July 10, 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, UNIT 1 (TAC NO. MB7162)

Dear Mr. Bakken:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your application for exemption dated December 10, 2002. The proposed exemption would allow the use of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (Code) Case N-641. This Code case permits the use of an alternative reference fracture toughness for reactor vessel materials in determining the reactor pressure vessel pressure-temperature curves, to maintain operator flexibility and safety during heatup and cooldown conditions.

The assessment is being forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

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

Docket No. 50-315

Enclosure: Environmental Assessment

cc w/encl: See next page

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UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an exemption from Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G for Facility Operating License No. DPR-58, issued to Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook (D. C. Cook) Nuclear Plant, Unit 1, located in Berrien County, Michigan. Therefore, as required by 10 CFR 51.21, the NRC is issuing this environmental assessment and finding of no significant impact.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

The proposed action would exempt the licensee from the requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G, which would allow the use of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) Code Case N-641 as the basis for revised reactor vessel pressure and temperature (P-T) curves, and low temperature overpressure protection system setpoints in the D. C. Cook Unit 1, technical specifications.

The regulation, at 10 CFR Part 50, Section 50.60(a), requires, in part, that except where an exemption is granted by the Commission, all light-water nuclear power reactors must meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50. Appendix G to 10 CFR Part 50 requires that P-T limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak-rate testing conditions. Specifically, 10 CFR Part 50, Appendix G, states, "The appropriate requirements on both the P-T limits and the minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the ASME Code, Section XI, Appendix G, limits.

ASME Code Case N-641 permits the use of alternate reference fracture toughness (i.e., use of "K_{IC} fracture toughness curve" instead of "K_{IA} fracture toughness curve," where K_{IC} and K_{IA} are "Reference Stress Intensity Factors," as defined in ASME Code, Section XI, Appendices A and G, respectively) for reactor vessel materials in determining the P-T curves and low temperature overpressure protection system setpoints for effective temperature and allowable pressure. Since the K_{IC} fracture toughness curve shown in ASME Code, Section XI, Appendix A, Figure A-2200-1 (the K_{IC} fracture toughness curve), provides greater allowable fracture toughness than the corresponding K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1 (the K_{IA} fracture toughness curve), using ASME Code Case N-641 to establish the P-T curves and low temperature overpressure protection system setpoints would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G. Therefore, an exemption to apply ASME Code Case N-641 is required.

The proposed action is in accordance with the licensee's application dated December 10, 2002.

The Need for the Proposed Action:

The proposed exemption is needed to allow the licensee to implement ASME Code Case N-641 in order to revise the method used to determine the P-T curves and because low temperature overpressure protection system setpoints based on the method specified by Appendix G to 10 CFR Part 50, unnecessarily restrict the P-T operating window.

The underlying purpose of Appendix G, is to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. This is accomplished through regulations that, in part, specify fracture toughness requirements for ferritic materials of the RCPB. Pursuant to 10 CFR Part 50, Appendix G, it is required that P-T limits for the reactor coolant system (RCS) be at least as conservative as those obtained by applying the methodology of the ASME Code, Section XI, Appendix G. Current P-T limits produce operational constraints by limiting the P-T range available to the operator to heat up or cool down the plant. The operating window through which the operator heats up and cools down the RCS, becomes more restrictive with continued reactor vessel service. Reducing this operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent low temperature overpressure protection system (OPPS) actuation due to pressure surges associated with normal plant evolutions, such as reactor coolant pump start and swapping operating charging pumps with the RCS in a water-solid condition. P-T limits for an increased service period of operation of 32 effective full-power years for D. C. Cook Unit 1, based on ASME Code, Section XI, Appendix G requirements, would significantly restrict the ability to perform plant heatup and cooldown, create an unnecessary burden to plant operations, and challenge control of plant evolutions required with OPPS enabled. Continued operation of D. C. Cook Unit 1 with P-T curves developed to satisfy ASME Code, Section XI, Appendix G, requirements without the relief provided by ASME Code Case N-641, would unnecessarily restrict the P-T operating window, especially at low temperature conditions. Use of the K_{IC} curve in determining the lower bound fracture toughness of RPV steels is more technically correct than use of the K_{IA} curve, since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the conservatism of the K_{IA} curve since 1974, when the curve was adopted by the ASME Code. This conservatism was initially necessary due to the limited knowledge of the fracture toughness of RPV materials at that time. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{IA} curve greatly exceeds the margin of safety required, and that the K_{IC} curve is sufficiently conservative to protect the public health and safety from potential RPV failure. Application of ASME Code Case N-641 will provide results that are sufficiently conservative to ensure the integrity of the RCPB, while providing P-T curves and low temperature overpressure protection system setpoints that are not overly restrictive. Implementation of the proposed P-T curves and low temperature overpressure protect system setpoints, as allowed by ASME Code Case N-641, will continue to provide significant safety margin for the RCPB.

In the associated exemption, the NRC staff has determined that, pursuant to 10 CFR Part 50, Section 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served by the implementation of ASME Code Case N-641.

Environmental Impacts of the Proposed Action:

The NRC has completed its evaluation of the proposed action and concludes that there are no significant environmental impacts associated with the use of the alternative analysis method to support the revision of the RCS P-T limits.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not have a potential to affect any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Environmental Impacts of the Alternatives to the Proposed Action:

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the "no-action" alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources:

The action does not involve the use of any different resource than those previously considered in the Final Environmental Statement for the Donald C. Nuclear Plant Units 1 and 2, dated August 1973.

Agencies and Persons Consulted:

On June 6, 2003, the staff consulted with the Michigan State official, Ms. Sara De Cair of the Department of Environmental Quality, regarding the environmental impact of the proposed action. The State official had no comments.

FINDING OF NO SIGNIFICANT IMPACT

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated December 10, 2002. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209 or 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 10th day of July 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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