submitted its amendment request prior to the date that the NRC issued GL 95-05. Consequently, the amendment request was based on the guidance in the draft GL.

The licensee's current proposal is applicable to Cycle 15 operation and is similar to the licensee's prior amendment proposals which were approved as documented in the references mentioned previously. Furthermore, the licensee's submittal is consistent with GL 95-05 except as noted below.

3.0 PROPOSED INTERIM TUBE REPAIR CRITERIA

Donald C. Cook Nuclear Plant, Unit 1, Technical Specifications 4.4.6.2, 4.4.6.4, 4.4.6.5, and 3.4.7.2 and Bases 3/4.4.6 and 3/4.4.7 would be revised by this amendment request to specify the voltage-based tube repair criteria for ODSCC confined to within the thickness of the tube support plates. Modifications have been made to the previously approved (Cycles 13 and 14) TSs pertaining to the implementation of the voltage-based tube repair criteria so as to make the currently proposed TSs similar to those permitted by GL 95-05. The changes in the TSs for Cycle 15 implementation of the voltage-based tube repair criteria include, in part:

a. Specifying that tube support plate indications left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during the following refueling outages.

b. Specifying that the implementation of the steam generator 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 ODSCC indications. The determination of the cold-leg 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.

c. Changing the Cycle 14 repair limits for tube support plate intersections with indications of ODSCC from 2.0 and 3.6 volts to the following for Cycle 15:

    1. Degradation attributed to ODSCC within the bounds of the tube support plate with bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service.

2. Degradation attributed to ODSCC within the bounds of the tube support plate with bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in c.3 below.

3. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 5.6 volts will be plugged or repaired.

d. Adding the following reporting requirements:

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 the estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing "Standard Review Plan - NUREG-0800" assumptions) during the previous operating cycle.

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

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

4. If the calculated conditional burst probability exceeds 1 x 10^{-2} per reactor-year, as calculated per WCAP-14277, "SLB [Steam Line Break] Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," notify the NRC and provide an assessment of the safety significance of the occurrence.

e. Continue to specify a limit on primary-to-secondary leakage of 150 gallons per day through any one steam generator for Cycle 15.

In addition to the above TS changes, the licensee has also made the following commitments for implementing the voltage-based tube repair criteria:

1. The requested actions of GL 95-05 will be followed with the exception of the use of a probe wear standard and the use of bobbin coil probes with the tolerance specified in Section 3.c.2 of the GL. These exceptions are discussed in Sections 4.1 and 4.2 of this evaluation. In addition, the licensee has proposed not to include the mid-cycle equation for determining the voltage limits in the event of a forced outage not attributable to ODSCC at the tube support plates.

2. Calculation of the conditional probability of burst and total leak rate during a main steamline break (MSLB) will follow the methodology described in WCAP-14277. As discussed in WCAP-14277, these methods are intended to be in accordance with the draft GL on voltage-based tube repair criteria.

The methods as specified in the draft generic letter are unchanged in GL 95-05.

3. The NRC will be notified prior to restart if any indications of primary water stress corrosion cracking (PWSCC) are detected at the tube support plate elevations. Furthermore, the data analysts will be briefed on the possibility that PWSCC can occur at tube support plate elevations.

4. No distribution cutoff will be applied to the voltage measurement variability distribution.

5. All intersections where copper signals interfere with the detection of flaws will be inspected with a motorized rotating pancake coil probe.

6. All intersections with large mixed residuals will be inspected with a rotating pancake coil probe.

7. All bobbin flaw indications with voltages greater than 1.5 volts will be inspected with a rotating pancake coil probe.

4.0 EVALUATION

4.1 Inspection Issues

The licensee's inspection program is consistent with the guidance of GL 95-05 with the exception of the probe wear re-inspection requirements and the use of bobbin coil probes with the appropriate tolerances specified in Section 3.c.2 of the GL. For the probe wear re-inspection requirements, the licensee proposes to use the same practices used during the last D.C. Cook Unit 1 steam generator inspections as discussed in Attachment 5 to a letter from the licensee to the NRC dated December 15, 1993. The requirements state that if any of the probe wear standard signal amplitudes prior to probe replacement exceed the ±15 percent limit, by a value of "X%", then any indications measured since the last acceptable probe wear measurement that are within "X%" of the plugging limit will be reinspected with the new probe. Alternatively, the voltage criterion may be lowered to compensate for the excess variation.

Section 3.c.2 of GL 95-05 specifies that the voltage response for the 40-percent to 100-percent through-wall holes of new bobbin coils calibrated on the 20-percent through-wall holes should not differ from the nominal voltage by more than ±10 percent. The licensee indicated that bobbin coil probes with such tolerances would not be available until after the licensee inspects the D.C. Cook Unit 1 steam generators in the fall 1995 outage.

With respect to the use of the proposed alternate procedures for re-inspecting tubes that fail to meet the probe wear criterion, the staff has concluded that alternate methods may be used provided an assessment is performed demonstrating that (1) they provide equivalent detection and sizing capability on a statistically significant basis when compared to the guidance in GL 95-05 and (2) they are consistent with current methods for determining the end-of-cycle (EOC) voltage distributions which are used in the tube integrity analyses. These assessments, along with the statistical criteria for demonstrating that the techniques are equivalent, should be provided to the NRC for review and approval. With respect to this cycle specific application, however, the NRC staff has concluded that the methods to meet the probe wear criterion are acceptable.

The licensee indicated that bobbin coil probes with the voltage response tolerances specified in GL 95-05 will not be available until approximately 6 months after the NRC issues the GL. The scheduled date for the inspection of the D.C. Cook Unit 1 steam generators is less than 2 months following the release of GL 95-05. The availability of the appropriate bobbin coil probes will be limited at the time of the inspection. Due to the difficulty in obtaining bobbin coil probes with the response characteristics specified in Section 3.c.2 of GL 95-05, the licensee's decision not to inspect with such probes in the Cycle 15 refueling outage is acceptable to the NRC staff.

As a result of the potential for the possible development of PWSCC flaws at dented tube support plate intersections, the licensee will brief its eddy current analysts of the potential for PWSCC to occur at these locations. Furthermore, the licensee has agreed to notify the NRC prior to plant restart if any PWSCC indications are detected at the tube support plate elevations. The staff notes that if PWSCC is detected at tube support plate elevations, an evaluation to ensure voltage-based repair criteria are only applied to ODSCC indications may need to be performed. The staff concludes that the inspection guidelines submitted by the licensee are acceptable since the proposed repair criteria are limited to one cycle. In addition, the staff finds that the calibration, recording, and analysis requirements are consistent with the methodology used in the development of the databases and supporting evaluations.

4.2 Structural Integrity

4.2.1 Deterministic Structural Integrity Assessment

The licensee's tube repair limits are based on a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data. This correlation is similar to that used in approving the voltage limits in the licensee's previous submittals and those used in GL 95-05. The staff finds the licensee's proposed voltage limits acceptable given the current burst pressure/bobbin voltage database, the licensee's growth rates, and the non-destructive examination uncertainty estimates.

To confirm the nature of the degradation occurring at the tube support plate elevations, tubes are periodically removed from the steam generators for destructive analysis. Tube pulls confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC and also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 contains guidance that states utilities should periodically remove at least two tube support plate intersections with large voltage indications for destructive examination. For subsequent tube pulls after the initial application of voltage-based repair criteria, GL 95-05 states that licensees should retrieve as many intersections as practical (minimum of two intersections) following accumulation of 34 effective full power months of operation or at a maximum interval of three refueling outages, whichever is shorter, following the previous tube pull. In 1992, the licensee removed nine tube support plate intersections for metallographic examination, burst testing, and leak rate testing from the D.C. Cook Unit 1 steam generators. As of the Cycle 15 refueling outage, D.C. Cook Unit 1 will have accumulated approximately 29.5 effective full power months since the 1992 refueling outage. In addition, the Cycle 15 refueling outage will only be the second outage since the previous tube pulls. Therefore, the licensee has elected to not remove any tube support plate intersections in accordance with the guidance in GL 95-05.

Metallurgical examination performed on the tubes removed during the 1992 refueling outage confirmed that the dominant degradation mechanism for the indications at the support plate elevations is axially oriented ODSCC. The maximum voltage of the intersections removed was 2.02 volts. The staff believes that no additional pulled tube data is required to support implementation of the 2.0-volt voltage limit for the next operating cycle (Cycle 15) provided no unusual inspection findings are identified during the inspection.

4.2.2 Probabilistic Structural Integrity Assessment

A probabilistic analysis for the potential for steam generator tube ruptures, given an MSLB, has been performed for the previous applications of this tube repair criteria. The draft GL contains additional guidance on this analysis. The licensee intends to perform this calculation per the guidance in the draft GL which will most likely result in a higher conditional probability of burst than would have been obtained using the previous methodology since it includes parametric uncertainty. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per reactor-year according to the guidance in the draft GL. This threshold value will provide 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 steam generators 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 that ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844. The guidelines in GL 95-05 for calculating the potential for tube ruptures given an MSLB are equivalent to those of the draft GL.

The licensee referenced WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated January 1995, as a document containing the details of the methodology for calculating the conditional probability of burst given an MSLB. The NRC staff previously approved the use of the methodology in WCAP-14277 as documented in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 106 to Facility Operating License NPF-8, Southern Nuclear Operating Company, Inc., Joseph M. Farley Nuclear Plant, Unit 2, Docket No. STN 50-364 dated April 7, 1995. The staff concludes that the licensee's proposed methodology is consistent with the guidance in GL 95-05 and is acceptable for use in this outage-specific application.

4.3 Leakage Integrity

4.3.1 Normal Operational Leakage

Consistent with prior amendments approving the use of the voltage-based repair criteria at D.C. Cook Unit 1, the licensee will continue to limit the amount of operating leakage through any one steam generator to 150 gallons per day. This requirement will be in effect for operation during Cycle 15.

4.3.2 Accident Leakage

The licensee indicated that it will calculate the leakage and MSLB tube burst probability following the guidance of the draft GL. The guidelines of the draft GL are consistent with GL 95-05 with regard to leakage and conditional burst probability calculations. In order to complete these calculations, the licensee will follow the methodology outlined in WCAP-14277. The model for calculating the steam generator tube leakage from the faulted steam generator during a postulated MSLB 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 (POL) 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 calculational methodology being proposed by the licensee for D.C. Cook Unit 1 for determining the amount of primary-to-secondary leakage under postulated accident conditions has previously been reviewed and approved by the staff as documented in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 106 to Facility Operating License NPF-8, Southern Nuclear Operating Company, Inc., Joseph M. Farley Nuclear Plant, Unit 2, Docket No. STN 50-364 dated April 7, 1995. The staff finds this methodology acceptable for an assessment of the D.C. Cook Unit 1 steam generators for Cycle 15. The staff notes that all applicable data should be included in the probability of leakage and conditional leak rate databases when performing this calculation. The staff notes that some minor variations in the details of the modeling may be necessary for the case where the p-value test is invalid at the 5 percent level. The staff, however, finds the licensee's proposal to perform the calculation using a methodology consistent with the guidance of GL 95-05 acceptable.

The licensee completed calculations for the allowable steam generator leak rate in the faulted steam generator for its application to amend the D.C. Cook Unit 1 TS to apply a voltage-based plugging criteria during the refueling outage for Cycle 14. The leakage value is intended to be consistent with maintaining the radiological consequences of a release outside containment to within a small fraction of the guideline values in 10 CFR Part 100. As a result, if the primary-to-secondary leakage during a postulated MSLB is less than this allowable limit, the steam generator tubing will maintain adequate leakage integrity under these conditions. The staff previously reviewed and approved the licensee's calculations for determining the maximum allowable primary-to-secondary leakage in the license amendment approving the use of the voltage-based criteria for D.C. Cook Unit 1 for fuel Cycle 14. The staff concludes that the licensee's calculation of the allowable steam generator leak rate is acceptable since the previous results of these calculations remain valid for operation in Cycle 15.

5.0 SUMMARY

The licensee submitted an application for a one cycle amendment to the Donald C. Cook Nuclear Power Plant Unit 1 TS which would permit the use of voltage-based steam generator tube repair criteria. The amendment application was submitted prior to the date when the NRC issued a final position regarding the implementation of voltage-based repair criteria in GL 95-05. Consequently, the licensee's submittal follows the guidelines provided in the draft version of the GL. The staff reviewed the proposed one cycle amendment to the D.C. Cook Unit 1 TS and concluded that the methods proposed by the licensee are also consistent with the guidance in GL 95-05 except as noted above. The staff concludes that 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 during Cycle 15 at the Donald C. Cook Nuclear Power Plant, Unit 1. The staff's approval of the proposed voltage-based repair criteria is based, in part, on the licensee being able to demonstrate that the projected EOC conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable following the Cycle 15 refueling outage.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (60 FR 37093). 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 the amendment.

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

Principal Contributor: P. Rush

Date: September 13, 1995