although Code Case N–640 was incorporated into the ASME Code recently, an exemption is still needed because the P–T limits required by 10 CFR 50.60 are based on the 1989 edition of the ASME Code.

The new P–T limits calculated by the methodologies that are subject to the exemptions are incorporated into the PNPP Technical Specifications by an associated proposed license amendment submitted by the licensee. The proposed action is in accordance with the licensee's application for exemption and amendment dated June 4, 2002.

The Need for the Proposed Action

The revised P–T limits are desired to allow required reactor vessel hydrostatic and leak tests to be performed at a significantly lower temperature. These tests are to be performed during the upcoming refueling outage scheduled to commence in April 2003. The lower temperature for the tests can reduce refueling outage critical path time by reducing or eliminating the heatup time to achieve required test conditions.

Environmental Impacts of the Proposed Action

The Commission has evaluated the proposed action and concludes that the exemption and associated license amendment described above would provide an adequate margin of safety against brittle failure of the PNPP reactor vessel. Since the proposed changes do not adversely affect the integrity of the reactor vessel, the function of the vessel to act as a radiological barrier during an accident is not affected.

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 not significant increase in occupational or public radiation exposure. Therefore, there are not significant environmental impacts associated with the proposed action.

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

Accordingly, the NRC concludes that there is not 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 resources that those previously considered in the Final Environmental Statement for the PNPP, dated April 1974.

Agencies and Persons Consulted

On March 11, 2003, the staff consulted with the Illinois State Official, Frank Niziolek of the Illinois Department of Nuclear Safety, regarding the environmental impact of the proposed action. The Staff 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 June 4, 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, 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 13th day of March, 2003.

For the Nuclear Regulatory Commission.

Anthony J. Mendiola,

Chief, Section 2, Project Directorate III–2, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 03–6543 Filed 3–18–03; 8:45 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–316]

Indiana Michigan Power Company, Donald C. Cook Nuclear Plant, Unit 2; 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–74, issued to Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook (D. C. Cook) Nuclear Plant, Unit 2, 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 2 Technical Specifications (TSs).

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 Appendix G 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 July 23, 2002, as supplemented by letters dated November 15, 2002, and January 24, 2003.

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 continued use of the method specified by Appendix G to 10 CFR part 50, unnecessarily restricts 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 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 2, based on ASME Code, section XI, Appendix G requirements, would significantly restrict the ability to perform plant heatup and cooldown, and create an unnecessary burden to plant operations, and challenge control of plant evolutions required with the Over Pressure Protection Section enabled. Continued operation of D.C. Cook Unit 2 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 protection system setpoints, as allowed by ASME Code Case N–641, does not significantly reduce the margin of safety.

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. Cook Nuclear Plant Units 1 and 2, dated August 1973.

Agencies and Persons Consulted

On February 10, 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 July 23, 2002, as supplemented by letters dated November 15, 2002, and January 24, 2003. 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 12th day of March 2003.

For the Nuclear Regulatory Commission.

L. Raghavan,

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

[FR Doc. 03–6544 Filed 3–18–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on April 9, 2003, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, April 9, 2003—3:30 p.m. until the conclusion of business

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301/415–7364) between 7:30 a.m. and 4:15 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: March 11, 2003.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 03–6547 Filed 3–18–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on April 9, 2003, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, April 9, 2003—8:30 a.m. until the conclusion of business.

The purpose of this meeting is to review the license renewal application for the St. Lucie nuclear plant and the NRC staff's initial Safety Evaluation Report. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, the Florida Power and Light Company, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Timothy Kobetz (telephone 301/415–8716) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: March 11, 2003.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 03–6548 Filed 3–18–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission (NRC) has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. Regulatory Guides are developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide is temporarily identified by its task number, DG–1107, which should be mentioned in all correspondence concerning this draft guide. Draft Regulatory Guide DG–1107, ''Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident'' is being developed to describe methods acceptable to the NRC staff for implementing requirements with respect to the sumps and suppression pools performing the functions of water sources for emergency core cooling, containment heat removal, or containment atmosphere clean up. Section 1.1.4 of DG–1107 contains discussions of active debris mitigation systems in lieu of the passive sump screens that are in many of the nuclear plants. Specifically, comments on alternative solutions to debris strainers for ensuring long-term cooling are solicited.

This draft guide has not received complete staff approval and does not represent an official NRC staff position.

Comments may be accompanied by relevant information or supporting data. Written comments may be submitted by mail to the Rules and Directives Branch,