October 6, 2006

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 1 (DCCNP-1) - ISSUANCE OF AMENDMENT REGARDING ELIMINATION OF THE RESISTANCE TEMPERATURE DETECTOR (RTD) BYPASS LOOP (TAC NO. MD2106)

Dear Mr. Nazar:

The Commission has issued the enclosed Amendment No. 296 to Renewed Facility Operating License No. DPR-58 for DCCNP-1. The amendment consists of changes to the Technical Specifications in response to your application dated May 31, 2006.

The amendment approved elimination of the 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 Technical Specifications, 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/

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures:
1. Amendment No. 296 to DPR-58
2. Safety Evaluation

cc w/encls: See next page

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NAME	PTam	THarris	JNakoski*	JRund#	LRaghavan for MMurphy
DATE	09/13/06	09/12/06	08/31/06	09/21/06	10/6/06

* Safety evaluation transmitted by memo of 8/31/06. Concurred on changes to the safety evaluation on 10/5/06.
# K. Winsberg reviewed revised version on 10/5/06.

OFFICIAL RECORD COPY

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 296
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 May 31, 2006, 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 Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to entry into Mode 2 from the fall 2006 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L. Raghavan/

Martin C. Murphy, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Operating License

Date of Issuance: October 6, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 296

RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of Renewed Facility Operating License No. DPR-58 with the attached revised page. The change area is identified by a marginal line.

REMOVE	INSERT
3	3

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The change area is identified by a marginal line.

REMOVE	INSERT
3.3.1-9	3.3.1-9

and 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 thee 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 3304 megawatts thermal in accordance with the conditions specified therein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and Appendix B, as revised through Amendment No. 296, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Less Than Four Loop Operation

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3.1-1 of Specification 3.3.1 of Appendix A to this renewed operating license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than found loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) Indiana Michigan Power Company shall implement and maintain, in effect, all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated December 12, 1977, July 31, 1979, January 10, 1981, February 7, 1983, November 22, 1983, December 23, 1983, March 16, 1984, August 27, 1985

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 296 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, Commission) dated May 31, 2006 (Accession No. ML061600449), Indiana Michigan Power Company (I&M, or the licensee) requested an amendment to the Operating License for Donald C. Cook Nuclear Plant, Unit 1 (DCCNP-1). The proposed amendment would allow the licensee to remove the resistance temperature detector (RTD) bypass piping and install fast response thermowell-mounted RTDs located in the reactor coolant system (RCS) loop piping. With this approval, the DCCNP-1 Technical Specifications (TS) would be modified, deleting Note 1 (regarding verification of reactor coolant system RTD bypass loop flow rate) from Surveillance Requirement (SR) 3.3.1.15.

The licensee stated that removal of the RTD bypass piping would occur during the fall 2006 refueling outage. The licensee expected that removal of the RTD bypass piping would result in a reduction of approximately 30 person-rem in radiation exposure to personnel working in containment during refueling outages. The licensee also expected that removal of the RTD bypass piping would reduce refueling outage costs and the likelihood of RCS leakage.

2.0 REGULATORY EVALUATION

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 overtemperature delta temperature (OTΔT) and overpower delta temperature (OPΔT) reactor protection logic. The NRC staff's review, therefore, focused upon the accident analyses that rely upon reactor trips from the OTΔT and OPΔT reactor protection logic.

In particular, the rod withdrawal at power and the inadvertent opening of a pressurizer safety or relief valve events can produce conditions that demand an OTΔT reactor trip. Analyses of these two events are used to determine the constants and lead/lag functions used in the OTΔT and OPΔT reactor trip setpoint equations. The OPΔT reactor protection logic is not credited in the accident analyses of the DCCNP-1 licensing basis. The NRC staff noted that I&M considered the rod withdrawal at power event, but not the inadvertent opening of a pressurizer safety or relief valve event, to show that the OTΔT reactor trip function, as modified, will continue to prevent departure from nucleate boiling (DNB).

Chapter 14 of the DCCNP-1 Update Final Safety Analysis Report (UFSAR) does not contain an analysis of the inadvertent opening of a pressurizer safety or relief valve event. DCCNP-1 was licensed to operate in 1974, about a year after the American Nuclear Society's Standard ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor [PWR] Plants" was issued. This standard categorizes PWR events into four classes, and sets acceptance criteria for each class. Although this standard is not referenced in the DCCNP-1 licensing basis, the accident analyses therein abide by the event classes and their corresponding acceptance criteria. I&M's application of May 31, 2006, references the predecessor of this standard, ANSI N18.2-1973.

The inadvertent opening of a pressurizer safety or relief valve event analysis is potentially affected by the changes that are proposed in I&M's application for amendment. This event causes an erosion of thermal margin, as RCS pressure decreases, until the reactor is tripped, either by a signal from the OTΔT reactor trip function or from low pressurizer pressure. Therefore, this event is a part of the NRC staff's safety evaluation. Using the review criteria set forth in Standard Review Plan (SRP) 15.6.1, the NRC staff performed a confirmatory evaluation and documented its results in Section 3.0 (on page 4 of this safety evaluation).

The licensee stated that the proposed amendment, when approved, will result in a reduction of approximately 30 person-rem in radiation exposure to personnel performing work in containment during refueling outages. The NRC staff recognizes that this reduction is a significant safety benefit.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of Licensing Basis Change

The OTΔT reactor trip function provides primary protection against 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 (T_{avg}), pressurizer pressure, and reactor axial flux difference. A reactor trip signal is generated when the measured ΔT exceeds the OTΔT setpoint in two or more reactor coolant loops.

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

Replacing the current RTD bypass system with thermowell-mounted RTDs will affect the RCS temperature measurement response, and consequently, the OTΔT and OPΔT reactor trip functions. The total time delay for these trip functions is assumed to be 8 seconds (Table 14.1-2 of ANS 51.1).

The following table (reproduced from the licensee's May 31, 2006, application) indicates that the total time delay for the OTΔT and OPΔT reactor trip functions would continue to be 8 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 current OTΔT and OPΔT reactor trip functions, as assumed in the accident analyses, are delayed by about 8 seconds, which consist of a 6-second lag for the RTD response, and a 2-second delay for the electronic process time needed to transmit the trip signal and release the rod cluster control assemblies (RCCAs). The table indicates that, after the RTD bypass lines are removed and the RTDs are replaced, the overall OTΔT and OPΔT reactor trip response times will continue to be 8 seconds or less.

The principal event in DCCNP-1's licensing basis that relies upon the OTΔT reactor trip function is the rod withdrawal at power. This event is analyzed over a range of initial power levels and reactivity insertion rates, and at the beginning and end of core life. The reactor trip signal is usually generated by high nuclear flux at higher reactivity insertion rates, and by OTΔT at lower reactivity insertion rates. In fact, this is one of two event analyses that are used to determine the constants and lead/lag functions used in the OTΔT reactor trip setpoint equation. The licensee evaluated the rod withdrawal at power event, in consideration of the modified time response of the OTΔT trip function, and concluded that the proposed change (i.e., removal of the RTD bypass lines) would have no significant effect on the results. Therefore, the analysis of this event, in Chapter 14 of the UFSAR, remains valid. The NRC staff reviewed the licensee's analysis and accepts that the analysis of this event in Chapter 14 of the UFSAR remain valid.

Another event that can be affected by a change in the time response characteristics of the OTΔT trip function is the inadvertent opening of a pressurizer safety or relief valve. The resulting decrease in reactor coolant system pressure could lead to DNB, if the reactor is not automatically tripped in a timely manner. The reactor trip signal can be generated by OTΔT or by low pressurizer pressure. A proper analysis of this event demonstrates defense-in-depth by considering two cases, that show that a reactor trip, derived from either signal, will prevent DNB. In practice, however, licensing basis analyses of this event typically consider just one case, in which the reactor is assumed to be tripped from either the OTΔT trip signal or from the low pressurizer pressure trip signal, whichever is generated first.

The NRC staff performed a confirmatory evaluation of the inadvertent opening of a pressurizer safety or relief valve by situating the trajectory of the transient onto the DCCNP-1 core limits plot (UFSAR Figure 14.1-1). The DCCNP-1 core limits plot depicts the locus of points at which the departure from nucleate boiling ratio (DNBR) is at its safety analysis limit (SAL), as a function of pressure, T_{avg} and ΔT. Primary coolant flow rate is assumed to be constant. Using this plot, it is possible to determine the pressure, at which the DNBR is at its SAL, as a function of T_{avg} for a constant value of ΔT. Using this function, and assuming a nominal operating point wherein the ΔT is about 65 °F, and T_{avg} is about 579 °F, then the pressure, at those conditions, would have to fall to about 1722 psia (from an initial value of 2100 psia) in order to reduce the DNBR to its SAL value. The low pressurizer pressure reactor trip signal will be generated at about 1840 psia. Subtracting about 50 psi for pressure measurement uncertainty puts the low pressurizer pressure reactor trip setpoint at about 1790 psia. Once the trip condition is reached, it would take about 2 seconds to transmit the signal to the RCCAs, and another 3 seconds for the RCCAs to drop into (and reach the bottom of) the core. Based upon analyses of the inadvertent opening of a pressurizer safety or relief valve in other Westinghouse PWRs of comparable design and pressurizer safety or relief valve capacities, the RCS depressurization rate would be estimated to not more than 10 psi per second. Therefore, in DCCNP-1, the RCS pressure would be estimated to be about 1740 psia at the time of reactor shutdown, 5 seconds after the OTΔT setpoint is reached. This is still about 18 psi higher than the pressure at which the DNBR SAL would be expected to be reached. Based upon this evaluation, the NRC staff concludes that DCCNP-1 would be adequately protected by the low pressurizer pressure reactor trip during an inadvertent opening of a pressurizer safety or relief valve event.

The core limits plot (UFSAR Figure 14.1-1) also depicts the OTΔT trip setpoint as a function of pressure, T_{avg} and ΔT. At nominal T_{avg} and ΔT conditions, the plot indicates that, during a RCS depressurization, the OTΔT setpoint would be reached at about the same time as the low pressurizer pressure reactor trip setpoint (1840 psia) is reached.

There are two factors that could cause the OTΔT reactor trip to occur before the low pressurizer pressure reactor trip. The first factor is the penalty term, in the OTΔT trip setpoint equation, for adverse axial flux difference. This could reduce the OTΔT trip setpoint, and thereby cause the OTΔT trip signal to be generated sooner. However, the adverse axial flux difference penalty is conservatively not modeled in the accident analyses. The second factor is the allowance for pressure measurement uncertainty in the low pressurizer pressure trip setpoint. This allowance, which is conservatively included in accident analysis assumptions, would cause the low pressurizer pressure setpoint to be reached later. Therefore, during an inadvertent opening of a pressurizer safety or relief valve event in DCCNP-1, the OTΔT trip signal could be expected to occur before the low pressurizer pressure trip. Since the latter trip function, from low pressurizer pressure, has been evaluated to be effective in protecting the DCCNP-1 core from DNB, then the former trip function, from the OTΔT trip setpoint equation, would also be effective.

Based upon this evaluation, the NRC staff concludes that DCCNP-1 would be adequately protected by either the low pressurizer pressure reactor trip or the OTΔT reactor trip during an inadvertent opening of a pressurizer safety or relief valve event.

The OPΔT trip function protects against high linear power density that could lead to fuel centerline melting. Fuel centerline melting could occur during the more severe, Condition III or

IV events, such as major steam line breaks. The OPΔT trip, and the high nuclear flux trip, could be demanded, for example, during a full power steamline break. In Westinghouse plants, the full power steamline break has been shown to be less limiting than the zero power steamline break. Protection for the zero power steamline break is provided primarily from the low pressurizer pressure and low steamline pressure trip functions, and by automatic steamline isolation. There are no events in the DCCNP-1 licensing basis that rely upon the OPΔT trip for protection.

3.2 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 had evaluated in detail the technical aspects of elimination of the RTD bypass loop in Section 3.1 above.

3.3 Summary of NRC staff Evaluation

The NRC staff has concluded that the safety-related effects of replacing the DCCNP-1 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 pressurizer safety or relief valve). There are no events in the DCCNP-1 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 amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components 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 (71 FR 38182). 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 Commission'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 Contributors: S. Miranda
C. Somers

Date: October 6, 2006