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

January 30, 2001

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT (TAC NO. MB0156)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your application dated September 30, 2000, as supplemented November 22, and December 20, 2000.

The amendment would allow an extension of the steam generator tube inspection surveillance requirements of Technical Specification Surveillance Requirement 4.4.5.3. Specifically, the licensee requested to extend the required inspection interval from 40 to 56 calendar months.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-316

Enclosures: 1. Amendment No. 232 to DPR-74
2. Safety Evaluation

cc w/encls: See next page

MRR-058

January 30, 2001

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/RA/

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated September 30, 2000, as supplemented November 22, and December 20, 2000, 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 the addition of paragraph 2.C.(3) (w) to the license. The changes are indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: License Page

Date of Issuance: January 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 232

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
License Page -9-	License Page -9-

(v) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall be described in the station chemistry manual and shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

(w) The steam generator tube inspection surveillance maximum inspection interval of Technical Specification 4.4.5.3 is extended until the start of cycle 13, but no later than June 30, 2002.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By application dated September 30, 2000, as supplemented November 22, and December 20, 2000, the Indiana Michigan Power Company (the licensee) requested an amendment to add a condition to the license for the Donald C. Cook (D.C. Cook) Nuclear Plant, Unit 2. The proposed license condition would allow an extension of the steam generator (SG) tube inspection surveillance requirements of Technical Specification Surveillance Requirement 4.4.5.3. Specifically, the licensee requested to extend the required inspection interval from 40 to 56 calendar months. By letter dated November 7, 2000, the Nuclear Regulatory Commission (NRC) staff issued a request for additional information (RAI). By letters dated November 22, and December 20, 2000, the licensee responded to the RAI.

The November 22, and the December 20, 2000, supplemental letters, did not change the scope of the proposed action and did not change the NRC's preliminary no significant hazards consideration determination.

2.0 BACKGROUND

For much of the first 40 calendar months of the current cycle (i.e., approximately 30 months), the licensee was in an extended shutdown. As a result, the SGs were not exposed to the high temperature conditions generally required for corrosion-induced degradation of the SG tubes. The actual operating time between inspections with approval of the proposed extension would only be approximately 18 effective full power months, much less than typically would occur in a 40 calendar month inspection interval.

The fundamental goal of the SG tube inspection and repair criteria is to ensure that the structural and leakage integrity of the SG tube bundle is maintained for the period of time between the inspections. Structural integrity involves demonstrating the tubes are capable of withstanding the loadings specified in the American Society of Mechanical Engineers (ASME) Code and Regulatory Guide 1.121, "Bases for Plugging Degraded Presssurized-Water Reactor (PWR) Steam Generator Tubes." Leakage integrity involves demonstrating the dose consequences from SG tube leakage are acceptable per General Design Criteria 19 of 10 CFR Part 50, Appendix A, and 10 CFR Part 100.

3.0 EVALUATION

3.1 SG Design and Experience

D.C. Cook Unit 2, replaced their original SGs in 1988 with SGs with alloy 690 thermally treated tubes. This tube material is the "material of choice" for newer SGs in the United States due to its increased resistance to degradation when compared to the original tube material (i.e., alloy 600 mill annealed). The current D.C. Cook Unit 2 SGs also incorporate other design features intended to reduce the potential for SG tube degradation including an increased row 1 U-bend radius, stainless steel quatrefoil hole tube support plates, and tubes which have been hydraulicly expanded against the tube sheet throughout the tubesheet thickness.

The experience with SGs with this tube material in the United States has been excellent. No corrosion related degradation has been detected to-date, although the operating experience is limited. The plant with the longest operating time with this SG tube material has 78 effective full power months of operation at a T-hot of 618 °F (as of their last SG inspection). D.C. Cook Unit 2, has 71 EFPM with a T-hot of 607 °F as of their last SG inspection. Some minor tube plugging/repair has occurred in alloy 690 thermally treated tubes primarily as a result of loose part damage, mechanical damage, and/or tube wear.

3.2 Prior Inspections

Following the 100 percent baseline inspection (performed prior to placing the SGs in service), the licensee for D.C. Cook Unit 2, performed the following bobbin coil inspections.

Bobbin Coil Inspections

Year	SG 21	SG 22	SG 23	SG 24	Notes
1990		6.5%	6.5%		No degradation
1992	6.5%			6.5%	No degradation
1994		6.5%	6.5%		Mechanical damage due to pressure pulse cleaning in SG 22 and SG 23 resulted in plugging 9 tubes. Additional bobbin coil exams were also performed in areas susceptible to the mechanical damage.
1997	50%	50%	50%	50%	One 28% through-wall defect attributed to a foreign object was located and plugged. 4 indications of tube wear was also located on 2 tubes.

In addition to the above bobbin coil examinations, the licensee also performed a 20 percent motorized rotating pancake coil (MRPC) examination in SG 24 at the top of the hot-leg side of the tubesheet in 1997. Four tubes were found to have indications of foreign object induced wear during these MRPC examinations and were plugged since a qualified sizing technique was not available. These indications were not detected by the bobbin coil. The licensee indicated these rotating probe examinations were performed to confirm the absence of circumferential cracking in response to Generic Letter 95-03, and were not considered within the scope of the technical specification inspection. The NRC staff, however, finds that these inspections are within the scope of the technical specifications. All SG tube inspections

performed must follow the repair criteria and sample expansion criteria of the technical specifications. Given the bobbin coil is not qualified to detect circumferential degradation, the only qualified technique used to inspect for circumferential degradation was the rotating pancake coil probe.

The current industry guidance for performing SG tube inspections calls, in part, for inspecting all tubes not inspected within the previous 60 effective full power months. The industry SG tube inspection guidelines also call for performing a 100 percent inspection after the first cycle of operation. The licensee currently does not meet the guidance for inspecting tubes within 60 effective full power months; however, the licensee intends to inspect the remaining tubes (not previously inspected) in the next refueling outage.

3.3 Degradation

The licensee indicated they had no "active degradation mechanism." An "active" degradation mechanism is defined by the industry as a combination of ten or more new indications of degradation (>20 percent through-wall) and previous indications of degradation which display an average growth rate equal to or greater than 25 percent of the repair limit per cycle in any one SG or, one or more new or previously identified indications of degradation, including cracks, which display a growth greater than or equal to the repair limit in one cycle of operation. The NRC staff finds this terminology misleading since tubes could have degradation which is progressing and still meet the definition of "no active degradation." Degradation due to fretting caused by loose parts and tube wear degradation has been observed at D.C. Cook Unit 2, as discussed below.

Fretting caused by loose parts was detected in the D.C. Cook Unit 2 SGs in 1997. In the areas where the fretting was detected and around the periphery of the tube bundle where loose parts are postulated to enter the tube bundle, inspections were performed by the licensee. Sludge lancing and secondary side inspections were also performed on all four SGs. As a result of these measures and the licensee's foreign material exclusion program, the NRC staff believes there is reasonable assurance of tube integrity for this mechanism until the next scheduled SG tube inspection.

Tube wear has also been detected on two tubes. The depth of the degradation on these tubes ranged from 4 percent to 11 percent through-wall. These tubes were left in service consistent with the technical specifications and were analyzed in the licensee's operational assessment as discussed below.

With respect to corrosion-induced degradation, none has been observed in any of the domestic SGs with alloy 690 thermally treated tubes. However, this does not mean it cannot occur. As such, the NRC staff requested the licensee provide an analysis of when degradation could be expected to occur. Based on an assessment performed by the licensee, corrosion degradation could start to occur after operating the SGs for approximately 13 effective full power years (EFPY). At the time of the next inspection, the licensee will have operated these SGs for approximately 7.3 EFPYs. The NRC staff finds that these types of analyses and models are highly uncertain and therefore, warrant prudent SG tube examinations to ensure their validity. Nonetheless, based on the operating experience of alloy 690 thermally treated tubes and alloy 600 thermally treated tubes (which have been used for longer periods of time and have similar favorable operating experience), the NRC staff concludes that there is reasonable assurance

that tube integrity will not be compromised from corrosion-induced degradation during the remainder of this operating interval.

3.4 Water Chemistry During Layup

Along with tube material, operating temperature and operating time, primary and secondary water chemistry also play a key role in SG tube degradation. As a result, the licensee evaluated its SG water chemistry program to support this amendment request. The licensee concluded that chemistry control during Cycle 11 and the following extended shutdown, did not create conditions that would affect the integrity of the SGs, or their ability to perform their intended safety function. Of particular note, however, is that for a period of approximately six months, the SGs could not be sampled due to replacement of sampling panels, although the licensee verified chemistry was acceptable prior to and subsequent to this period.

3.5 Tube Integrity Assessment

As discussed above, SG tube integrity involves demonstrating the tubes retain adequate structural and leakage integrity. Damage to tubes can occur as a result of mechanical means (e.g., tube wear and foreign objects) and corrosion. As noted above, corrosion related degradation was not considered a credible concern by the licensee for this operating cycle, and the NRC staff finds with this conclusion acceptable.

For foreign object damage, the licensee concluded that such damage need not be addressed in its operational assessment given the foreign material exclusion program, secondary side visual inspection results, performance of eddy current inspections in susceptible locations (e.g., tubes in the periphery), licensee's removal of foreign objects and sludge lancing. The NRC staff finds the above actions will provide reasonable assurance of adequate tube integrity from foreign object damage until the next SG tube inspection.

For tube wear, the licensee addressed two concerns: the indications known to be left in service and the existence of indications in tubes not inspected during the prior outage (and/or missed indications). In this case, the NRC staff finds that the existence of indications in tubes not inspected during the prior outage (and/or missed indications) is the more limiting case for D.C. Cook Unit 2.

In addressing the structural integrity of SG tubes, a determination of the structural limit of the tubes is needed along with an understanding of degradation growth and non-destructive examination (NDE) and material property uncertainties. The licensee provided information related to these items. Specifically, the licensee provided information related to the structural limit, the growth rate, the total uncertainty (material properties and non-destructive examination uncertainties), the repair limit (also referred to as the operational assessment limit), and the condition monitoring limit (also referred to as the structurally significant flaw size and maximum allowable flaw size). The condition monitoring limit represents the structural limit reduced by allowances for total uncertainty (i.e., equivalently, it is the repair limit plus an allowance for growth). The use of this terminology was clarified via conference call on January 3, 2001.

In the determination of the structural limit for tube wear and loose parts, the licensee assumed a flaw which was infinitely long, uniform, and affected 360 degrees of the tube circumference. This flaw geometry tends to be conservative for tube wear, since tube wear typically occurs

over a limited length and circumferential extent (i.e., it only affects the tube near the tube support plate land). The licensee also assumed a flaw which was limited in axial length to 0.2-inch, which is less conservative.

The determination of the appropriate allowances for growth and NDE uncertainty is very challenging. For example, given the limited amount of degradation and the limited inspections performed by the licensee, they can not determine when the degradation started. As a result, the licensee does not know conclusively whether the degradation currently observed started when the SGs were put in service, started at the beginning of the last cycle, or at some mid-point during the last cycle. The licensee assumed the degradation commenced at the beginning of the cycle prior to its detection. If the licensee assumed the degradation started at the time of replacement, the growth rate would be lower (less conservative). However, the licensee could have assumed it started in the middle of the cycle, although the NRC staff finds that under the present circumstances (i.e., state of degradation, nature of degradation), it is reasonable to assume it started at the beginning of the prior cycle.

Based on its analyses, the licensee concluded it could operate for the remainder of the cycle without exceeding the appropriate regulatory criteria. The NRC staff performed a similar analysis using different input parameters for growth and uncertainty and concluded there was reasonable assurance of tube integrity during the remainder of this cycle for indications of tube wear. The NRC staff's analysis assumed the degradation in the uninspected portion of the tube bundle is no worse than what was observed during the last inspection (this assumes no major difference in the propensity for tube wear and the rate of tube wear between the inspected and non-inspected portions of the tube bundle).

For leakage integrity under postulated accident conditions, the licensee assessed leakage through defects known to exist and through tube plugs. Based on its analysis, the licensee concluded there is adequate leakage integrity. The licensee's analysis, however, excluded leakage through tube plugs installed in tubes without known defects (i.e., tubes "administratively" plugged). Based on experience with alloy 600 mill annealed tubes, it appears that degradation can continue to proceed (and possibly initiate) in tubes even after tube plugging. Although the licensee's analysis ignores the leakage contribution from these plugged tubes (i.e., approximately nine tubes), the potential leakage through the plugs is minor and given the number of plugs installed in the SGs, the potential leakage contribution from these plugs would not affect the licensee's conclusion. As a result, the NRC staff finds that the licensee's conclusion that leakage integrity will be maintained until the next planned SG inspection, is acceptable.

4.0 SUMMARY

The NRC staff concludes there is reasonable assurance that steam generator tube integrity will be maintained until the next scheduled inspection (i.e., not to exceed 56 calendar months from the prior inspection), based on the licensee's analyses as confirmed through independent calculations performed by the NRC staff. Licensee conditions and the technical specifications maintain specific parameters for both primary and secondary steam generator water chemistry and temperature which will assure that these parameters will not effect the rate and/or initiation of steam generator tube degradation. This conclusion is primarily based on the plant being in an extended shutdown period under conditions where corrosion-induced and mechanically-induced corrosion (i.e., wear) are considered relatively inactive.

In the unlikely event of a loss of tube integrity, the NRC staff notes the licensee has more restrictive primary-to-secondary leakage limits than currently in their technical specifications (i.e., the unit would be required to shutdown if a primary-to-secondary leakage exceeds 75 gallons per day. Current technical specification limit is 500 gallons per day). These more restrictive limits provide additional confidence that if a leak were to occur, the plant may be shutdown prior to a SG tube rupture. In addition, even if a defect were to instantaneously rupture (from a leakage value below the technical specification limit), the licensee has analyzed a SG tube rupture as part of its design basis.

The original proposed license condition was changed to be more explicit. The changes to the license condition were discussed and agreed to by the licensee in a telephone conversation on January 22, 2001. The final license condition states that the "steam generator tube inspection surveillance maximum inspection interval of Technical Specification 4.4.5.3. is extended until the start of Cycle 13, but no later than June 30, 2002." Based on the above, the NRC staff finds the proposed License Condition to extend Technical Specification Surveillance Requirement 4.4.5.3 from 40 to 56 calendar months is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The NRC has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (65 FR 62387). 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.

6.0 CONCLUSION

The NRC 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 NRC'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: Ken Karowski

Date: January 30, 2001