September 19, 2007

Mr. Mano K. Nazar
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 2 (DCCNP-2) - ISSUANCE OF AMENDMENT RE: REPLACEMENT OF RESISTANCE-TEMPERATURE DETECTORS AND ASSOCIATED PIPING (TAC NOS. MD3462 AND MD3463)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. DPR-74 for Donald C. Cook Nuclear Plant, Unit 2 (DCCNP-2). The amendment consists of changes to the technical specifications (TS) in response to your application dated November 3, 2006, as supplemented on June 27, 2007. Your application covered both Donald C. Cook Nuclear Plant, Units 1 and 2, but your June 27, 2007, supplement withdrew the proposed changes affecting the Unit 1 TS; thus, the enclosed amendment pertains only to DCCNP-2.

The amendment approved elimination of the resistance temperature detector (RTD) bypass piping and installing fast response thermowell-mounted RTDs in the reactor coolant system loop piping. The amendment also revised Surveillance Requirement 3.3.1.15 of the TS, deleting the requirement to perform surveillance on the reactor coolant system RTD bypass loop flow rate.

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

Sincerely,

/RA/

Adrian Muñiz, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 280 for DCCNP-2
2. Safety Evaluation

cc w/encls: See next page

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 280
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 November 3, 2006, as supplemented by letter dated June 27, 2007, 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 Renewed Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and Appendix B, as revised through Amendment No. 280, are hereby incorporated in the renewed operating 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 prior to entry into Mode 2 during the unit's fall 2007 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/P. Tam for/

Travis L. Tate, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications
and Renewed Facility Operating License

Date of Issuance: September 19, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 280

RENEWED FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of Renewed Facility Operating License No. DPR-74 with the attached revised page. The change area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The change areas are identified by marginal lines.

REMOVE	INSERT
3.3.1-9	3.3.1-9

radiation monitoring equipment calibration, and as fission detectors in amounts as required.

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3468 megawatts thermal in accordance with the conditions specified therein and in attachment 1 to the renewed operating license. The preoperational tests, startup and other items identified in Attachment 1 to this renewed operating license shall be completed. Attachment 1 is an integral part of this renewed operating license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and Appendix B, as revised through Amendment No. 280, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

(a) Deleted by Amendment No. 76

(b) Deleted by Amendment No. 2

(c) Leak Testing of Emergency Core cooling System Valves

Indiana Michigan Power company shall prior to completion of the first inservice testing interval test each of the two valves in series in the

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 280 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NOS. 50-316

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated November 03, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML063320468), as supplemented by letter dated June 27, 2007 (ADAMS Accession No. ML071930148), Indiana Michigan Power Company (I&M, the licensee) requested an amendment to the technical specifications (TS) to Renewed Facility Operating License for Donald C. Cook Nuclear Plant, Unit 1 and Unit 2 (DCCNP-1 and DCCNP-2). The amendment would reflect a plant modification that will replace the reactor coolant system (RCS) resistance temperature detectors (RTDs) and bypass piping with fast-response RTD detectors mounted in thermowells directly in the primary loop piping of DCCNP-2. The proposed TS changes affect the applicable notes in the DCCNP-2 TS surveillance requirement (SR) for channel calibration of the overtemperature differential temperature (OTΔT) and overpower differential temperature (OPΔT) Reactor Trip System (RTS) functions. The proposed change would also affect both units TS allowable values (AV) for OTΔT and OPΔT RTS functions.

By letter dated June 27, 2007, I&M withdrew the proposed changes to the DCCNP-1 and DCCNP-2 AV for OTΔT and OPΔT RTS functions. This withdrawal eliminates TS changes proposed for DCCNP-1 and renders the TS changes proposed for DCCNP-2 identical to the TS changes approved by the NRC for DCCNP-1 due to a similar modification in DCCNP-1 approved by Amendment No. 296, dated October 6, 2006. The notice of partial withdrawal is issued by letter dated the same date as this safety evaluation. The licensee's June 27, 2007, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2007.

The existing Note 1 in DCCNP-2 RTS TS SR 3.3.1.15 requires that an OTΔT or OPΔT channel calibration include verification of RCS RTD bypass loop flow rate. Note 2 states that normalization of the ΔT is not required to be performed until 72 hours after thermal power is greater than or equal to 98 percent of rated thermal power. I&M proposed to revise DCCNP-2 TS by deleting existing Note 1 and providing necessary editorial changes.

Enclosure

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.36, "Technical specifications," provides the regulatory requirements for the content required in a licensee's TSs. As stated in 10 CFR 50.36, the TSs will include SRs to assure that the limiting conditions for operation (LCO) will be met.

10 CFR Section 20.1101 requires ALARA (acronym for "as low as is reasonably achievable") efforts, meaning making every reasonable effort to maintain exposures to radiation as far below the dose limits as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, and the economics of improvements in relation to benefits.

NUREG-0800, "Standard review Plan for the review of Safety Analyses Reports for Nuclear Power Plants," provides guidance to NRC staff reviewers in the Office of Nuclear Reactor Regulation in performing safety reviews of applications to approve standard designs and sites for nuclear power plants. NUREG-0800, Section 15.6.1, "Inadvertent Opening of a PWR [pressurized-water reactor] pressurizer pressure relief valve or a BWR [boiling water reactor] pressure relief valve," states that an accidental depressurization of the reactor coolant system (RCS) could be caused by the inadvertent opening of a pressure relief valve, which in turn could be caused by a spurious electrical signal or by an operator error.

3.0 TECHNICAL EVALUATION

3.1 Description of Change

Removal of the RTD bypass piping involves the replacement of each RTD, installed in the RTD bypass piping, with three RTDs, installed in thermowells, that are situated 120 degrees apart around the RCS pipe walls. The new RTDs will have a slightly longer response time, which will alter the time response characteristics of the OTΔT and OPΔT reactor protection logic.

The NRC staff's review, therefore, focused upon: (1) the accident analyses that rely upon reactor trip (RT) from the OTΔT and OPΔT reactor protection logic; (2) the effect of the new arrangement on signal time delay. In particular, the rod withdrawal at power and the inadvertent opening of a pressurizer safety and relief valve are two events that can produce conditions that can demand an OTΔT RT.

Analyses of these two events are used to determine the constants and lead/lag functions used in the OTΔT and OPΔT RT setpoint equations in Westinghouse-designed reactors. The OPΔT reactor protection logic is not discussed herein, since it is not credited in the accident analyses of the DCCNP-2 licensing basis. With regard to the OTΔT RT, the licensee evaluated the rod withdrawal at power event; but not the inadvertent opening of a pressurizer safety or relief valve event. The licensee noted that the analysis was not performed because the inadvertent opening of a pressurizer safety or relief valve event is not in the DCCNP-2 licensing basis. Indeed, Chapter 14 of the updated final safety analysis report (UFSAR) does not contain an analysis of the inadvertent opening of a pressurizer safety or relief valve event.

DCCNP-2 was licensed to operate in 1977, about four years after the American Nuclear Society's Standard N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized

Water Reactor Plants" was issued. This standard categorizes PWR events into four classes (Conditions I, II, III, and IV), and sets acceptance criteria for each of these classes. Although this standard is not referenced in the DCCNP-2 licensing basis, the accident analyses therein abide by the event classes and their corresponding acceptance criteria. The NRC staff's review criteria for the inadvertent opening of a pressurizer safety or relief valve event, are contained in NUREG-0800 (Standard Review Plan) Section 15.6.1.

3.2 Evaluation of Licensing Basis Change

The OTΔT RT function provides primary protection against departure from nucleate boiling (DNB) in Westinghouse PWRs. The measured ΔT, which is indicative of nuclear power, is compared to the OTΔT setpoint, which is calculated from continually updated values of measured average temperature (Tavg), pressurizer pressure, and reactor axial flux difference. A RT signal is generated when the measured ΔT exceeds the OTΔT setpoint in two or more reactor coolant loops.

The OPΔT RT function provides protection against fuel centerline melting in Westinghouse PWRs. The OPΔT RT function generates a reactor trip signal in a manner that is similar to the OTΔT RT function.

Replacing the current RTD bypass system with thermowell-mounted RTDs will affect the RCS temperature measurement response, and consequently, the OTΔT and OPΔT RT functions. The total time delay for these trip functions is assumed to be 8 seconds.

The following table (reproduced from the licensee's November 03, 2006, application) indicates that the total time delay for the OTΔT and OPΔT RT functions would continue to be eight seconds or less, after the RTD bypass system is removed.

Response Time Parameters for RCS Temperature Measurement		
Component	Existing RTD Bypass System (seconds)	New Fast-Response Thermowell RTD System (seconds)
RTD bypass piping transport and thermal lag	4	N/A
RTD response time	2	4
Electronics signal processing, reactor trip signal, trip breaker opening, and rod cluster control assembly gripper release	2	2
Total Response Time	8	Less than or equal to 8

The events that can be affected by changes in the OTΔT RT response time characteristics are Condition II events. As such, they are expected to result, at worst, in a RT with the plant being capable of returning to operation. In addition, a Condition II event must not be allowed to develop into a more serious fault, i.e., a Condition III or IV event, nor result in fuel rod failures, nor cause overpressurization of the RCS or main steam system.

The NRC staff requested an analysis, or equivalent, to provide reasonable assurance that the modified OTΔT trip will not significantly reduce a margin of safety (e.g., thermal margin) during an inadvertent opening of a pressurizer relief or safety valve, an event that could demand a RT through the OTΔT trip logic. This event causes an erosion of thermal margin, as RCS pressure decreases, until the reactor is tripped, either by a signal from the OTΔT RT function or from low pressurizer pressure. The inadvertent opening of a pressurizer safety or relief valve event was not analyzed or evaluated, since it is not in the DCCNP-2 licensing basis.

The licensee provided a description of the results of an evaluation of the inadvertent opening of a pressurizer safety or relief valve event in the licensing basis of a comparable Westinghouse, four-loop plant. The conclusion was that the expected change in minimum calculated departure from nucleate boiling ratio (DNBR) would be small enough (< 2 percent) to be accommodated by the available DNBR margin to the safety limit.

This conclusion seemed reasonable to the NRC staff, given the small changes in RTD response times that are proposed by the licensee. However, without seeing the evaluation results, the input parameters, or a description of the methods used, the NRC staff could not judge whether the licensee's response had provided reasonable assurance that the DNB safety limit would not be violated. Therefore, the NRC staff performed an evaluation of the inadvertent opening of a pressurizer relief or safety valve, based on thermal core limits of DCCNP-2, and concluded that there would be sufficient margin in the minimum calculated DNBR to deal with the new RTD response times.

This event, the inadvertent opening of a pressurizer safety or relief valve, is one of two events that are typically analyzed to determine the trip setpoints and response time characteristics of the OTΔT RT logic. The other event, the rod withdrawal at power, is the more limiting event. Since the inadvertent opening of a pressurizer safety or relief valve is not in the licensing basis of DCCNP-2, and not analyzed, the event cannot be used to determine the trip setpoints and response time characteristics of the OTΔT RT logic.

According to the licensee, the proposed response time characteristics of the OTΔT RT logic, as determined via a series of rod withdrawal at power analyses, would provide adequate protection during the inadvertent opening of a pressurizer safety or relief valve, based upon comparison of DCCNP-2 to another plant of similar design. The licensee concludes that the event's effect on DNBR margin would be too small to result in a violation of the DNBR safety limit.

The NRC staff's evaluation leads to the same conclusion. The NRC staff also notes that the licensee's approach, and their judgment with respect to the possible erosion of DNBR margin, would not be necessary if the inadvertent opening of a pressurizer safety or relief valve event were analyzed, as part of the licensing basis of DCCNP-2, to help determine the trip setpoints and response time characteristics of the OTΔT RT logic, as is done for other Westinghouse-designed reactors. In this instance, the lack of an analysis precluded the opportunity to adjust the trip setpoints and response time characteristics of the OTΔT RT logic to accommodate the requirements of the inadvertent opening of a pressurizer safety or relief valve event.

The licensee stated that the bypass piping removal plant modification and proposed amendment, when approved, will result in a reduction of approximately 40 percent (30 person-rem) of the overall radiation exposure to DCCNP-2 personnel performing work in the containment during refueling outages. The NRC staff recognizes that this reduction is a significant safety benefit.

3.3 Evaluation of TS Changes

The licensee proposed to delete Note 1 of SR 3.3.1.15, which requires surveillance of the RTD loop bypass flow rate. This change is acceptable, since the RTDs and the associated bypass loops will be replaced by thermowell-mounted RTDs, a system for which verification of RTD loop bypass flow rate would no longer be applicable. The NRC staff has evaluated in detail the technical aspects of elimination of the RTD bypass loop in Section 3.1 above.

3.4 Summary

The NRC staff has concluded that the safety-related effects of replacing the DCCNP-2 RTDs and the associated bypass loops with thermowell-mounted RTDs (i.e., slightly changing the response time characteristics of the OTΔT and OPΔT trip functions) are not significant. The NRC staff concludes that there will continue to be adequate protection after the proposed modification is implemented against the two postulated accidents that rely on the OTΔT trip function for DNB protection (i.e., rod withdrawal at power, and inadvertent opening of a pressurize safety or relief valve). There are no events in the DCCNP-2 licensing basis that rely upon the OPΔT trip for protection.

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

The amendment changes requirements with respect to installation and use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 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 (72 FR 153). 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 Commission 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: E. Eagle, NRO
S. Miranda, NRR

Date: September 19, 2007

cc:

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