September 21, 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 - ISSUANCE OF AMENDMENT REGARDING REACTOR TRIP ON LOW TURBINE OIL PRESSURE (TAC NO. MD3161)

Dear Mr. Nazar:

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 15, 2006, as supplemented by letters on April 20, July 6 and July 25, 2007.

The amendment approves a plant design change that modifies the turbine control system and changes the TSs, increasing the associated allowable low control fluid oil pressure value from greater than or equal to ($\geq$) 57 pounds per square inch gauge (psig) to $\geq$750 psig.

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 Nos. 50-316

Enclosures:
1. Amendment No. 281 to DPR-74
2. Safety Evaluation

cc w/encls: See next page

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Amendment Accession Number: **ML072180639**
Package No.: ML072690337
TS: ML072690083

* Safety evaluation transmitted by memo on date shown

OFFICE	LPL3-1/G E	LPL3-1/PM	LPL3-1/LA	EICB/BC	SBPB/BC	ITSB/BC	OGC/NLO	LPL3-1/BC(A)
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DATE	9/12/07	9/13/07	9/12/07	8 /23/07*	8/22/07*	9/14/07	9/14/07	9/21/07

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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. 281
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 September 15, 2006, as supplemented by letters on April 20, July 6, and July 25, 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. 281, 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 1 after the unit's Cycle 17 (fall 2007) refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by T. McGinty 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 Renewed Operating License
and Technical Specifications

Date of Issuance: September 21, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 281

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

REMOVE	INSERT
3.3.1-13	3.3.1-13

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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and Appendix B, as revised through Amendment No. 281, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Additional Conditions</u>

(a) Deleted by Amendment No. 76

(b) Deleted by Amendment No. 2

(c) <u>Leak Testing of Emergency Core cooling System Valves</u>

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 (NRR)

AMENDMENT NO. 281 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By application to the Nuclear Regulatory Commission (NRC, the Commission) dated September 15, 2006 (Agencywide Documents Management System (ADAMS) Accession No. ML062690500), as supplemented by letters on April 20 (ADAMS Accession No. ML071220081), July 6 (ADAMS Accession No. ML071980393), and July 25, 2007 (ADAMS Accession No. ML072200252), Indiana Michigan Power Company (I&M, or the licensee) requested an amendment to the Technical Specifications (TSs) appended to Renewed Facility Operating License for Donald C. Cook Nuclear Plant, Unit 2 (DCCNP-2). Specifically, the amendment would reflect a plant design change that will modify the current turbine control system, and would change the TS, increasing the associated allowable low control fluid oil pressure from greater than or equal to (≥) 57 pounds per square inch gauge (psig) to ≥750 psig.

The licensee's April 20, July 6, and July 25, 2007, supplements 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 November 21, 2006 (71 FR 67396).

2.0 REGULATORY EVALUATION

The following regulatory requirements and guidance documents pertain to this proposed amendment:

(1) *Title 10 of the Code of Federal Regulations* (10 CFR) Part 50, Section 50.36, "Technical Specifications," requires that the TSs include limiting safety system settings. Specifically, if a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

(2) Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 4, "Environmental and dynamic effects design bases," requires, in part, that structures, systems and components (SSCs) important to safety shall be designed to accommodate the effects of and be compatible with, the environmental conditions associated with normal operation, maintenance, testing and postulated accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping and

discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power units.

(3) GDC 13, "Instrumentation and Control," requires, among other things, that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

(4) GDC 20, “Protection System Functions,” requires that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

(5) NUREG-0800, Standard Review Plan (SRP) Chapter 10.2, Turbine Generator, Section II, Acceptance Criteria, provides review guidance to meet GDC 4.

(6) Regulatory Guide (RG) 1.105, “Setpoints for Safety-Related Instrumentation,” describes a method acceptable to the NRC staff for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

(7) Regulatory Issue Summary (RIS) 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, Technical Specifications,” addressed limiting safety system settings during periodic testing and calibration of instrument channels.

3.0 TECHNICAL EVALUATION

The licensee proposed to replace the present DCCNP-2 turbine control system with a new electronically controlled programmable logic control system manufactured by Triconix. The turbine control system controls the speed of the turbine when the generator is not synchronized with the grid, and controls the output of the unit when the generator is synchronized with the grid.

The licensee proposed to revise TS 3.3.1, Table 16.a, to reflect the installation of a new turbine controller that requires the normal control fluid oil pressure to be increased from 114 psig to 1600 psig. Accordingly, the licensee requested to change the TS allowable low fluid oil pressure from $\geq$57 psig to $\geq$750 psig. The turbine control system is a non-safety-related system that provides input to the reactor protection system for the reactor trip on low turbine oil pressure. This reactor trip on low turbine oil pressure is an anticipatory trip that actuates shortly after a turbine trip to minimize the reactor pressure/temperature transient for a loss-of-load accident. In the September 15, 2006, application the licensee stated that the reactor trip on low turbine oil pressure is not credited in any safety analysis.

3.1 Instrumentation Setpoint Methodology

In response to the NRC staff’s request for additional information (RAI) dated January 11, 2007 (ADAMS Accession No. ML070110441), regarding the interface between the turbine control system and pressure switches used to sense low fluid oil pressure, the licensee, in its RAI response letter dated April 20, 2007, has confirmed that there is no interface between the two and are completely independent to each other. Based on this the NRC staff did not review the turbine control system modification.

Following a turbine trip the control fluid is drained from the piping and the pressure rapidly decreases. The decreased pressure is sensed by the three pressure switches located in the control oil header and a reactor trip is initiated when the decreased pressure is sensed by two out of three switches.

The low pressure setpoint must therefore, provide an un-ambiguous, non-spurious indication of turbine trip status. Accordingly, the licensee has selected a setpoint of ≥800 psig based on the nominal system value (1600 psig) and the value at which the main feedpump turbine stop valve would begin to close (approximately 575 - 600 psig). Based on this limiting setpoint, the licensee has chosen a TS allowable value (AV) of ≥750 psig to ensure the operation of this function. This value is based on the minimum required electrohydrolic control (EHC) pressure, the expected calibration tolerance and calibration frequency of the pressure switches, and the expected time-based drift of the pressure switches.

The licensee has defined the as-left tolerance or calibration tolerance equal to the manufacturer's reference accuracy. The pressure switches are left at or below this value after each calibration. The licensee has also defined the as-found tolerance band or the limit at which the pressure switch is determined to be out-of-tolerance as the expected deviation from the as-left condition. As-found tolerance is determined by taking the square root of the sum of the squares of drift plus sum of the maintenance and test equipment accuracy and calibration accuracy. Calibration accuracy is defined to the manufacturer's reference accuracy.

Based on the above review, the NRC staff determines that the licensee's method of calculating as-left and as-found tolerance meet the NRC staff's guidance provided in RIS 2006-17. The manufacturer's reference accuracy for these pressure switches is identified as 0.5 percent of the setpoint.

In response to the NRC staff's RAI, the licensee has stated in its letter dated April 20, 2007, that the purpose of this trip function is to provide an anticipatory reactor trip that minimizes the reactor pressure/temperature transient for a load rejection in excess of the capability of the steam dump system. The licensee also stated that the plant safety analyses did not credit the operation of this function and there is no analytical limit associated with this setpoint, which is used to protect a design or license basis limiting condition. DCCNP-2 is designed to withstand a complete loss of load and not sustain core damage or challenge the reactor coolant system pressure limitations. Core protection is provided by the pressurizer pressure, i.e., high trip function, and RCS integrity are ensured by the pressurizer safety valves. Based on this review, the NRC staff agrees that this setpoint is not related to a safety limit.

In response to the NRC staff's RAI regarding demonstrating the operability of the instrument function to meet its specified safety function in accordance with applicable design requirements and associated analyses, the licensee discussed the plant procedures used for this purpose in its July 25, 2007, letter. According to the licensee's program, these pressure switches are considered critical parameters because they are specified in the TS. According to plant procedure, any time an item in the critical parameters list is found out of tolerance, a condition report is generated. This condition report is evaluated by the licensee's engineering department using a form provided in the procedure. On the first and subsequent out-of-tolerance conditions, the assigned engineer must address potential operability or maintenance rule concerns. If two consecutive out-of-tolerance conditions occur, the procedure requires that the evaluation must include an explicit disposition which concludes that the component must be repaired or replaced, or the evaluation justifies that no action is needed. On such basis, the NRC staff finds that the licensee's procedure adequately addresses the operability of the pressure switches.

In summary, the NRC staff has reviewed the licensee's setpoint methodology and calculation, and concludes that the methodology demonstrates that the proposed setpoint, as-left and as-found tolerance value meet the NRC staff's guidance set forth in RIS 2006-17. The proposed TS change will allow the instrumentation that performs this trip function to be tested and verified to be operable within the capabilities of the pressure switches. In addition, the licensee's setpoint calibration procedures will maintain the trip setpoint within the established setting tolerance to ensure that the instrument will be capable of performing its specified safety function. Based on its review of the licensee's calculation and justification, the NRC staff finds the proposed TS change acceptable.

3.2 Turbine Control System Integrity

By letter dated April 20, 2007 (ADAMS Accession No. ML071070594), the NRC staff requested additional information under this topic. The licensee's response to the NRC staff's questions is described below:

As indicated in the licensee's application, the replacement of the turbine control system will increase the normal control oil fluid operation pressure from 114 psig to 1600 psig, and will revise the allowable low fluid oil pressure value for turbine trip from $\geq$57 psig to $\geq$750 psig. Because of the large increase in piping pressure, the NRC staff asked the licensee to explain: (1) to what extent this change involves any associated piping design changes that are necessary to accommodate the increased pressure, and (2) whether or not pipe failures associated with this change need to be considered due to the large increase in turbine oil operating pressure, and if so, (3) to what extent any safety-related equipment needs to be protected from the effects of these postulated pipe failures.

In its letter dated July 6, 2007, the licensee stated that the design change will replace all associated hydraulic piping. The new design will use schedule 80 stainless steel piping, which is capable of withstanding the 1600 psig system pressure, for supply piping and high-pressure hose connections to the hydraulic actuators. There is no safety-related equipment that would be adversely impacted by a postulated break in the electro-hydraulic piping. A pipe break in the hydraulic system would result in a hydraulic fluid spill in the turbine building and would trip the main turbine. However, the break would have little impact on the remainder of the facility. The pumping rate of the supply pumps is less than 50 gallons per minute with an open-ended pipe at zero psig. The system operating temperature (approximately 120 °F) precludes any impacts due to thermal expansion or fluid vaporization. Therefore, no pipe failure impacts are postulated associated with the design changes.

The NRC staff's review found the licensee's response acceptable, on the basis that all the required components are replaced with new piping to meet the manufacturer's specification, and failure of the components will not affect any safety-related equipment. Therefore, the design change meets the requirements of GDC 4.

There was insufficient information in the original application regarding the replacement of the existing mechanical-hydraulic control system. In the RAI dated April 20, 2007, the NRC staff asked the licensee to describe: (1) the system design and how it operates, how oil pressure varies with time during normal plant operation and following a turbine trip; and (2) the basis for how the 750 psig trip set point was established to assure that a spurious trip will not occur and safety functions will not be challenged, including uncertainty considerations.

In its letter dated July 6, 2007, the licensee stated that the new system will control the operation of the main feed pump turbines and the main turbine, all of which are located in the turbine building. The hydraulic system, which provides motive force for the main feed pump turbine and main turbine valve actuators and trip blocks, contains three pressure switches that provide input to the reactor protection system. The circuitry associated with the pressure switches and the reactor protection system is independent of the control system. The EHC is pressurized by a pump to a nominal steady pressure of 1600 psig. There is no significant pressure transients associated with the EHC operation. Following a turbine trip, EHC header drain valves open, and the non-compressible hydraulic oil drains from the header, resulting in the control oil pressure dropping below the proposed allowable value in milliseconds.

The licensee also stated that the safety analyses do not credit the operation of this function for core protection. Therefore, there is no associated analytical limit for the low fluid pressure allowable value. The proposed TS allowable value of ≥750 psig, represents a nominal value (approximately one-half of the nominal operating pressure) that ensures the operation of this anticipatory reactor trip function. The chosen TS allowable value includes consideration of the minimum required EHC pressure, the expected calibration tolerance and calibration frequency of the pressure switches, and the expected time-based (drift) pressure switch set-point changes.

The NRC staff's review found the licensee's response acceptable because the proposed TS allowable value has been selected to ensure the operation of the anticipatory reactor trip function, and there is no impact on design-basis accident analysis.

3.3 Summary of NRC Staff Evaluation

This proposed change will allow the instrumentation that performs this trip function to be tested and verified to be operable within the capabilities of the pressure switches. In addition, the licensee's setpoint calibration procedures will maintain the trip setpoint within the established setting tolerance to ensure that the instrument will be capable of performing its specified safety function. Based on its review of the licensee's calculation and justification, the NRC staff finds the proposed TS change acceptable.

Based on the above review, the NRC staff found that the turbine trip function initiates an anticipatory reactor trip. However, the licensee does not credit the anticipatory reactor trip for protection of fission product barriers. Therefore, the proposed amendment has insignificant impact on the plant's safety analyses. Furthermore, the turbine control system is a non-safety-related system and there are no safety-related systems or components nearby. As such, failure of any piping components of the turbine control system will not preclude operation of the safety systems and components.

On the basis of this review, the NRC staff concludes that the proposed amendment meets all the regulatory requirements and guidance set forth in Section 2, and 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

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 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 published November 21, 2006 (71 FR 67396). 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: J. Guo, NRR
H. Garg, NRR

Date: September 21, 2007

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 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

Kimberly Harshaw, 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

Susan D. Simpson
Regulatory Affairs Manager
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michigan Department of Environmental
Quality
Waste and Hazardous Materials Div.
Hazardous Waste & Radiological
Protection Section
Nuclear Facilities Unit
Constitution Hall, Lower-Level North
525 West Allegan Street
P. O. Box 30241
Lansing, MI 48909-7741

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

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