February 25, 2003

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB4837 AND MB4838)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 274 to Facility Operating License No. DPR-58 and Amendment No. 254 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 11, 2002, as supplemented November 11, 2002.

The amendments would revise the Surveillance Requirements for containment leakage rate testing in TS 4.6.1.2 to allow a one-time extension of the interval between integrated leakage rate tests from 10 to 15 years.

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,

/RA/

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

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 274 to DPR-58
2. Amendment No. 254 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

February 25, 2003

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB4837 AND MB4838)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 274 to Facility Operating License No. DPR-58 and Amendment No. 254 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 11, 2002, as supplemented November 11, 2002.

The amendments would revise the Surveillance Requirements for containment leakage rate testing in TS 4.6.1.2 to allow a one-time extension of the interval between integrated leakage rate tests from 10 to 15 years.

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,
/RA/

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

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 274 to DPR-58
2. Amendment No. 254 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

WBeckner	RPalla	RDennig
PUBLIC	GHill(4)	JStang
PDIII-1 Reading	OGC	THarris
AVegel, RIII	ACRS	
JPulsipher	HAshar	

*******NLO WITH COMMENTS AND CHANGES
**Safety Evaluation input provided by memos dated
ADAMS Accession No. ML030160330 *See previous concurrence

OFFICE	PM:PD3-1	LA:PD3-1	BC:SPLB**	SC:SPSB**	SC:EMEB**	OGC***	SC:PD3-1
NAME	JStang	THarris	SWeerakkody for JHannon	MRubin	DTerao	CBray	LRaghavan
DATE	02/11/03	02/11/03	1/30/03	1/23/03	12/20/02	02/12/03	02/24/03

OFFICIAL RECORD COPY

cc:

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

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

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

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

David W. Jenkins, Esquire
Indiana Michigan Power Company
One Cook Place
Bridgman, MI 49106

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

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

Drinking Water and Radiological
Project Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr. Blvd.
P. O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Scot A. Greenlee
Director, Nuclear Technical Services
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

Michael J. Finissi
Plant Manager
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Joseph E. Pollock
Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 274
License No. DPR-58

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 April 11, 2002, as supplemented November 11, 2002, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 274 , 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 its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 25, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 274

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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 marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 6-2	3/4 6-2

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 254
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 April 11, 2002, as supplemented November 11, 2002, 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-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254 , 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 its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 25, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 254

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 marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 6-2	3/4 6-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 254 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated April 11, 2002, as supplemented November 11, 2002, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would revise the Surveillance Requirements for containment leakage rate testing in TS 4.6.1.2 to allow a one-time extension of the interval between integrated leakage rate tests (ILRTs) from 10 to 15 years.

The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

2.0 BACKGROUND

Pursuant to 10 CFR Part 50, Appendix J, Option B, a Type A test is required to be conducted at a periodic interval based on historical performance of the overall containment system. D. C. Cook Units 1 and 2 TS 4.6.1.2 requires that a program be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. Further, it requires that this program be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

Type A testing is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances.

The most recent two Type A tests were last completed for Unit 1 and Unit 2 on October 1, and May 12, 1992, respectively. The current due dates for the next ILRTs are January 1, and August 12, 2003, respectively. The licensee is requesting an addition to TS 4.6.1.2, “Containment Leakage Rate Testing Program,” which would indicate that they are allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS says that the next Type A test to be performed after the May 12, 1991 (the date of the last Type A test), Type A test shall be performed no later than May 11, 2006.

3.0 <u>EVALUATION</u>

3.1 <u>Probabilistic Safety Analysis</u>

3.1.1 Assessment

The licensee’s April 11, 2002, application included a risk impact assessment of extending the Type A test interval to 15 years. The supplemental letter dated November 11, 2002, provided additional risk impact assessment analysis. The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing,” and RG 1.174, “An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, “Performance-Based Containment Leak-Test Program,” September 1995, provided the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC’s rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), the results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and BWR representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 in 10 years to 1 in 20 years leads to an “imperceptible” increase in risk on the order of 0.2 percent and a fraction of one person-rem per year.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the

Option B rulemaking in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6}/year and increases in large early release frequency (LERF) less than 10^{-7}/year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3 in 10 year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and whose methodology is consistent with previously approved submittals. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 3 in 10 year test interval to a 1 in 15 year test interval, the increase in the total integrated plant risk, in person-rem/year, is estimated to be about 0.03 percent. This increase is comparable to that estimated in NUREG-1493, in which it was concluded that a reduction in the frequency of tests from 3 in 10 years to 1 in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.23×10^{-7}/year. However, there is some likelihood that the undetected flaw in the containment liner estimated as part of the Class 3b frequency would be detected as part of the IWE visual examination of the containment surfaces (as identified in American Society of Mechanical Engineers [ASME] Boiler and Pressure Vessel Code, Section XI, Subsection IWE). The most recent visual examination of the Cook containment was performed in 2000 for both units. The next scheduled IWE containment inspection is 2003 for both units. Visual inspections are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the impact of hypothetical corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A ILRT. When possible corrosion of the containment surfaces is considered, the increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.32×10^{-7}/year. Therefore, the NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be about 0.3 percent for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.1.2 Summary

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.2 Degradation of Containment Pressure Boundary

This evaluation discusses the licensee’s actions taken to address aging degradation of the containment pressure boundary as it relates to the proposed one-time TS amendment of extending the time interval for performing the containment ILRT from the currently required 10 years to 15 years.

3.2.1 Inservice Inspection (ISI) for Primary Containment Integrity

D. C. Cook Units 1 and 2 utilize a Westinghouse pressurized-water reactor with an ice condenser-type of containment. The containment design includes a reinforced, concrete vessel with a steel liner. The lower and upper containment areas are separated by divided barriers. Each containment vessel is penetrated by access penetrations, process piping and electrical penetrations. The integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J. The overall leak-tight integrity of the primary containment is verified through ILRTs. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design-basis accident pressure. The last ILRTs of D. C. Cook Units 1 and 2 were performed in October 1992, and May 1992, respectively. The next ILRTs are scheduled during the outages in calendar year 2003. With the extension of the ILRT interval, the next overall verification will be performed no later than October 2007, and May 2007, for Units 1 and 2, respectively. The licensee provided information related to the ISI of the containment and discussed potential areas of degradation in the containment that might not be apparent in the risk assessment. In addition, in its letter dated November 11, 2002, the licensee provided responses to the NRC staff’s request for additional information to explicitly address four issues related to the containment degradation. The NRC staff’s evaluation of the licensee’s responses to the ISI related issues is discussed in the following paragraphs.

The licensee is using the 1992 Edition and the 1992 Addenda of the American Society of Mechnical Engineers (ASME) Code, Section XI, Subsections IWE, and IWL for ISI of the containments. In the April 11, 2002, application, the licensee states that the pressure-retaining capability of seals and gaskets is verified by the performance of LLRTs once each inspection period (40 months). Furthermore, the licensee states that the bellows at Cook Nuclear Plant are not part of the containment isolation barrier. Based on these statements, the NRC staff finds that the degradation of seals and gaskets of the containment access penetrations as well as other penetrations utilizing resilient seals will be adequately monitored and maintained.

During the 1998 and 1999 inspections, the licensee performed an examination of the steel liners of the D. C. Cook Units 1 and 2 containments in accordance with the requirements of the ASME Code, Section XI which has been incorporated by reference into 10 CFR 50.55a. In the April 11, 2002, application, the licensee provided a brief description of the corrosion and pitting found along the moisture barrier seals near the containment cylinder base in D. C. Cook Units 1 and 2. The licensee's evaluation indicated that the structural integrity of the containments to withstand normal operating loads and severe accident loads was not affected by the as-found condition of the liner. The licensee made modifications to the floor-liner seals to prevent further degradation of the liner.

In response to the NRC staff's question on details of the corrosion found, the licensee provided the following information:

The nominal thickness of the liner is 0.375 inches. There are 61 locations in Unit 1, where the depth of the corrosion pits exceeded 0.125 inches with the pit depth ranging from 0.141 inches, to a maximum of 0.172 inches at four locations. Similar, but less extensive corrosion, was found in Unit 2 where the depth of the corrosion pits exceeded 0.125 inches at two locations (considered by the licensee as acceptable without engineering evaluation). The deepest pit had a depth of 0.141 inches.

Approximately three years after the installation of the new redesigned moisture-barrier seals, sections of the seals were removed in both units, and the licensee performed visual examination where corrosion was previously identified. The licensee stated it found no moisture intrusion and no active corrosion. Moreover, the licensee plans to perform VT-3 visual examination of moisture-barrier seal areas during each inspection period.

On the basis of the mitigative and corrective actions taken by the licensee and the fact that the licensee is planning to monitor the seal areas for signs of moisture intrusion and potential corrosion, the NRC staff finds the identified degradation and licensee's corrective actions for the liner to be acceptable. The NRC staff in NRC Inspection Reports 50-315/99026 and 50-316/99026 also found the licensee's corrective actions acceptable.

In 1999, an inspection of the Unit 2 containment liner identified a 3/16 inch diameter through-wall hole at an elevation of 602 ft, 3 7/8 in. The licensee identified an apparent cause as an inadequate repair of a hole drilled inadvertently during plant construction. The section of the liner plate containing the hole was removed. The examination of the exposed concrete indicated that the drill bit seemed to have penetrated the concrete behind the liner. The licensee also identified a wooden handle of a wire brush in the vicinity of the hole. The licensee performed ultrasonic thickness measurements of a 6-inch radius area around the hole and identified a minimum liner thickness of 0.303 inches, except at a location ½ inch below the hole where the liner thickness was 0.187 inches. The corrective actions taken by the licensee included (1) removal of an approximately 6-inch-square liner plate, (2) removal of the wire brush to the extent possible, (3) repair of the affected concrete area, (4) replacement of the liner section, (5) vacuum-box testing of the repair, and (6) a local leak rate testing of the repair. The licensee considered the corrective actions as an adequate restoration of the affected liner to the design configuration.

On the basis of the corrective actions taken, and the fact that the area will be subjected to general visual examinations during the subsequent inspection periods, the NRC staff finds the corrective actions acceptable.

In the November 11, 2002, letter, the licensee provided a summary of the concrete examinations of the D. C. Cook containments. The examinations were performed in the fall of 2001, in accordance with the requirements of Subsection IWL (Ref. 5.10). The summary report provides a description of the concrete degradations (scaling, leaching, pop-outs, scaling) found during the examinations, and concludes that the examination did not reveal any condition that would potentially affect the structural integrity or the calculated design safety margin of the containments. A review of the summary report indicates that the licensee has a detailed procedure for examining and evaluating the containment concrete as required by ASME Code, Section XI. The NRC staff finds that the process to be implemented for concrete examinations is capable of detecting gross degradation, and adequate corrective actions will be taken when necessary to ensure the structural integrity of the containments. On this basis, the NRC staff finds the licensee’s process for performing concrete examinations to be acceptable.

Based on the above discussion, the NRC staff finds that implementation of the licensee’s containment ISI program, including the areas subjected to subsequent inspections, provides reasonable assurance that the identified degradation occurring in the accessible areas of the containments is being adequately monitored.

In response to the staff’s question on incorporating the potential degradation in uninspectable areas of the containments in the risk assessment, the licensee considered the following steps in its risk assessment:

- The likelihood of a corrosion-related liner flaw was determined.
- The likelihood of a corrosion-related liner flaw was adjusted for age.
- The change in flaw likelihood for an increase in inspection interval was determined.
- The likelihood of a breach in containment for a given liner flaw was determined.
- The likelihood of failure to detect a flaw by visual inspection was determined considering the portion of the liner that is uninspectable.
- The likelihood of non-detected containment leakage due to the increase in in test interval was determined.

The acceptance of the licensee’s risk-assessment is discussed elsewhere in this safety evaluation.

Based on its review of the information provided in the licensee's amendment request and response to the NRC staff's questions, the NRC staff finds that (1) the structural degradation of the accessible areas of the Unit 1 and 2 containments will be adequately monitored through the periodic ISI conducted as required by Subsections IWE and IWL of Section XI of the ASME Code, and (2) the integrity of the penetrations and containment isolation valves will be periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Subarticle IWE-5000 of Section XI of the ASME Code. Significant degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 or 10 CFR 50.73.

3.2.2 Summary

Based on the above evaluation, the NRC staff finds that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure-retaining components of the D. C. Cook, Units 1 and 2 containments. Thus, granting a one-time 5-year extension to the current 10-year test interval for the containment integrated leak-rate testing, as proposed by the licensee in Section 4.6.1.2 of the TS change request, is acceptable.

On the basis of findings discussed above, the NRC staff concludes that a one-time extension of performing the ILRT as proposed by the licensee in Section 3.6.1.2 of the proposed TS amendment request is acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change 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 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 (67 FR 34488). Accordingly, the amendments meet 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 amendments.

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 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.

Principal Contributor: J. Pulsipher
H. Ashar
R. Palla

Date: February 25, 2003