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SUBJECT: Forwards NIS-1 Rept for Inservice Insp activities performed at facility. See Reports

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Cook Nuclear Plant
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Cook Nuclear Plant Unit 1
Docket No. 50-316
License No. DPR-74
ISI NIS-1 Report

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Attn: J. B. Martin

August 24, 1994

Dear Mr. Martin:

Attached please find a copy of the NIS-1 Report for Inservice Inspection activities performed at the Cook Nuclear Power Plant, Unit 1, State of Michigan, Number M-09672-M, located in Bridgman, Michigan. The Plant is owned by Indiana Michigan Power Company, One Summit Square, Ft. Wayne, Indiana 46802.

The unit's commercial service date is August 23, 1975 and has a gross generating capacity of 1080 MWe.

The Authorized Code Inspector is Mr. M. K. Muterspaugh from Factory Mutual Engineering, whose address is 30150 Telegraph Road, Bingham Office Park Suite 141, Bingham Farms, Michigan 48025.

The examinations, tests, replacements and repairs performed, conditions observed, and corrective measures taken are summarized in the attached report. ISI activities were performed in accordance with the rules and requirements of ASME Code Section XI 1983 Edition, Summer of 1983 Addenda.

Respectively,

L. S. Gibson
Assistant Plant Manager
D. C. Cook Nuclear Plant

cbm

Attachment

c: See attached distribution.

A047

c: E. E. Fitzpatrick (w/o attachment)
G. Charnoff (w/o attachment)
J. R. Padgett - MI Public Serv. Commission (w/o attachment)
D. R. Hahn - NFEM Section Chief (w/o attachment)
W. T. Russell - NRR Director (w/o attachment)
J. A. Isom - D. C. Cook Resident Inspector (w/o attachment)
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DC. COOK UNITS 1 & 2 Confirmatory Order Modifying Post-TMI Requirements Pertainin g to Containment Hydrogen Monitors

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

OP-3

February 4, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 RE: CONFIRMATORY ORDER MODIFYING POST-TMI REQUIREMENTS PERTAINING TO CONTAINMENT HYDROGEN MONITORS (TAC NOS. MA7761 AND MA7762)

Dear Mr. Powers:

The Commission has issued the enclosed Order that modifies the current requirement for establishing a continuous indication of hydrogen concentration in the Donald C. Cook Nuclear Plant, Units 1 and 2, containments following severe accidents. This requirement was contained in Attachment 6 to Item II.F.1 in NUREG-0737, "Clarification of TMI Action Plan Requirements," which was imposed by Confirmatory Order dated March 14, 1983. By letter dated December 22, 1999, Indiana Michigan Power Company requested relief from the requirement to have monitoring of containment hydrogen concentration available within 30 minutes following initiation of safety injection, using risk insights as the basis for this request. The request is part of an initiative undertaken by the Commission and the Nuclear Energy Institute to incorporate risk-informed and performance-based insights into the regulation of nuclear power plants.

The enclosed Order has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

ML003680431
Template = NRR-080

Enclosure: Order

cc w/encl: See next page

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February 4, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
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500 Circle Drive
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/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Order

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 4, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 RE: CONFIRMATORY ORDER MODIFYING POST-TMI REQUIREMENTS PERTAINING TO CONTAINMENT HYDROGEN MONITORS (TAC NOS. MA7761 AND MA7762)

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Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Order

cc w/encl: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

7590-01P

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of	)	Docket Nos. 50-315 and 50-316
	)	
Indiana Michigan Power Company	)	
	)	
(Donald C. Cook Nuclear Plant, Units 1 and 2)	))	License Nos. DPR-58 DPR-74

CONFIRMATORY ORDER MODIFYING POST-THREE MILE ISLAND

REQUIREMENTS PERTAINING TO CONTAINMENT HYDROGEN MONITORS

I.

Indiana Michigan Power Company (IM or the licensee) is the holder of Facility Operating License Nos. DPR-58, and DPR-74 issued by the Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR Part 50. The licenses authorize the operation of Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, located in Berrien County, Michigan.

II.

As a result of the accident at Three Mile Island, Unit 2 (TMI-2), the NRC issued NUREG-0737, "Clarification of TMI Action Plan Requirements," in November 1980. Generic Letters 82-05 and 82-10, issued on March 17 and May 5, 1982, respectively, requested licensees of operating power reactors to furnish information pertaining to their implementation of specific TMI Action Plan items described in NUREG-0737. Orders were issued to licensees confirming their commitments made in response to the generic letters. The Confirmatory Order that was issued to IM on March 14, 1983, required the licensee to implement and maintain the

various TMI Action Plan Items, including Item II.F.1, Attachment 6 pertaining to monitoring of the hydrogen concentration in the containment following a safety injection.

Significant improvements have been achieved since the TMI accident in the areas of understanding risks associated with nuclear plant operations and developing better strategies for managing the response to potential severe accidents at nuclear power plants. Recent insights pertaining to plant risks and severe accident assessment tools have led the NRC staff to conclude that some TMI Action Plan items can be revised without reducing, and perhaps enhancing, the ability of licensees to respond to severe accidents. The NRC's efforts to understand the risks associated with commercial nuclear power plant operations more effectively and to reduce unnecessary regulatory burden on licensees and the public have prompted the NRC's decision to revise the post-TMI requirement to monitor containment hydrogen concentration.

The Confirmatory Order of March 14, 1983, imposed requirements upon the licensee to have continuous monitoring of containment hydrogen concentration provided in the control room, as described by TMI Action Plan Item II.F.1, Attachment 6. Information about hydrogen concentration supports the licensee's assessments of the degree of core damage and whether a threat to the integrity of the containment may be posed by hydrogen gas combustion. TMI Action Item II.F.1, Attachment 6, states:

- If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

This requirement to have monitoring of the hydrogen concentration in the containment within 30 minutes following the start of safety injection has defined both design and operating characteristics for hydrogen monitoring systems at nuclear power plants since the implementation of NUREG-0737. In addition, the technical specifications of most nuclear power

plants and NRC regulation 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," require availability of hydrogen monitors.

By letter dated December 22, 1999, IM used the Oconee and Arkansas Nuclear One confirmatory orders of November 29, 1999, and September 28, 1998, respectively, as guidance to request relief for the two CNP units from the requirement to have indication of hydrogen concentration in the containment within 30 minutes of the initiation of safety injection. Specifically, the licensee requested that risk-informed insights be used to determine the functional requirements for monitoring of containment hydrogen concentration that would allow extending the monitoring requirement to more than 30 minutes following initiation of safety injection. The basis for this request was that the additional time would allow the operators to complete their initial accident assessment and mitigation duties before redirecting their attention to the relatively longer-term recovery actions, such as actuating the hydrogen recombiners, that are not needed for at least 24 hours.

Based on the staff's evaluation of the justification provided by the licensee, and improved understanding of insights pertaining to plant risks, severe accident assessment, and emergency planning since the TMI-2 accident, the staff has concluded that the licensee's request should be approved. Giving the licensee the flexibility and responsibility for determining the appropriate time limit for establishing monitoring of containment hydrogen concentration will preclude control room personnel from being distracted from various important tasks in the early phases of accident mitigation, while allowing cognizant personnel, mostly outside the control room, to be aware of hydrogen concentration based on a risk-informed functional assessment at a reasonable time following an accident. Because the appropriate balance between control room activities and longer-term management of the response to severe accidents can best be determined by the licensee, the NRC staff has determined that the licensee may elect to adopt

a risk-informed functional requirement in lieu of the current 30-minute time limit for establishing monitoring of the hydrogen concentration as imposed by the Order dated March 14, 1983, and as described by TMI Action Item II.F.1, Attachment 6, in NUREG-0737. The appropriate functional requirement is as follows:.

> Procedures shall be established for ensuring that monitoring of hydrogen concentration in the containment atmosphere is available in a sufficiently timely manner to support the implementation of the Donald C. Cook Nuclear Plant Emergency Plan (and related procedures) and related activities such as guidance for severe accident management. Hydrogen monitoring will be initiated based on: 1) the appropriate priority for establishing monitoring of hydrogen concentration within the containment in relation to other activities in the control room, 2) the use of the monitoring of hydrogen concentration by decision makers for severe accident management and emergency response, and 3) insights from experience or evaluation pertaining to possible scenarios that result in significant generation of hydrogen that would be indicative of core damage or a potential threat to the integrity of the containment building. Affected licensing basis documents and other related documents will be appropriately revised and/or updated in accordance with applicable NRC regulations.

The licensee's Post Accident Monitoring Instrumentation Technical Specifications and 10 CFR 50.44 require the licensee to maintain the ability to monitor hydrogen concentration in the containment. However, the details pertaining to design and manner of operation of the hydrogen monitoring system are determined by the licensee.

III.

Accordingly, pursuant to Sections 103, 104b, 161b, 161i, 161o, 182, and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202 and 10 CFR Part 50, IT IS HEREBY ORDERED that:

> NRC License Nos. DPR-58 and DPR-74 are modified as follows:
>
> The licensee may elect to either maintain the 30-minute time limit for monitoring of hydrogen in the containment, as described by TMI Action Plan Item II.F.1, Attachment 6, in NUREG-0737 and required by the Confirmatory Order of March 14, 1983, or modify the time limit in the manner specified in Section II of this Order.

The Director, Office of Nuclear Reactor Regulation, may, in writing, relax or rescind any of the above conditions upon demonstration by the licensee of good cause.

IV.

Any person adversely affected by this Confirmatory Order, other than the licensee, may request a hearing within 20 days of its issuance. Where good cause is shown, consideration will be given to extend the time to request a hearing. A request for extension of time must be made in writing to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and include a statement of good cause for the extension. Any request for a hearing shall be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, ATTN: Chief, Rulemakings and Adjudications Staff, Washington, DC 20555-0001. Copies of the hearing request shall also be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, to the Deputy Assistant General Counsel for Hearings and Enforcement at the same address, to the Regional Administrator, NRC Region III, 801 Warrenville Road, Lisle, IL 60532-4351, and to David W. Jenkins, Esquire, Indiana Michigan Power Company, Nuclear Generation Group, One Cook Place, Bridgman, MI 49106, attorney for the licensee. If such a person requests a hearing, that person will set forth with particularity the manner in which his interest is adversely affected by this Order and will address the criteria set forth in 10 CFR 2.714(d).

If the hearing is requested by a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing will be whether this Confirmatory Order should be sustained.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above will be final 20 days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV will be final when the extension expires if a hearing request has not been received.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 4th day of February 2000.

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FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 4th day of February 2000

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50-315/316

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 7, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2, RE: MOVEMENT OF STEAM GENERATOR SECTIONS IN THE AUXILIARY BUILDING FOR STEAM GENERATOR REPLACEMENT PROJECT (TAC NOS. MA6665 AND MA6666)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 233 to Facility Operating License No. DPR-58 and Amendment No. 216 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated September 23, 1999, as supplemented October 11, 1999. Following initial review of your application, the staff issued a request for additional information on October 26, 1999, which you responded to by letter dated November 10, 1999.

The amendments provide approval to move steam generator sections through the auxiliary building and to disengage crane travel interlocks, and provide relief from performance of Technical Specification Surveillance Requirement 4.9.7.1. The loads to be moved are in support of the Unit 1 Steam Generator Replacement Project. Since the Unit 1 steam generator sections are heavier than those evaluated in the Updated Final Safety Analysis Report for the auxiliary building crane over the planned load path, your staff concluded that the proposed activity may increase the probability of occurrence or the consequences of an accident and requested prior NRC approval in accordance with 10 CFR 50.59.

DF01

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 233 to DPR-58
2. Amendment No. 216 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

Original signed by:
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 233 to DPR-58
2. Amendment No. 216 to DPR-7
3. Safety Evaluation

cc w/encls: See next page

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Sincerely,

Original signed by:
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 233 to DPR-58
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3. Safety Evaluation

cc w/encls: See next page

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cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 233
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 September 23, 1999, as supplemented October 11 and November 10, 1999, 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.

ML993500441

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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 233 , are hereby incorporated in the 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 within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 7, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 233

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 9-8	3/4 9-8

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1. Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

This Surveillance Requirement is not required during the movement of steam generator sections in the auxiliary building for the Unit 1 steam generator replacement project. When crane travel interlocks are disengaged, administrative controls shall be in place to prevent loads from passing over the spent fuel pool.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be $\leq$ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

* Shared system with Cook Nuclear Plant - Unit 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
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 23, 1999, as supplemented October 11 and November 10, 1999, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 216 , are hereby incorporated in the 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 within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 7, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 9-7	3/4 9-7

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

This Surveillance Requirement is not required during the movement of steam generator sections in the auxiliary building for the Unit 1 steam generator replacement project. When crane travel interlocks are disengaged, administrative controls shall be in place to prevent loads from passing over the spent fuel pool.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be $\leq$ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

* Shared system with Cook Nuclear Plant - Unit 1.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated September 23, 1999, as supplemented October 11 and November 10, 1999, the Indiana Michigan Power Company (IM, or the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The proposed amendments would provide approval to move steam generator sections through the auxiliary building and to disengage crane travel interlocks, and provide relief from performance of Technical Specification Surveillance Requirement (TSSR) 4.9.7.1. The loads to be moved are in support of the Unit 1 Steam Generator Replacement Project (SGRP). Since the Unit 1 steam generator sections are heavier than those evaluated in the Updated Final Safety Analysis Report (UFSAR) for the auxiliary building crane over the planned load path, the licensee concluded that the proposed activity may increase the probability of occurrence or the consequences of an accident and requested prior NRC approval in accordance with 10 CFR 50.59.

The October 11, 1999, submittal provided corrected TS pages. The November 10, 1999, submittal was in response to a NRC request for additional information dated October 26, 1999, and provided clarifying information to the original submittal. This information was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

2.0 EVALUATION

The Unit 1 original Westinghouse Model 51 steam generators will be replaced with Babcock and Wilcox (B&W) Model 51R steam generators due to the degrading condition of the original steam generator tubes. The steam generator (SG) replacement will involve partial disassembly of the reinforced concrete enclosures surrounding each SG and implementation of a two-piece replacement methodology. This approach includes cutting the SGs into an upper section (steam dome) and lower section (SG lower assembly). Both sections will be removed from containment. The steam domes will be refurbished and returned along with the replacement lower sections supplied by B&W.

The SG sections will be moved between the containment equipment hatch and crane bay using the auxiliary building cranes. On March 8, 1988, the NRC approved TS Amendment No. 100 to Facility Operating License No. DPR-74 for the CNP Unit 2 SGRP. In order to facilitate the Unit 2 SGRP, the licensee modified its existing auxiliary building crane to meet the single-failure-proof criteria of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The licensee performed approved modifications to the auxiliary building to support the modifications to the auxiliary building crane. The licensee also procured an additional single-failure-proof crane for the auxiliary building to be used in tandem with the existing crane in order to support the steam generator section moves through the auxiliary building. This approach was acceptable to the NRC. The load size and weight, handling equipment and methods, and load paths for movement of the Unit 1 SG sections through the auxiliary building are similar to those approved by the NRC for the Unit 2 SGRP in Amendment No. 100. The Unit 2 SGRP moved loads up to approximately 277 tons using the tandem crane arrangement. However, since that approval was only applicable to the Unit 2 SGRP, the licensee made this request for Unit 1. The licensee proposes to move loads up to approximately 270 tons using the tandem crane arrangement for the Unit 1 SGRP. Because the Unit 1 SG sections are heavier than those evaluated for a seismic event in the UFSAR and the proposed activity may increase the probability of occurrence or the consequences of an accident, NRC approval of the proposed load handling is required in accordance with 10 CFR 50.59.

The licensee proposes to (1) perform load handling for 16 SG sections that are heavier than the loads previously evaluated for the proposed load path for CNP's heavy loads program and (2) disengage the crane travel interlocks of TSSR 4.9.7.1 to accommodate movement of the cranes at the southwest corner of the spent fuel pool (SFP). The TS requirements are the same for Unit 1 and Unit 2, and the cranes and SFP are common to both units.

2.1 <u>HANDLING OF HEAVY LOADS CONSISTENT WITH NUREG-0612</u>

In Generic Letter 85-11, "Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612," the NRC concluded that satisfying the NUREG-0612 Phase I guidelines assures that the potential for a load drop is extremely small. The handling of heavy loads at CNP is consistent with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Plants," as concluded by the NRC and documented in a safety evaluation report dated September 20, 1983. The licensee evaluated the proposed handling of the Unit 1 SG sections with respect to the seven guidelines of NUREG-0612 as provided below.

2.1.1 <u>Safe Load Path</u>

Primary reliance for safe load handling during the proposed activity is placed on the use of single-failure-proof cranes; however, the licensee performed a review and walkdown of the load path though the auxiliary building to identify potential interactions with equipment important to safety.

The load drop evaluation included conservative assumptions. In the unlikely event of a drop, the SG section is assumed to penetrate all intervening structures, systems, and components and stop at the building foundation. All components beneath the entire load path are assumed to lose functional capability, regardless of where along the path the drop occurs. For example, when evaluating affected equipment or components in the west end of the load path, no credit is taken for equipment in the east end of the load path to mitigate the event. This is a conservative assumption since a dropped steam generator component is not large enough to impact all equipment below the load path simultaneously.

Postulated load drops were not considered directly over the SFP, because the SG sections do not travel directly over the SFP and the center of gravity of the SG sections is maintained outside of the exclusion area at all times. The load path for Unit 1 SG sections differs from the load path used for the Unit 2 SGRP in 1988. Because of the Unit 1 containment equipment hatch and radiation shield wall, movement of each Unit 2 SG lower assembly required one end of the section to be moved over the southwest corner of the SFP. This is not true for movement of the Unit 1 SG sections. The licensee's evaluation concluded that the integrity of the SFP would be maintained. The conclusion was based on the geometry of the SFP and adjacent building structure elements, as well as the relative member sizes of the SFP wall and connecting building structure elements. Although cracking and localized damage to the 5-feet 2-inch thick reinforced concrete pool wall would occur, the steel liner would remain intact due to its ductility.

If a load drop of a SG section is postulated that results in damage to the SFP cooling piping external to the SFP, a potential loss of SFP cooling could occur. Assuming the maximum design basis heat load in the SFP without an initial loss of water inventory, the design basis analysis for compete loss of SFP cooling demonstrates that bulk boiling would not occur for 5.74 hours. The decay heat of fuel assemblies currently in the SFP is relatively low compared to the maximum values assumed in the design basis analysis. The licensee has calculated that the current time for bulk boiling in the SFP is greater than 31.6 hours from a starting point of 116°F and greater than 22.1 hours from a starting point of 144°F. Therefore, significant time would exist for operator actions to either restore normal SFP cooling or provide alternate cooling methods for a complete loss of SFP cooling. The SFP is designed and maintained to prevent inadvertent draining of the pool if the external SFP cooling piping were damaged. Therefore, adequate SFP water inventory would be ensured even in the event of damage to the external SFP cooling water piping. Mitigation of a loss of SFP cooling event is governed by approved plant operating procedures, which include listing acceptable sources of makeup water to the SFP.

Since the SG sections do not travel directly over the SFP and the center of gravity of the SG sections is maintained outside of the exclusion area at all times, the licensee concluded that potential damage to new or spent fuel assemblies in the SFP cannot result from a direct drop of a section. The licensee also considered the possibility for a dropped component to roll into the SFP. The qualitative evaluation divided the load path into five representative positions for the steam generator sections and considered three orientations for dropping of each section, each end falling first and a horizontal drop. The postulated load drops would result in significant damage to the runway beams or concrete floor in the load path and possible damage to the SG section itself; however, the damage and appurtenances on the SG sections would resist rolling of the loads toward the SFP. The licensee concluded that there was reasonable assurance that movement of the SG sections could be performed without the loads traveling into the SFP.

Although equipment important to safety could be affected, the licensee concluded that the operating unit's safe shutdown and reactor decay heat removal requirements continue to be satisfied. Potential damage to Unit 1 safety-related systems and portions of the common safety-related systems affecting Unit 1 would have no safety significance for the Unit 1 reactor and supporting systems, since removal and replacement of SG components in Unit 1 can only be performed while the Unit 1 reactor is defueled. The licensee evaluation concluded that, with the exception of component cooling water (CCW), potentially affected Unit 2 safety-related systems and portions of common safety-related systems affecting Unit 2 are not needed in order to safely shut down the Unit 2 reactor in the event of a load drop. There is no possibility of a load drop directly affecting reactor coolant system pressure boundary piping or creating an accident for the operating Unit 2 reactor. The most critical Unit 2 system that could be affected by a load drop in the auxiliary building is CCW. Unit 2 CCW equipment potentially damaged by a load drop includes the supply and return lines to the associated SFP heat exchangers. These lines are part of the miscellaneous CCW header supplied from one of the separate trains of CCW. Damage to this piping could result in temporary loss of both trains of Unit 2 CCW until the ruptured pipe on the miscellaneous CCW header could be isolated. The isolation valves for this piping are located outside the load path. Once the ruptured piping on the miscellaneous CCW header is isolated, complete support of residual heat removal would be restored. Existing plant procedures already provide for isolation of the affected sections of CCW piping. Therefore, this potential failure would not prevent safe shutdown and adequate decay heat removal for Unit 2.

The licensee evaluated the potential release of radioactive materials from sources other than the SFP. The evaluation determined that the only potential release of liquid radioactive material from load drop is a rupture of the radioactive waste holding tanks in the auxiliary building. As documented in UFSAR chapter 14.2.2, any spillage of fluid due to a tank rupture would drain to the sump tank or waste holdup tanks or would accumulate in the sump areas. Prior to release to the environment, sampling is required to ensure that discharge is within licensed limits. The postulated load drop could rupture waste gas vent lines to the suction header of the waste gas compressors. This would result in the release of a small amount of radioactive gas, but would not result in the release of any of the contents of the waste gas decay tanks. A rupture of one waste gas decay tank has been evaluated in UFSAR chapter 14.2.3. Therefore, the licensee concluded that the consequences of the current design basis waste decay tank rupture bound the postulated load drop.

As stated in their November 10, 1999, submittal, the licensee intends to implement the following measures prior to performing the SG load handling in the auxiliary building in order to mitigate the consequences of a potential load drop.

- An operator briefing for response to a load drop will be completed prior to movement of SG sections in the auxiliary building. This briefing will highlight the equipment potentially impacted by dropping of a SG section and the applicable response as defined in existing procedures.

- No movement of fuel assemblies will be allowed in either Unit 1 or Unit 2 containment buildings, or in the auxiliary building.

- The SFP will be isolated from Unit 2 containment.

- The weir gate between the SFP and the fuel transfer canal will be closed and pressurized.

- The SFP area exhaust fans will be required to be operable.

- Any additional items that result from compliance with the approved plant procedure for conducting infrequently performed evolutions will be incorporated into the specific heavy load procedure that governs the movement of SG sections in the auxiliary building.

Based on the above evaluation, the staff concludes that safe load paths have been implemented in a manner consistent with NUREG-0612 and are acceptable.

2.1.2 Load Handling Procedures

The licensee intends to provide load handling procedures that are specific to the upper and lower SG sections. Load paths will be defined within the load handling areas and qualified personnel will direct the crane operator to ensure conformance to the prescribed load path. The procedures will also address equipment identification, inspection and acceptance criteria, step-by-step load handling sequences, and special precautions.

The staff concludes that load handling procedures will be implemented in a manner consistent with NUREG-0612 and are acceptable.

2.1.3 Operator Training

The licensee stated that crane operators are trained and qualified using maintenance skills training lesson plans that include the requirements of American National Standards Institute (ANSI) B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)", Chapter 2-3, "Qualifications for Operators." Station-qualified crane operators used for the SG section lifts will receive both classroom and hands-on training based on these lesson plans. Training will include orientation with the specific procedures to be used for the SG section lifts prior to beginning the corresponding crane operations.

The staff concludes that crane operator training will be implemented in a manner consistent with NUREG-0612 and is acceptable.

2.1.4 Special Lifting Devices

The licensee stated that its heavy loads program includes the use of special lifting devices and requires design, fabrication, and testing that provide load handling reliability consistent with ANSI N14.6-1978, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." The special lifting devices used for the Unit 2 SGRP will be used for Unit 1. Prior to use, the devices will be inspected to verify there has been no significant corrosion or structural distress and tested in accordance with ANSI N14.6-1978.

The staff concludes that design and testing of special lifting devices will be implemented in a manner consistent with NUREG-0612 and is acceptable.

2.1.5 Lifting Devices (Not Specially Designed)

The licensee stated that standard lifting devices used for the movement of the SG sections will be selected and used in accordance with the guidelines of ANSI B30.9-1996, "Slings." Due to the relatively slow hoist speeds of the cranes at CNP subject to NUREG-0612, the NRC

previously concluded in its September 20, 1983, safety evaluation report that the dynamic loads imposed on these slings are reasonably small and may be disregarded when determining the static load to be used when selecting and using slings.

The staff concludes that installation and use of slings is consistent with NUREG-0612 and is acceptable.

2.1.6 Cranes (Inspection, Testing, and Maintenance)

The licensee stated that the auxiliary building cranes at CNP are inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976. In preparation for the Unit 1 SGRP, the cranes will be inspected to confirm consistency with the single-failure-proof guidelines of NUREG-0612 and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

The staff concludes that crane component inspection and testing is consistent with NUREG-0612 and is acceptable.

2.1.7 Crane Design

The licensee stated that both auxiliary building cranes were determined to be acceptable for ANSI B30.2-1976 and Crane Manufacturers Association of America (CMAA)-70-1975, "Specifications for Electric and Overhead Traveling Cranes."

The staff concludes that crane design is consistent with NUREG-0612 and is acceptable.

2.2 CONSISTENCY WITH NUREG-0554

The licensee confirmed that the auxiliary building cranes are designed to meet the single-failure-proof criteria of NUREG-0554. The west auxiliary building crane has a design rated load (DRL) capability of 150 tons and a maximum critical load (MCL) capability of 55 tons. The east auxiliary building crane has a DRL of 150 tons and MCL of 60 tons. The two auxiliary building cranes were modified for tandem configuration as a lift system to handle the eight lifts of the Unit 2 SGRP. The tandem configuration retains the single-failure-proof features of the individual cranes and provides a DRL of 300 tons. The NRC previously found that the licensee has demonstrated that the cranes meet the single-failure-proof criteria of NUREG-0554, as documented in the safety evaluation report for Amendment No. 100 for Unit 2.

2.3 SEISMIC CONSIDERATIONS

The SG sections are considered critical loads per NUREG-0554 because they are brought through the auxiliary building in the vicinity of the SFP. For these loads, the design basis seismic capability of the load handling equipment and structures, as analyzed in the UFSAR, is exceeded. Notwithstanding the low probability of an SSE (safe shutdown earthquake) during the movement of the SG sections, the licensee studied the design adequacy of the auxiliary building cranes and structure for the tandem crane 300-ton design rated load to demonstrate that the load is safely retained even in the event of an SSE.

The engineering study follows design basis methodology except in two aspects. First, the pendulum effect of the lifted load was incorporated into the analysis, thereby determining realistic effects of the seismic accelerations of the load on the crane and supporting structure. Second, seismic vertical response spectra were generated instead of the design basis assumption of 2/3 of the horizontal spectra at the crane rail elevation. A mass and vertical stiffness mathematical model was developed that follows the criteria and methodology described in the UFSAR for the development of the design basis seismic horizontal response spectra. The design basis includes conservatism because it ignores vertical soil-structure interaction and the significantly stiffer, nearly seismically rigid, vertical dynamic response of the auxiliary building. Using calculated seismic vertical response spectra produces a large reduction in the maximum spectral amplitude compared to the design basis. Using the more realistically calculated seismic vertical response spectra, the licensee engineering study determined that the resultant wheel loads associated with the tandem crane, a 300 ton load, and an SSE are bounded by those previously evaluated in the existing CNP design basis. The NRC previously reviewed the two cranes and the auxiliary building structure for the Unit 2 SGRP (Unit 2 Amendment No. 100) and concluded that the resulting stresses are below the allowable values for the conditions imposed by two 150-ton single-failure-proof cranes lifting the steam generator lower assembly (the heaviest component) and by the combined 115-ton load during the SSE. As part of CNP's larger effort to improve design basis documentation, the engineering study is being reviewed. It is not expected that the review will impact the study's conclusions, but the licensee intends to complete the study prior to moving SG sections.

A telephone call was held on November 30, 1999, between F. Lyon and B. Thomas (NRC) and L. Lahti, W. MacRae, S. McBee, and others (IM) to clarify the information contained in the September 23, 1999, application. An additional telephone call was held on December 3, 1999, between F. Lyon and B. Jain (NRC) and J. Burford, W. MacRae, S. McBee, and others (IM) to clarify the information contained in the September 23, 1999, application. In the discussions, the licensee confirmed that design margins in the rope tension, crane bridge stress, supporting concrete structures stresses, and the wheel loads are within the existing design basis for CNP. The licensee stated that it considered variation in lifted loads and assured itself that the variation in crane's natural frequencies due to lifted loads of less than 300 tons will not reduce the design margins in various crane components and supporting structures. The licensee also stated that in the development of the vertical seismic spectra at the crane level, soil-structure-interaction effects were modeled in accordance with the staff's guidelines for SEP [systematic evaluation program] facilities contained in NRC letter LS05-80-12-035 from D. Crutchfield (NRC) to all SEP facilities, dated December 15, 1980. The vertical seismic model was analyzed for two foundation seismic motion input spectra shapes: the Housner's spectra, and the NUREG-0098 spectra. The licensee enveloped the vertical spectra at the crane level from the two seismic inputs. The licensee also stated their intent to verify that movement resulting from the pendulum motion of a seismic event would not impact safety-related equipment.

The runway beam system used for moving the loads through the containment equipment hatch uses a simple design of carts with guided rollers. The structure is temporary and is supported by the containment equipment hatch and the auxiliary building and containment building floor slabs. The licensee evaluated the runway beam system for the static and dynamic loads imposed by the SG sections, cart, and rigging. Similar to the Unit 2 SGRP, the loaded runway beam system was not evaluated for seismic loads, but the design provides a defined travel path that is located within an evaluated auxiliary building load handling area. Consistent with NUREG-0612 safe load path guidelines, this minimizes the potential for impacting equipment important to safety and ensures that the requirements for safe shutdown, decay heat removal, and SFP cooling continue to be met in the event of a load drop.

Based upon the cranes meeting the single-failure-proof criteria of NUREG-0554 and the previous NRC evaluation and approval for the movement of heavy loads during the Unit 2 SGRP, the staff finds the results of the licensee's seismic evaluation reasonable.

2.4 TECHNICAL SPECIFICATIONS

The licensee proposes to disengage the crane travel interlocks to accommodate movement of the cranes at the southwest corner of the SFP and to change TSSR 4.9.7.1 by adding the statement, "This Surveillance Requirement is not required during the movement of steam generator sections in the auxiliary building for the Unit 1 steam generator replacement project. When crane travel interlocks are disengaged, administrative controls shall be in place to prevent loads from passing over the spent fuel pool."

TS 3.9.7 prohibits the movement of loads in excess of 2500 pounds over fuel assemblies in the SFP. Associated TSSR 4.9.7.1 requires that the crane interlocks that limit crane travel and help ensure compliance with TS 3.9.7 are demonstrated operable within seven days of crane use and at least once per seven days thereafter.

The TS Bases state that the restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident, consistent with the activity release assumed in the accident analysis. The licensee states that, since no portion of the steam generators will pass over any part of the SFP, TS 3.9.7 will be met during the proposed load handling. The center of gravity of the rigging assembly remains outside of the SFP exclusion zone, which consists of the SFP and an additional margin beyond the border of the pool. Maintaining the center of gravity at this distance provides additional assurance that a drop of the rigging assembly would not result in impact to spent fuel assemblies, thus meeting TS 3.9.7 and the purpose of the associated interlocks. Furthermore, NUREG-0612, Section 5.1.2, states that meeting the single-failure-proof criteria of NUREG-0554 is a satisfactory alternative to crane travel interlocks. Procedural controls discussed in Sections 2.1.1, 2.1.2, and 2.1.3 above provide reasonable assurance to prevent loads from passing over the SFP.

Therefore, the staff finds that the proposed change to TSSR 4.9.7.1 is acceptable.

2.5 CONCLUSION

Based on the above evaluation, the staff finds that the movement of steam generator sections and disengaging the crane travel interlocks of TSSR 4.9.7.1 to accommodate movement of the cranes for the Unit 1 SGRP are acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 57665). Accordingly, the amendments meet 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 amendments.

5.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Lyon

Date: December 7, 1999

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Subject:
REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF A MENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6665 AND MA6666)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 7, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6665 AND MA6666)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: September 23, 1999, as supplemented October 11 and November 10, 1999

Brief description of amendments: The amendments provide approval to move steam generator sections through the auxiliary building and to disengage crane travel interlocks, and provide relief from performance of Technical Specification Surveillance Requirement 4.9.7.1. The loads to be moved are in support of the Unit 1 Steam Generator Replacement Project.

Date of issuance: December 7, 1999

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 233 and 216

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 26, 1999 (64 FR 57665). The October 11, 1999, submittal provided corrected TS pages. The November 10, 1999, submittal was in response to a NRC request for additional information dated October 26, 1999, and provided clarifying information to the original submittal. This information was

DFx2

within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 1999.

No significant hazards consideration comments received: No.

within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 1999.

No significant hazards consideration comments received: No.

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within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 1999.

No significant hazards consideration comments received: No.

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Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

November 30, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TIGHTER RESTRICTIONS ON ALLOWED OUTAGE TIME FOR REFUELING WATER STORAGE TANK (RWST) WATER LEVEL INSTRUMENTATION (TAC NOS. MA3797 AND MA3798)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. DPR-58 and Amendment No. 215 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 8, 1998.

The amendments would change the TSs for both units to place tighter restrictions on the allowed outage time for the RWST water level instrumentation.

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

Sincerely,

Original signed by:

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 232 to DPR-58
2. Amendment No. 215 to DPR-74
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 30, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TIGHTER RESTRICTIONS ON ALLOWED OUTAGE TIME FOR REFUELING WATER STORAGE TANK (RWST) WATER LEVEL INSTRUMENTATION (TAC NOS. MA3797 AND MA3798)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. DPR-58 and Amendment No. 215 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 8, 1998.

The amendments would change the TSs for both units to place tighter restrictions on the allowed outage time for the RWST water level instrumentation.

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

Sincerely,

John F. Stang

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No.232 to DPR-58
2. Amendment No.215 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232.
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 October 8, 1998, 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 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.

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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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 232, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: November 30, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 232

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 3-54	3/4 3-54
B 3/4 3-6	B 3/4 3-6

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The post-accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-11 (except item 8), either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE post-accident monitoring channels one less than required by Table 3.3-11, item 8, Refueling Water Storage Tank Water Level:

1. Either restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours, and

2. Within one hour, bypass the Residual Heat Removal Pump trip function from the Refueling Water Storage Tank Water Level for the pump associated with the out-of-service instrument.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Appendix R remote shutdown instrumentation ensures that sufficient instrumentation is available to permit shutdown of the facility to COLD SHUTDOWN conditions at the local shutdown indication (LSI) panel. In the event of a fire, normal power to the LSI panels may be lost. As a result, capability to repair the LSI panels from Unit 2 has been provided. If the alternate power supply is not available, fire watches will be established in those fire areas where loss of normal power to the LSI panels could occur in the event of fire. This will consist of either establishing continuous fire watches or verifying OPERABILITY of fire detectors per Specification 4.3.3.7 and establishing hourly fire watches. The details of how these fire watches are to be implemented are included in a plant procedure.

3/4.3.3.8 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. The allowable out-of-service time for the Refueling Water Storage Tank (RWST) level channels is required to provide the overall reliability to support the manual transfer from injection to recirculation following an accident. The bypassing of the Residual Heat Removal (RHR) pump trip from the RWST low level, with a level channel out-of-service, ensures that RHR pump will be available to meet its Engineered Safety Features (ESF) Function of injecting water into the core. The loss of the RHR pump protection will be mitigated by the operator's action to switch from injection to recirculation using the approved Emergency Operating Procedure which causes the RHR pump suction to be realigned well before the RHR pump trip setpoint. The associated RHR pump can be considered OPERABLE with the RWST level channel out-of-service once the trip function has been by-passed since the pump would be available to fulfill its ESF function.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 215
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 October 8, 1998, 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 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 215, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: November 30, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 215

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 3-45	3/4 3-45
B 3/4 3-3	B 3/4 3-3

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10 (except item 8), either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE post-accident monitoring channels one less than required by Table 3.3-10, item 8, Refueling Water Storage Tank Water Level:

1. Either restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours, and

2. Within one hour, bypass the Residual Heat Removal Pump trip function from the Refueling Water Storage Tank Water Level for the pump associated with the out-of-service instrument.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. The allowable out-of-service time for the Refueling Water Storage Tank (RWST) level channels is required to provide the overall reliability to support the manual transfer from injection to recirculation following an accident. The bypassing of the Residual Heat Removal (RHR) pump trip from the RWST low level, with a level channel out-of-service, ensures that the RHR pump will be available to meet it Engineered Safety Features (ESF) Function of injecting water into the core. The loss of RHR pump protection will be mitigated by the operator's action to switch from injection to recirculation using the approved Emergency Operating Procedure which causes the RHR pump suction to be realigned well before the RHR pump trip setpoint. The associated RHR pump can be considered OPERABLE with the RWST level channel out-of-service once the trip function has been by-passed since the pump would be available to fulfill its ESF function.

3/4.3.3.7 Deleted.

3/4.3.3.9 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the Waste Gas Holdup System. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 215 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated October 8, 1998, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would change the TSs for both units to place tighter restrictions on the allowed outage (AOT) time for the refueling water storage tank (RWST) water level instrumentation. In addition, the proposed change adds a required action to the TSs requiring the Residual Heat Removal (RHR) pump trip function from the RWST to be bypassed if the required number of RWST level instruments are not restored to operable within 72 hours.

2.0 EVALUATION

The RWST level instrumentation provides level indication in the control room for the operators to switch from the injection phase to the recirculation phase following a loss-of-coolant accident (LOCA). The instrumentation also provides a bi-stable input to trip the RHR pump when the RWST level falls below 20 percent to protect the RHR pump. The AOT of the RWST level instrumentation is required to provide the overall reliability of the instrument.

The proposed TS change is being requested in response to concerns raised during the architect engineering inspection performed by the Nuclear Regulatory Commission in late 1997. During the inspection, concerns were raised that the AOTs for the RWST level instruments were not as restrictive as other emergency core cooling systems (ECCS) sub-systems. The RWST water level instrumentation has no automatic engineered safety function but is used to initiate switchover from the injection to the recirculation phase following a LOCA.

The proposed TS change will reduce the AOT permitted by the TS action statement from the current TS time of 30 days if one or both instruments are out of service to 72 hours. In addition, the revised TS will also limit plant operation with both RWST level instruments out of service because this condition is outside of operation of the facility as defined in the TSs. Therefore, TS 3.0.3 would be applicable and require the unit to be in cold shutdown within 36 hours. Changing the AOT for the RWST level instrumentation from 30 days to 72 hours is conservative and will ensure that instruments are available on a greater frequency while the units are in

power operation. The proposed change brings the AOT for the RWST level instrumentation into alignment with the AOT constraints applied to other ECCS sub-systems.

The proposed TS change requires operators to bypass the RWST low-level trip of the RHR pumps within 1 hour of when an RWST level instrument becomes inoperable. Bypassing the low-level trip provided by the RWST level instrumentation ensures that the RHR pumps will be available to meet the Engineered Safety Features function of injecting water into the core following a design basis accident. The loss of the RHR pump protection will be mitigated by the operator's action to switch from injection to recirculation using the approved Emergency Operating Procedure, which causes the RHR pump suction to be realigned well before the low-level trip setpoint has been reached. The associated RHR pump can be considered operable with the RWST level instrumentation out of service once the tank low-level trip function has been bypassed.

The proposed TS changes reduce the AOT for the RWST level instrumentation, which will result in a more available instrument during power operation and brings the AOT for the RWST instrument into alignment with the AOT for other ECCS subsystems. In addition, the proposed TS requires the bypass of the low-level trip function of the RWST level instrumentation when an instrument is out of service to assure that the associated RHR pump will perform its intended function. Therefore, the staff finds that the proposed changes to the TSs acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 47532). Accordingly, the amendments meet 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 amendments.

5.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Stang

Date: November 30, 1999

November 30, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TIGHTER RESTRICTIONS ON ALLOWED OUTAGE TIME FOR REFUELING WATER STORAGE TANK (RWST) WATER LEVEL INSTRUMENTATION (TAC NOS. MA3797 AND MA3798)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. DPR-58 and Amendment No. 215 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 8, 1998.

The amendments would change the TSs for both units to place tighter restrictions on the allowed outage time for the RWST water level instrumentation.

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

Sincerely,

Original signed by:

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 232 to DPR-58
2. Amendment No. 215 to DPR-74
3. Safety Evaluation

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DATED: November 30, 1999

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-58, DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-74, DONALD C. COOK NUCLEAR PLANT, UNIT 2

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Subject:
REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA3797 AND MA3798)

Body:
PDR ADOCK 05000315 P

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

November 30, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA3797 AND MA3798)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: October 8, 1998

Brief description of amendments: The proposed amendments would change the Technical Specifications for both units to place tighter restrictions on the allowed outage time for the refueling water storage tank water level instrumentation.

Date of issuance: November 30, 1999

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 232 and 215

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: August 31, 1999 (64 FR 47532)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 30, 1999.

No significant hazards consideration comments received: No.

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November 30, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA3797 AND MA3798)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 30, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA3797 AND MA3798)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: October 8, 1998

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Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

November 23, 1999

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Reissued due to incorrect date

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA6766 AND MA6767)

Dear Mr. Powers:

Enclosed is a copy of a "Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," related to your request for license amendments dated November 5, 1999. The proposed license amendments would revise Technical Specification (TS) Surveillance Requirement 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. In addition, the proposed license amendments would revise TS 3.5.1 to change "pressurizer pressure" to "reactor coolant system pressure" in the applicability and action statement requirements. The Bases for TS 3/4.5.1 will also be revised to reflect both changes. Additionally, administrative changes are proposed to the page format.

This notice has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by:

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Notice

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 22, 1999

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA7049 AND MA7050)

Dear Mr. Powers:

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Sincerely,

John F. Stang

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Dockets Nos. 50-315 and 50-316

Enclosure: Notice

cc w/encl: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. DPR-58 and DPR-74 issued to Indiana Michigan Power Company (the licensee) for operation of the Donald C. Cook Nuclear Power Plant, Units 1 and 2, located in Berrien County, Michigan.

The proposed amendments would revise Technical Specification (T/S) Surveillance Requirement 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. In addition, the proposed license amendments would revise T/S 3.5.1 to change "pressurizer pressure" to "reactor coolant system pressure" in the applicability and action statement requirements. The Bases for T/S 3/4.5.1 will also be revised to reflect both changes. Additionally, administrative changes are proposed to the page format.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR

50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   The ECCS [emergency core cooling system] accumulators are used to mitigate the consequences of an accident after the event has occurred and do not initiate any accident previously evaluated. Demonstrating how power is removed from the valve operator does not initiate an accident. Inadvertently closing the valves cannot initiate an accident. Therefore, there is no significant increase in the probability of occurrence of an accident previously evaluated.

   The ECCS accumulators will still perform their function of injecting borated water into the reactor coolant loops following a large break loss-of-coolant accident, as described in Section 14.3.1 of the Updated Final Safety Analysis Report (UFSAR). A spurious closure of an accumulator outlet isolation valve is not a credible event. Performing T/S Surveillance Requirement 4.5.1.c provides assurance that one of the two actions required for spurious closure of the valve is precluded. The proposed change to the surveillance continues to provide assurance that power will be removed from each accumulator isolation valve operator so that the valves remain open. The consequences of accidents previously evaluated remained bounded because the accumulators will still function as assumed in the UFSAR accident analysis. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

   Changing "pressurizer pressure" to "RCS [reactor coolant system] pressure" has no significant effect on the applicability of the T/S requirements. RCS pressure and pressurizer pressure instrumentation measure a similar parameter in the primary coolant system. Since the RCS is a closed-loop fluid system, pressure instruments should indicate approximately the same value. There is no significant difference between the instrument readings because they are corrected for range, height, and accuracy. There is no significant change in the margin of pressure between when the accumulators are required to be aligned at 1000 psig and the upper limit specified in T/S 3.5.1.d of 658 psig.

   The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not increase the probability of occurrence or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes to T/S 3/4.5.1 and the associated Bases do not involve any physical changes to the plant, but do change the way the plant is operated by changing the method for ensuring spurious closure of the accumulator isolation valve will not occur. The proposed change to T/S Surveillance Requirement 4.5.1.c does not create any new operator actions. The position of the accumulator isolation valve remains open in Modes 1, 2, and 3 with RCS pressure greater than 1000 psig, which meets its design safety function. The proposed change does not increase the possibility of the accumulator valve repositioning. In order for repositioning to happen, the operator must close the molded-case circuit breaker coupled with either an active single failure or deliberate operator action in the control room. The proposed change of verifying that power is removed from the accumulator isolation valve provides the same level of protection. Two positive actions are required for the accumulator isolation valve to reposition.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

T/S Surveillance Requirement 4.5.1.c provides requirements that ensure that a single action will not cause an inadvertent closure of the accumulator isolation valves. The proposed change continues to ensure that two positive actions, an operator action to restore the breaker and a single failure, are required for valve closure.

Changing "pressurizer pressure" to "RCS pressure" does not impact operation of the accumulators. The proposed changes do not impact the nitrogen cover pressure as stated in T/S 3.5.1.c. The accumulators would not be expected to inject borated water until RCS pressure lowers to 658 psig (the upper limit specified in T/S 3.5.1.d). The change does not affect when this would occur after an accident. Therefore, changing "pressurizer pressure" to "RCS pressure" has no impact on plant operation.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By December 23, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to David. W. Jenkins, Esq., American Electric Power, Nuclear Generation Group, One Cook Place, Bridgman, MI 49106, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(I)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated November 5, 1999, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http:www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 18th day of November 1999.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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Docket: 05000316, Notes: N/A

CP-1

November 22, 1999

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA6766 AND MA6767)

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Original signed by:

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Notice

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December 23, 1999

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA6766 AND MA6767)

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Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 22, 1999

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA7049 AND MA7050)

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Project Directorate III
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Dockets Nos. 50-315 and 50-316

Enclosure: Notice

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cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. DPR-58 and DPR-74 issued to Indiana Michigan Power Company (the licensee) for operation of the Donald C. Cook Nuclear Power Plant, Units 1 and 2, located in Berrien County, Michigan.

The proposed amendments would revise Technical Specification (T/S) Surveillance Requirement 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. In addition, the proposed license amendments would revise T/S 3.5.1 to change "pressurizer pressure" to "reactor coolant system pressure" in the applicability and action statement requirements. The Bases for T/S 3/4.5.1 will also be revised to reflect both changes. Additionally, administrative changes are proposed to the page format.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR

9933302 94

50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   The ECCS [emergency core cooling system] accumulators are used to mitigate the consequences of an accident after the event has occurred and do not initiate any accident previously evaluated. Demonstrating how power is removed from the valve operator does not initiate an accident. Inadvertently closing the valves cannot initiate an accident. Therefore, there is no significant increase in the probability of occurrence of an accident previously evaluated.

   The ECCS accumulators will still perform their function of injecting borated water into the reactor coolant loops following a large break loss-of-coolant accident, as described in Section 14.3.1 of the Updated Final Safety Analysis Report (UFSAR). A spurious closure of an accumulator outlet isolation valve is not a credible event. Performing T/S Surveillance Requirement 4.5.1.c provides assurance that one of the two actions required for spurious closure of the valve is precluded. The proposed change to the surveillance continues to provide assurance that power will be removed from each accumulator isolation valve operator so that the valves remain open. The consequences of accidents previously evaluated remained bounded because the accumulators will still function as assumed in the UFSAR accident analysis. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

   Changing "pressurizer pressure" to "RCS [reactor coolant system] pressure" has no significant effect on the applicability of the T/S requirements. RCS pressure and pressurizer pressure instrumentation measure a similar parameter in the primary coolant system. Since the RCS is a closed-loop fluid system, pressure instruments should indicate approximately the same value. There is no significant difference between the instrument readings because they are corrected for range, height, and accuracy. There is no significant change in the margin of pressure between when the accumulators are required to be aligned at 1000 psig and the upper limit specified in T/S 3.5.1.d of 658 psig.

   The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not increase the probability of occurrence or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes to T/S 3/4.5.1 and the associated Bases do not involve any physical changes to the plant, but do change the way the plant is operated by changing the method for ensuring spurious closure of the accumulator isolation valve will not occur. The proposed change to T/S Surveillance Requirement 4.5.1.c does not create any new operator actions. The position of the accumulator isolation valve remains open in Modes 1, 2, and 3 with RCS pressure greater than 1000 psig, which meets its design safety function. The proposed change does not increase the possibility of the accumulator valve repositioning. In order for repositioning to happen, the operator must close the molded-case circuit breaker coupled with either an active single failure or deliberate operator action in the control room. The proposed change of verifying that power is removed from the accumulator isolation valve provides the same level of protection. Two positive actions are required for the accumulator isolation valve to reposition.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

T/S Surveillance Requirement 4.5.1.c provides requirements that ensure that a single action will not cause an inadvertent closure of the accumulator isolation valves. The proposed change continues to ensure that two positive actions, an operator action to restore the breaker and a single failure, are required for valve closure.

Changing "pressurizer pressure" to "RCS pressure" does not impact operation of the accumulators. The proposed changes do not impact the nitrogen cover pressure as stated in T/S 3.5.1.c. The accumulators would not be expected to inject borated water until RCS pressure lowers to 658 psig (the upper limit specified in T/S 3.5.1.d). The change does not affect when this would occur after an accident. Therefore, changing "pressurizer pressure" to "RCS pressure" has no impact on plant operation.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By December 23, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective,

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to David. W. Jenkins, Esq., American Electric Power, Nuclear Generation Group, One Cook Place, Bridgman, MI 49106, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(I)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated November 5, 1999, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http:www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 18th day of November 1999.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

11/19/99

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November 19, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF OUTAGE-RELATED AMENDMENTS (TAC NOS. MA6471 AND MA6472)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 231 to Facility Operating License No. DPR-58 and Amendment No. 214 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated September 10, 1999.

Units 1 and 2 are currently in extended outages. The units are defueled with the reactor coolant system (RCS) depressurized and there is no forced circulation of the reactor coolant. The licensee is replacing the steam generators in Unit 1 resulting in changes to the RCS configuration, coolant levels, and the ability to successfully sample the RCS. As a result, this amendment consists of changes to TS 3/4.4.7 "Reactor Coolant System Chemistry;" TS 3/4.11.2.2, "Radioactive Effluents, Gas Storage Tanks;" TS Table 4.4-3, "Reactor Coolant System Chemistry Limits Surveillance Requirements;" and TS Table 3.4-1, "Reactor Coolant System Chemistry Limits."

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

Sincerely,
Original signed by:
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 231 to DPR-58
2. Amendment No. 214 to DPR-74
3. Safety Evaluation

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November 19, 1999

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Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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1. Amendment No. 231 to DPR-58
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 19, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 231 to DPR-58
2. Amendment No. 214 to DPR-74
3. Safety Evaluation

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Donald C. Cook Nuclear Plant, Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 231
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 September 10, 1999, 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.

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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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 231, are hereby incorporated in the 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: November 19, 1999

ATTACHMENT TO LICENSE AMENDMENT No. 231

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
3/4 11-3	3/4 11-3

CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the Parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to $\leq$ 500 psig, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3. Performance of this surveillance is not required when the reactor is defueled with no forced circulation.

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	≤ 0.10 ppm	≤ 1.00 ppm
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

*Limits not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At lease once per 72 hours

*Not required with $T_{avg} \leq 250°F$.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.2 The quantity of radioactivity contained in each gas storage tank shall be limited to 43,800 curies noble gas (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days whenever radioactive materials are added to the tank and at least once per 24 hours during primary coolant system degassing operations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
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 10, 1999, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214, are hereby incorporated in the 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: November 19, 1999

ATTACHMENT TO LICENSE AMENDMENT No. 214

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 4-17	3/4 4-17
3/4 4-18	3/4 4-18
3/4 11-3	3/4 11-3

CHEMISTRY.

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.41.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the Parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to ≤ 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3. Performance of this surveillance is not required when the reactor is defueled with no forced circulation.

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	≤ 0.10 ppm	≤ 1.00 ppm
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

*Limits not applicable with T_{avg} ≤ 250°F.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.2 The quantity of radioactivity contained in each gas storage tank shall be limited to 43,800 curies noble gas (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days whenever radioactive materials are added to the tank and at least once per 24 hours during primary coolant system degassing operations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT No. 231 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT No. 214 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated September 10, 1999, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would revise TS 3/4.4.7 "Reactor Coolant System Chemistry;" TS 3/4 11.2.2, "Radioactive Effluents, Gas Storage Tanks;" TS Table 4.4-3, "Reactor Coolant System Chemistry Limits Surveillance Requirements;" and TS Table 3.4-1, "Reactor Coolant System Chemistry Limits."

Units 1 and 2 are currently in extended outages. The units are defueled with the reactor coolant system (RCS) depressurized and there is no forced circulation of the reactor coolant. Upcoming plant modifications and maintenance, including the replacement of the Unit 1 steam generators, necessitate changes to the RCS configuration and coolant levels. As a result of these changes, the licensee has proposed to change the above sections of the TSs to reflect the plant conditions, as well as making the Unit 1 and Unit 2 TSs consistent, which would more align the TSs to current industry standards.

2.0 EVALUATION

2.1 Proposed Change to Unit 1 and 2 TS Surveillance Requirement 4.4.7

The current Unit 1 and Unit 2 TS Surveillance Requirement 4.4.7 states that "The Reactor Coolant System Chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3." The licensee proposed to modify the TS by removing the requirement to perform the required surveillance when the reactor is defueled with no forced circulation. The TS is therefore proposed to read as "The Reactor Coolant System Chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3. *Performance of this surveillance is not required when the reactor is defueled with no forced circulation.*"

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TS 3/4.4.7 requires periodic sampling and analysis of the RCS to verify that chemistry parameters are below established limits. This TS places concentration limits on dissolved oxygen, chloride, and fluoride concentrations in the RCS. Sampling of the RCS chemistry provides assurance that the concentration of corrosive contaminants in the RCS is within acceptable levels and that the structural integrity of the RCS is maintained. Under normal operating procedures, the normal sampling system, which consists of several sampling points on two RCS hot legs and two residual heat removal system trains, is used to sample the chemistry parameters. However, the normal sampling system can not be used correctly under low-temperature, low-pressure conditions (*i.e.*, under conditions of no forced circulation and the reactor vessel being defueled). When the RCS is drained below mid-loop, the coolant remaining in the piping system low points is not in contact with coolant in other system low points. Therefore a representative sample of all coolant can not be taken. The RCS is designed with alternate sampling locations. However, these sampling locations are also insufficient for adequate test results due to the collection of reactor coolant in low points of the system. Moreover, some low points do not have sampling capability. With the lack of adequate sampling points, the licensee will not be able to take a representative sample of RCS coolant.

The Electric Power Research Institute document, TR-105714, "PWR Primary Water Chemistry Guidelines," dated March 1999, states that coolant temperature contributes more significantly to the rate at which stress corrosion and cracking occurs than does coolant chemistry. The licensee states that "The proposed change to modify the RCS chemistry sampling when fuel is off loaded and forced coolant circulation is not in use would only be in effect during low temperature and low pressure conditions." Therefore stress corrosion is not likely to occur under the conditions the licensee has proposed. Furthermore, no chemical contaminants are expected to be added to the system while under low-temperature, low-pressure conditions (*i.e.*, no change is expected in RCS chemistry). Additionally, administrative controls on RCS makeup sources, which consist of the primary water storage tank and refueling water storage tank, ensure that the concentration of chemical contaminants from these sources will not exceed the TS limits while RCS chemistry sampling is suspended. The licensee also states that the "RCS chemistry sampling is to be reinstated within 72 hours of returning forced circulation to operation and prior to entering Mode 6."

Based on the above, the staff finds that suspension of Surveillance Requirement 4.4.7 with the reactor defueled with no forced circulation does not constitute a reduction in safety. Therefore the staff finds the proposed change acceptable.

2.2 <u>Proposed Editorial Change to Unit 1 and 2 Table 3.4-1</u>

<u>Changes to Unit 1 Table 3.4-1</u>

Unit 1 Table 3.4-1 defines the chemistry limits in terms of steady-state and transient limits. The licensee proposes to remove asterisks for a footnote from the allowable chemistry limits of steady state and transient limits for dissolved oxygen. The asterisk is proposed to be placed by the dissolved oxygen parameter. Additionally, the licensee proposes to

modify the footnote. The footnote currently states "Limit not applicable with average temperature less than or equal to 250 degrees Fahrenheit." The footnote is proposed to read as "Limits not applicable with average temperature less than or equal to 250 degrees Fahrenheit."

The proposed changes with the asterisks for Table 3.4-1 are meant to provide consistency between the Unit 1 and Unit 2 TSs and NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." The proposed changes are not intended to affect the TS requirement. The plural of the word "limit" is proposed because the word applies to both the steady state and transient limits. The editorial change is not intended to alter the requirement or safety function.

The staff finds that the proposed editorial changes do not represent a reduction in safety or alter the TS requirement. The editorial changes are intended to maintain consistency and enhance usability and clarity of the TS. Therefore, the staff finds the proposed changes are acceptable.

<u>Change to Unit 2 Table 3.4-1</u>

Unit 2 Table 3.4-1 footnote reads as "Limit not applicable with average temperature less than or equal to 250 degrees Fahrenheit." The footnote is proposed to read as "Limits not applicable with average temperature less than or equal to 250 degrees Fahrenheit."

The plural of the word "limit" is proposed because the word applies to both the steady state and transient limits. This editorial change is not intended to alter the requirement or safety function, but is intended to provide clarity and consistency between the two units.

The staff finds that the proposed editorial change does not represent a reduction in safety or alter the TS requirement. The editorial change is intended to maintain consistency and enhance the usability and clarity of the TS. Therefore, the staff finds proposed change is acceptable.

2.3 <u>Proposed Change to Unit 1 Table 4.4-3</u>

Current Unit 1 Table 4.4.3 designates a maximum time interval between samples of 72 hours for the chemistry parameters of dissolved oxygen, chloride, and fluoride. Table 4.4.3 designates a minimum RCS analysis frequency of three times per 7 days. The licensee proposes to *remove* the "minimum analysis frequencies" requirement in Unit 1 Table 4.4-3 of "3 times per 7 days" for the dissolved oxygen, chloride, and fluoride parameters. The licensee also proposes to *remove* the "maximum time between analyses" requirement of 72 hours for dissolved oxygen, chloride, and fluoride. The licensee proposes to consolidate the two requirements by *inserting* a "sample and analysis frequency" requirement of "at least once per 72 hours" for dissolved oxygen, chloride, and fluoride. Additionally, the licensee proposes to *add* an asterisk to the dissolved oxygen parameter for reference to a footnote. The proposed changes would make the Unit 1 TS similar to the current Unit 2 TS.

The proposed change to determine the chemistry parameters concentration is not intended to affect the maximum interval between samples. It is intended to change the RCS chemistry sampling from three times per 7 days with a maximum interval of 72 hours *to* a frequency of at least once per 72 hours. The proposed change is consistent with the approved Unit 2 TS and with guidance provided in NUREG-0452. Retaining the bounding 72-hour surveillance requirement, while deleting the redundant requirement to sample three times per 7 days, provides assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective actions. The requirement itself is not altered.

The staff finds that the proposed change does not constitute a reduction in safety and is intended to maintain consistency between the Unit 1 and Unit 2 TSs and enhance the usability and clarity of the TS. Therefore, the staff finds the proposed change is acceptable.

2.4 Proposed Change to Unit 1 and 2 TS Surveillance Requirement 4.11.2.2

Current Unit 1 and 2 TS Surveillance Requirement 4.11.2.2 states that "The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days whenever radioactive materials are added to the tank and at least once per 24 hours during primary coolant system degassing operations, by analysis of the Reactor Coolant System noble gases." The licensee proposes to *delete "by analysis of the Reactor Coolant System noble gases."* The proposed surveillance requirement is to read as "The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days whenever radioactive materials are added to the tank and at least once per 24 hours during primary coolant system degassing operations."

The proposed change would delete the descriptive methods used to demonstrate compliance with the TS and is not intended to alter the general requirement to verify compliance with the TS limits. The change is intended to allow for alternate demonstrations of how the TS can be met. An example of an alternate testing method is direct gas sampling of the gas storage tanks. The current method requiring analysis of RCS noble gases is described in the licensee's Updated Final Safety Analysis Report. The licensee states that "Plant procedures will be revised to specify allowable sampling methods" (e.g., direct gas sampling of the gas storage tank or analysis of the RCS noble gases), and that "Implementation of alternative sampling approaches will be evaluated in accordance with 10 CFR 50.59." The licensee also states that "occupational dose associated with all sampling and analysis activities will be maintained within the established regulatory and procedural limits. Adherence to as low as reasonably achievable (ALARA) principles will provide additional assurance that these activities will not result in a significant increase in radiation exposure." The proposed change provides consistency with the gas storage tank sampling requirements in NUREG-0452.

The staff finds that the proposed change does not constitute a reduction in safety. The change is intended to allow for alternate methods of meeting the requirement and will be controlled by the licensee in accordance 10 CFR 50.59. Therefore, the staff finds the proposed change is acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 54376). Accordingly, the amendments meet 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 amendments.

5.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: K. Leigh
J. Stang

Date: November 19, 1999

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Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 19, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6471 AND MA6472)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: September 10, 1999

Brief description of amendments: The amendments revise Technical Specification (TS) 3/4.4.7 so that the surveillance requirement does not need to be performed when the reactor is defueled with no forced circulation. The revision to TS 3/4.4.7 also includes changes to Tables 3.4-1 and 4.4-3. TS Table 4.4-3 is revised to change the reactor coolant system (RCS) chemistry sampling frequency from three times per 7 days with a maximum interval of 72 hours to a frequency of at least once per 72 hours. An editorial change to Unit 1 Tables 3.4-1 and 4.4-3 relocates the asterisk for the footnote to a position adjacent to the parameter "dissolved oxygen," from its current position next to the allowable chemistry limit in Table 3.4-1 and the analysis frequency in Table 4.4-3. An editorial change also corrects the footnote for Table 3.4-1 for Unit 1 and Unit 2 by making the word "limit" plural, as it applies to both the steady-state and transient limits. Surveillance Requirement 4.11.2.2 is revised to delete the phrase "by analysis of the Reactor Coolant System noble gases."

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Date of issuance: November 19, 1999

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 231 and 214

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 6, 1999 (64 FR 54376)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 1999.

No significant hazards consideration comments received: No.

Date of issuance: November 19, 1999

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 231 and 214

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 6, 1999 (64 FR 54376)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 1999

No significant hazards consideration comments received: No.

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Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 231 and 214

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 6, 1999 (64 FR 54376)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 1999

No significant hazards consideration comments received: No.

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Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

OP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 18, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (REPEAT OF INDIVIDUAL NOTICE) (TAC NOS. MA7049 AND MA7050)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: November 5, 1999

Description of amendment requests: The proposed license amendments would revise Technical Specification (T/S) Surveillance Requirement 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. In addition, the proposed license amendments would revise T/S 3.5.1 to change "pressurizer pressure" to "reactor coolant system pressure" in the applicability and action statement requirements. The Bases for T/S 3/4.5.1 will also be revised to reflect both changes. Additionally, administrative changes are proposed to the page format.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

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The ECCS [emergency core cooling system] accumulators are used to mitigate the consequences of an accident after the event has occurred and do not initiate any accident previously evaluated. Demonstrating how power is removed from the valve operator does not initiate an accident. Inadvertently closing the valves cannot initiate an accident. Therefore, there is no significant increase in the probability of occurrence of an accident previously evaluated.

The ECCS accumulators will still perform their function of injecting borated water into the reactor coolant loops following a large break loss-of-coolant accident, as described in Section 14.3.1 of the Updated Final Safety Analysis Report (UFSAR). A spurious closure of an accumulator outlet isolation valve is not a credible event. Performing T/S Surveillance Requirement 4.5.1.c provides assurance that one of the two actions required for spurious closure of the valve is precluded. The proposed change to the surveillance continues to provide assurance that power will be removed from each accumulator isolation valve operator so that the valves remain open. The consequences of accidents previously evaluated remained bounded because the accumulators will still function as assumed in the UFSAR accident analysis. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

Changing "pressurizer pressure" to "RCS [reactor coolant system] pressure" has no significant effect on the applicability of the T/S requirements. RCS pressure and pressurizer pressure instrumentation measure a similar parameter in the primary coolant system. Since the RCS is a closed-loop fluid system, pressure instruments should indicate approximately the same value. There is no significant difference between the instrument readings because they are corrected for range, height, and accuracy. There is no significant change in the margin of pressure between when the accumulators are required to be aligned at 1000 psig and the upper limit specified in T/S 3.5.1.d of 658 psig.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not increase the probability of occurrence or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes to T/S 3/4.5.1 and the associated Bases do not involve any physical changes to the plant, but do change the way the plant is operated by changing the method for ensuring spurious closure of the accumulator isolation valve will not occur. The proposed change to T/S Surveillance Requirement 4.5.1.c does not create any new operator actions. The position of the accumulator isolation valve remains open in Modes 1, 2, and 3 with RCS pressure greater than 1000 psig, which meets its design

safety function. The proposed change does not increase the possibility of the accumulator valve repositioning. In order for repositioning to happen, the operator must close the molded-case circuit breaker coupled with either an active single failure or deliberate operator action in the control room. The proposed change of verifying that power is removed from the accumulator isolation valve provides the same level of protection. Two positive actions are required for the accumulator isolation valve to reposition.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

T/S Surveillance Requirement 4.5.1.c provides requirements that ensure that a single action will not cause an inadvertent closure of the accumulator isolation valves. The proposed change continues to ensure that two positive actions, an operator action to restore the breaker and a single failure, are required for valve closure.

Changing "pressurizer pressure" to "RCS pressure" does not impact operation of the accumulators. The proposed changes do not impact the nitrogen cover pressure as stated in T/S 3.5.1.c. The accumulators would not be expected to inject borated water until RCS pressure lowers to 658 psig (the upper limit specified in T/S 3.5.1.d). The change does not affect when this would occur after an accident. Therefore, changing "pressurizer pressure" to "RCS pressure" has no impact on plant operation.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 (c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involves no significant hazards consideration.

<u>Attorney for licensee</u>: David W Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

<u>NRC Section Chief</u>: Claudia M. Craig

safety function. The proposed change does not increase the possibility of the accumulator valve repositioning. In order for repositioning to happen, the operator must close the molded-case circuit breaker coupled with either an active single failure or deliberate operator action in the control room. The proposed change of verifying that power is removed from the accumulator isolation valve provides the same level of protection. Two positive actions are required for the accumulator isolation valve to reposition.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

T/S Surveillance Requirement 4.5.1.c provides requirements that ensure that a single action will not cause an inadvertent closure of the accumulator isolation valves. The proposed change continues to ensure that two positive actions, an operator action to restore the breaker and a single failure, are required for valve closure.

Changing "pressurizer pressure" to "RCS pressure" does not impact operation of the accumulators. The proposed changes do not impact the nitrogen cover pressure as stated in T/S 3.5.1.c. The accumulators would not be expected to inject borated water until RCS pressure lowers to 658 psig (the upper limit specified in T/S 3.5.1.d). The change does not affect when this would occur after an accident. Therefore, changing "pressurizer pressure" to "RCS pressure" has no impact on plant operation.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 (c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involves no significant hazards consideration.

<u>Attorney for licensee</u>: David W Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

<u>NRC Section Chief</u>: Claudia M. Craig

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safety function. The proposed change does not increase the possibility of the accumulator valve repositioning. In order for repositioning to happen, the operator must close the molded-case circuit breaker coupled with either an active single failure or deliberate operator action in the control room. The proposed change of verifying that power is removed from the accumulator isolation valve provides the same level of protection. Two positive actions are required for the accumulator isolation valve to reposition.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

T/S Surveillance Requirement 4.5.1.c provides requirements that ensure that a single action will not cause an inadvertent closure of the accumulator isolation valves. The proposed change continues to ensure that two positive actions, an operator action to restore the breaker and a single failure, are required for valve closure.

Changing "pressurizer pressure" to "RCS pressure" does not impact operation of the accumulators. The proposed changes do not impact the nitrogen cover pressure as stated in T/S 3.5.1.c. The accumulators would not be expected to inject borated water until RCS pressure lowers to 658 psig (the upper limit specified in T/S 3.5.1.d). The change does not affect when this would occur after an accident. Therefore, changing "pressurizer pressure" to "RCS pressure" has no impact on plant operation.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 (c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involves no significant hazards consideration.

Attorney for licensee: David W Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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Subject:
REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

OP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 17, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA7041 AND MA7042)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: November 3, 1999

Description of amendment requests: The proposed amendments would allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed T/S [Technical Specification] change to allow storage and use of fuel rods clad with ZIRLO does not significantly increase the probability of occurrence of an accident. Fuel assemblies are not an initiator or precursor to any previously evaluated accident. The proposed T/S change does not change or alter the design criteria for the systems or components used to mitigate the consequences of any design basis accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design

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methodology. Therefore, accident analysis results are not impacted. The operating limits are not changed and the analysis methods to demonstrate operation within the limits remain in accordance with NRC-approved methodologies. Other than the changes to the fuel rod cladding there are no physical changes to the plant associated with this T/S change. A safety analysis is still required to be performed for each specific reload cycle to demonstrate compliance with fuel safety design bases. The 10 CFR 50.46 emergency core cooling system acceptance criteria are applied to the ZIRLO clad fuel rods. The use of fuel assemblies containing ZIRLO clad fuel rods does not result in a change to the reload design and safety analysis limits. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel rod cladding material used, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks does not change or alter the design criteria for the systems or components used to mitigate the consequences of any design basis accident. This alternate methodology is more conservative with respect to determining the reactivity of the stored fuel assemblies than the methodology currently specified in the T/S. Therefore, the probability of an accidental criticality is less with the proposed T/S change than currently assumed. Since a criticality accident is precluded by the proposed T/S change, the consequences of a criticality accident are not changed by the use of this alternate methodology.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is administrative, and does not alter the design and analysis requirements that ensure storage of fuel in safe configurations. The existing T/S requirements for maximum enrichment, reactivity, and spacing of fuel assemblies in the spent fuel storage racks and new fuel storage racks are not altered by this change.

Based on the above discussions, design basis accident analyses affected by these T/S changes remain valid, and the consequences of an accident previously evaluated are not significantly increased by these changes.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed T/S change to allow storage and use of fuel rods clad with ZIRLO cannot create a new or different kind of accident. Fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for fuel assemblies with Zircaloy-4 clad fuel rods. The design and performance criteria continue to be met and no new failure mechanisms have been identified. Since the original design criteria are met, the ZIRLO clad fuel rods cannot be an initiator for any new accident. The ZIRLO cladding material offers improved corrosion resistance and structural integrity. The proposed changes do not affect the design or operation of any other system or component in the plant. The safety functions of the other structures, systems, or components are not changed in any manner, nor is the reliability of any other structure, system, or component reduced. The changes do not affect the manner by which the facility is operated and do not change any other facility design feature, structure, or system. No new or different types of permanent plant equipment are installed by this proposed T/S change. In addition, the use of ZIRLO fuel assemblies does not involve any alterations to permanent plant equipment or plant operating procedures that would introduce any new or unique operational mode or accident precursor.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks ensures that a conservative methodology is used to verify the licensing basis reactivity limits are not exceeded. The proposed change does not affect any permanent plant equipment or plant operating procedures, and cannot be an initiator of an event.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is an administrative change only. The proposed change does not affect any permanent plant equipment or plant operating procedures, and cannot be an initiator of an event.

Since there is no change to the permanent facility or plant operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, it is concluded that the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed T/S change to allow storage and use of fuel rods clad with ZIRLO does not change the reactor fuel reload design and safety analysis limits. The use of these fuel assemblies takes into consideration the core operating conditions allowed in the T/S. For each cycle reload core, the fuel assembly design and core configuration are evaluated using NRC-approved

reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects. The design basis and modeling techniques for fuel assemblies with Zircaloy-4 clad fuel rods remain valid for fuel assemblies with ZIRLO clad fuel rods. Use of ZIRLO cladding material has no effect on the criticality analysis for the spent fuel storage racks and the new fuel storage racks. Furthermore, it has no effect on the thermal-hydraulic and structural analysis for the spent fuel pool. Therefore, the design and safety analysis limits specified in the T/S are maintained with this proposed change.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks ensures that a conservative methodology is used to verify the licensing basis reactivity limits are not exceeded. Therefore, the existing T/S margin for reactivity control in the new fuel storage racks is maintained by this proposed change.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is an administrative change, and does not alter any of the existing T/S limits governing storage and use of reactor fuel.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

<u>Attorney for licensee</u>: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

<u>NRC Section Chief</u>: Claudia M. Craig

reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects. The design basis and modeling techniques for fuel assemblies with Zircaloy-4 clad fuel rods remain valid for fuel assemblies with ZIRLO clad fuel rods. Use of ZIRLO cladding material has no effect on the criticality analysis for the spent fuel storage racks and the new fuel storage racks. Furthermore, it has no effect on the thermal-hydraulic and structural analysis for the spent fuel pool. Therefore, the design and safety analysis limits specified in the T/S are maintained with this proposed change.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks ensures that a conservative methodology is used to verify the licensing basis reactivity limits are not exceeded. Therefore, the existing T/S margin for reactivity control in the new fuel storage racks is maintained by this proposed change.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is an administrative change, and does not alter any of the existing T/S limits governing storage and use of reactor fuel.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is an administrative change, and does not alter any of the existing T/S limits governing storage and use of reactor fuel.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR HEARING.

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 4, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING - (REPEAT OF INDIVIDUAL NOTICE) (TAC NOS. MA6766 AND MA6767)

Indiana Michigan Power Company, Docket No. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: October 1, 1999

Brief description of amendment request: The proposed amendments involve the resolution of an unreviewed safety question related to certain small-break loss-of-coolant accident scenarios for which there may not be sufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system and containment spray system pumps during and following switchover to cold leg recirculation. Resolution of this issue consists of a combination of physical plant modifications, new analyses of containment recirculation sump inventory, and resultant changes to the accident analyses to ensure sufficient water inventory in the containment recirculation sump. In addition, the licensee proposes to change the Technical Specifications dealing with the refueling water storage tank inventory and temperature, the required amount of ice in each ice basket in the containment, and the delay to start the containment air recirculation/ hydrogen skimmer fans.

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[illegible]

Date of publication of individual notice in FEDERAL REGISTER: October 29, 1999 (64 FR 58458)

Expiration date of individual notice: November 29, 1999

Date of publication of individual notice in FEDERAL REGISTER: October 29, 1999 (64 FR 58458)

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REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR HEARING - (REPEAT OF INDIVIDUAL NOTICE)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

November 4, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING - **(REPEAT OF INDIVIDUAL NOTICE)** (TAC NOS. MA6665 AND MA6666)

Indiana Michigan Power Company, Docket No. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: September 23, 1999, as supplemented October 11, 1999

Brief description of amendment request: The proposed amendments involve movement of loads in excess of the design-basis seismic capability of the auxiliary building load handling equipment and structures. The proposed amendment requests approval to move the steam generator sections through the auxiliary building and to disengage crane travel interlocks, and also requests relief from performance of Technical Specification Surveillance Requirement 4.9.7.1.

Date of publication of individual notice in FEDERAL REGISTER: October 26, 1999 (64 FR 57665)

Expiration date of individual notice: November 26, 1999

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November 4, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING - (REPEAT OF INDIVIDUAL NOTICE) (TAC NOS. MA6665 AND MA6666)

Indiana Michigan Power Company, Docket No. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

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Date of publication of individual notice in FEDERAL REGISTER: October 26, 1999 (64 FR 57665)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 4, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section
Project Directorate IV & Decommissioning
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING - (REPEAT OF INDIVIDUAL NOTICE) (TAC NOS. MA6665 AND MA6666)

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Date of publication of individual notice in FEDERAL REGISTER: October 26, 1999 (64 FR 57665)

Expiration date of individual notice: November 26, 1999

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Donald C Cook, issuance of amendments regarding administrative changes, MA4922 & MA4923.

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Docket: 05000316

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OP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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NRR-058

March 31, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADMINISTRATIVE CHANGES (TAC NOS. MA4922 AND MA4923)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 243 to Facility Operating License No. DPR-58 and Amendment No. 224 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendment consists of changes to Appendix A, Technical Specifications (TSs), in response to your application dated December 3, 1998.

The amendment would make administrative changes to several TS to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical. The proposed changes have been reviewed by the NRC staff and are in accordance with the regulations.

Amendment No. 216 for Unit 1 contains a TS page that is affected by the enclosed Amendment No. 243. Since Amendment No. 216 may not be implemented until December 31, 2000, the NRC is issuing two sets of TS pages with the enclosed amendment. The first set should be inserted when Amendment No. 243 is implemented. The second set replaces the Amendment No. 216 page that is affected by Amendment No. 243 and should be inserted into Amendment No. 216.

DF01

ML003701978

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 243 to DPR-58
2. Amendment No. 224 to DPR-74
3. Safety Evaluation Report

cc w/encls: See next page

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

Sincerely,
/RA/
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 243 to DPR-58
2. Amendment No. 224 to DPR-74
3. Safety Evaluation Report

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cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 243
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (NRC) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated December 3, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amendment (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 the 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 243, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

NBajwa for

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 31, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 243

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 0-3	3/4 0-3
3/4 3-21a	3/4 3-21a
3/4 4-38	3/4 4-38
3/4 4-40	3/4 4-40
3/4 7-15	3/4 7-15
3/4 9-1	3/4 9-1
3/4 9-13	3/4 9-13
5-6	5-6
5-7b	5-7b
6-4	6-4

SURVEILLANCE REQUIREMENTS

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Deleted

4.0.7 Deleted

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3	14*
Pump Start		2/bus (T11A – Train B; T11D – Train A)			
Valve Actuation (Both trains)		2/bus on (T11A & T11B or 2/busses T11C & T11D)			
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2	18*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kV Bus Degraded Voltage	3/Bus (T11A – Train B; T11D – Train A)	2/Bus (T11A – Train B; T11D – Train A)	2/Bus (T11A – Train B; T11D – Train A)	1, 2, 3, 4	14*

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a - MODES 1, 2, 3 and 4.
Specification 3.7.3.1.b - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.3.1.b is applicable:

With no flowpath to Unit 2 available, return at least one flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return at least one flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

c. By verifying pump performance pursuant to Specification 4.0.5.

d. At least once per 18 months during shutdown, by verifying that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6

ACTION:

a. With the requirements of the above specification not satisfied, 1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, and 2) initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool† until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

   1. Deleted

   2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

---

* The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D.C. COOK - UNIT 2.

† This does not include the main load block. For purposes of this specification, a de-energized main load block need not be considered a load.

1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

Figure 5.6-3 intentionally deleted.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must be qualified as specified in Section 6.2.2.g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 DELETED

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 243

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of Amendment No. 216 with the attached revised page. This page replaces Amendment No. 216 that is affected by the issuance of the enclosed Amendment No. 243. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3/4 9-1	3/4 9-1

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6

ACTION:

a. With the requirements of the above specification not satisfied, 1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, and 2) initiate and continue boration at greater than or equal to 34 gpm of 6,550 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 316

DONALD C. COOK NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (NRC) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated December 3, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amendment (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 the 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 224 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

N Bajwa for

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 31, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 224

TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 0-3	3/4 0-3
3/4 0-4	3/4 0-4
3/4 3-11	3/4 3-11
3/4 3-20	3/4 3-20
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-34	3/4 3-34
3/4 3-44d	3/4 3-44d
3/4 3-47	3/4 3-47
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-33	3/4 4-33
3/4 4-35	3/4 4-35
3/4 4-37	3/4 4-37
3/4 5-4	3/4 5-4
3/4 5-8	3/4 5-8
3/4 6-12	3/4 6-12
3/4 6-14	3/4 6-14
3/4 6-47	3/4 6-47
3/4 7-12	3/4 7-12
3/4 7-13	3/4 7-13
3/4 7-16a	3/4 7-16a
3/4 7-20	3/4 7-20
3/4 8-13	3/4 8-13
3/4 8-15	3/4 8-15
3/4 9-12	3/4 9-12
5-6	5-6
5-8	5-8
6-4	6-4

SURVEILLANCE REQUIREMENTS

b. Surveillance Intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Deleted

4.0.7 Deleted

SURVEILLANCE REQUIREMENTS

4.0.8 Deleted

4.0.9. Deleted

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2,8), M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)	M	1, 2
8. Overpower ΔT	S	R(9)	M	1, 2
9. Pressurizer Pressure -- Low	S	R	M	1, 2
10. Pressurizer Pressure -- High	S	R	M	1, 2
11. Pressurizer Water Level -- High	S	R	M	1, 2
12. Loss of Flow-Single Loop	S	R(8)	M	1

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kV Bus Degraded Voltage	3/Bus (T11A-Train B) (T11D-Train A)	2/Bus (T11A-Train B) (T11D-Train A)	2/Bus (T11A-Train B) (T11D-Train A)	1, 2, 3, 4	14*
9. MANUAL					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation-Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	2/train	1/train	2/train	1, 2, 3, 4	18
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
d. Steam Line Isolation	2/steam line (1 per train)	2/steam line (1 per train)	2/operating steam line (1 per train)	1, 2, 3	20

TABLE 4.3-2
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure -- High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure -- Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines -- High	S	R	M	N.A.	1, 2, 3
f. Steam Line Pressure -- Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure -- High-High	S	R	M(3)	N.A.	1, 2, 3
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	See Functional Unit 9				
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
b. Phase "B" Isolation					
1) Manual	See Functional Unit 9				
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
3) Containment Pressure-- High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Purge and Exhaust Isolation					
1) Manual	See Functional Unit 9				
2) Containment Radioactivity -- High	S	R	Q	N.A.	1, 2, 3, 4
4. STEAM LINE ISOLATION					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure -- High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines -- High Coincident with T_{avg} -- Low-Low	S	R	M	N.A.	1, 2, 3
e. Steam Line Pressure -- Low	S	R	M	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level -- High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kV Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 4.3-6A

APPENDIX R REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

	INSTRUMENT	LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Steam Generators 1 and 4 Level	LSI Cabinet 1 and LSI Cabinet 4	M	R
2.	Steam Generators 2 and 3 Level	LSI Cabinet 2 and LSI Cabinet 4	M	R
3.	Steam Generators 1 and 4 Pressure	LSI Cabinet 4 and LSI Cabinet 5	M	R
4.	Steam Generators 2 and 3 Pressure	LSI Cabinet 4 and LSI Cabinet 6	M	R
5.	Reactor Coolant Loop 4 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 5	M	R
6.	Reactor Coolant Loop 4 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 5	M	R
7.	Reactor Coolant Loop 2 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 6	M	R
8.	Reactor Coolant Loop 2 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 6	M	R
9.	Pressurizer Level	LSI Cabinet 3	M	R
10.	Reactor Coolant System Pressure	LSI Cabinet 3	M	R
11.	Charging Cross-Flow Between Units	Corridor Elev 587'	N/A	R*
12.	Source Range Neutron Detector (N-23)	LSI Cabinet 4	N/A	R

* Charging Cross-Flow between Units is an instrument common to both Unit 1 and 2. This surveillance will only be conducted on an interval consistent with Unit 1 refueling.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure	M	R
2.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4.	Reactor Coolant Pressure - Wide Range	M	R
5.	Pressurizer Water Level	M	R
6.	Steam Line Pressure	M	R
7.	Steam Generator Water Level - Narrow Range	M	R
8.	RWST Water Level	M	R
9.	Boric Acid Tank Solution Level	M	R
10.	Auxiliary Feedwater Flow Rate	M	R
11.	Reactor Coolant System Subcooling Margin Monitor	M	R
12.	PORV Position Indicator - Limit Switches	M	R
13.	PORV Block Valve Position Indicator - Limit Switches	M	R
14.	Safety Valve Position Indicator - Acoustic Monitor	M	R
15.	Incore Thermocouples (Core Exit Thermocouples)	M	R(1)
16.	Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	M(2)	R(3)
17.	Containment Sump Level	M	R
18.	Containment Water Level	M	R

(1) Partial range channel calibration for sensor to be performed below P-12 in MODE 3.

(2) With one train of Reactor Vessel Level Indication inoperable, Subcooling Margin Indication and Core Exit Thermocouples may be used to perform a CHANNEL CHECK to verify the remaining Reactor Vessel Indication train OPERABLE.

(3) Completion of channel calibration for sensors to be performed below P-12 in MODE 3.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 6.9.1	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3.	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1.	N/A	N/A

$S = 3(N+n)\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One of the containment atmosphere particulate radioactivity monitoring channels (ERS-2301 or ERS-2401),

b. The containment sump level and flow monitoring system, and

c. Either the containment humidity monitor or one of the containment atmosphere gaseous radioactivity monitoring channels (ERS-2305 or ERS-2405).

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring channels are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Containment humidity monitor (if being used) - performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

g. With PORVs and block valves not in the same line inoperable due to causes other than excessive seat leakage, within 1 hour restore the valves to OPERABLE status or close and de-energize the associated block valve and place the associated PORV in manual control in each respective line. Apply the portions of ACTION c or d above, relating to the OPERATIONAL MODE, as appropriate for two or three lines unavailable.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

c. At least once per 18 months by operating solenoid air control valves and check valves in PORV control systems through one complete cycle of full travel, and

d. At least once per 18 months by performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, c, or d in Specification 3.4.11.

4.4.11.3 Deleted.

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. IMO-390	a. RWST to RHR	a. Open
b. IMO-315	b. Low head SI to Hot Leg	b. Closed
c. IMO-325	c. Low head SI to Hot Leg	c. Closed
d. IMO-262*	d. Mini flow line	d. Open
e. IMO-263*	e. Mini flow line	e. Open
f. IMO-261*	f. SI Suction	f. Open
g. ICM-305*	g. Sump Line	g. Closed
h. ICM-306*	h. Sump Line	h. Closed

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

* These valves must change position during the switchover from injection to recirculation flow following LOCA.

[illegible]

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits, at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F as determined at least once per hour when any RCS cold leg temperature is between 152°F and 200°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.

d. At least once per 5 years by verifying the flow rate from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve specified shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-2.

b. Visually inspecting at least 95 percent of the seal's entire length and:

1. Verifying that the seal and seal mounting bolts are properly installed, and

2. Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flow path in support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a. - MODES 1, 2, 3, 4.
Specification 3.7.3.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.3.1.b is applicable:

With no flowpath to Unit 1 available, return at least one flowpath to available status within 7 days, or provide equivalent shutdown capability in Unit 1 and return at least one flow path to available status within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

c. By verifying pump performance pursuant to Specification 4.0.5.

d. At least once per 18 months during shutdown, verify that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1

a. At least two independent essential service water loops shall be OPERABLE.

b. At least one essential service water flowpath associated with support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.4.1.a. - MODES 1, 2, 3, and 4.
Specification 3.7.4.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.4.1.a is applicable:

With only one essential service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.4.1.b is applicable:

With no essential service water flow path available in support of Unit 1 shutdown functions, return at least one flow path to available status within 7 days or provide equivalent shutdown capability in Unit 1 and return the equipment to service within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two essential service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7.1 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.1.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspection

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 3.7-9. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 3.7-9 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 156.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified as acceptable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.13 volts under float charge.

2. The specific gravity, corrected to 77°F, and full electrolyte level (fluid at the bottom of the maximum level indication mark), of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the top of the minimum level indication mark and the bottom of the maximum level indication mark.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The battery charger will supply at least 140 amperes at greater than or equal to 250 volts for at least 4 hours.

d. At least once per 18 months, perform a battery service test during shutdown (MODES 5 or 6), by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status the actual or simulated emergency loads for the design duty cycle which is based on the composite load profile. The composite load profile envelopes both the LOCA/LOOP and Station Blackout profiles and provides the basis for the times listed in Table 4.8-2. The battery charger will be disconnected throughout the test. The battery terminal voltage shall be maintained greater than or equal to 210 volts throughout this test.

e. At least once per 60 months, conduct a performance test of battery capacity during shutdown (MODES 5 or 6), by verifying that the battery capacity is at least 80% of the manufacturer's rating. When this test is performed in place of a battery service test, a modified performance test shall be conducted.

Annual performance tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity, and has shown no signs of degradation, performance testing at two year intervals is acceptable until the battery shows signs of degradation.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 250-volt D.C. bus, and

1 - 250-volt battery bank and charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 250-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 250-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool+ until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

   1. Deleted.

   2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

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* The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D. C. COOK - UNIT 1.

\+ This does not include the main load block. For purposes of this specification, a de-energized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

CRITICALITY - SPENT FUEL (Continued)

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$- 22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$- 26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

5.6.1.2 Fuel stored in the spent fuel storage racks shall have a nominal fuel assembly enrichment as follows:

	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
1)	Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2)	Exxon/ANF	15 x 15	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

Figure 5.6-3 intentionally deleted.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must be qualified as specified in Section 6.2.2.g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 DELETED

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO243 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 224TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated December 3, 1998, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2 (D.C. Cook). The proposed amendments would make administrative changes to several TSs to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard TSs, provide clarification, and correct typographical errors.

2.0 EVALUATION

The evaluation of the proposed changes are described in the following paragraphs:

A. Proposed Revision to Boron Sampling Requirements in Mode 6

The current Unit 1 TS Surveillance Requirement 4.9.1.2 states that "The boron concentration of the reactor coolant system and refueling canal shall be determined by chemical analysis at least three times per seven days with a maximum time interval between samples of 72 hours." The licensee proposes to change the TS to read as "The boron concentration of the reactor coolant system and refueling canal shall be determined by chemical analysis at least once per 72 hours."

The purpose of TS 4.9.1.2 is to assure that a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The licensee has proposed to remove the Unit 1 restriction to determine the concentration at least three times per seven days in order to maintain consistency with the Unit 2 surveillance requirement, NUREG-1431, "Standard Technical Specifications," and NUREG-0452, "Standard Technical Specifications," Revision 4, fall 1981. The 72-hour maximum interval between samples is not changed. NUREG-1431 states that a minimum frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The

change does not constitute a reduction in safety. Therefore, the staff finds the proposed change acceptable.

B. Proposed Revision to Footnote for TS 3.9.12, Action a

TS 3.9.12, Action a, is modified by a footnote describing operation of the drumming room roll-up door. The current footnote states that "The crane bay roll-up door and the drumming room roll-up door may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool." The licensee has proposed to revise the footnote of TS 3.9.1.2, Action a, to read "The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool."

In the NRC's Safety Evaluation, the staff found that the operation of the crane bay roll-up door and drumming room roll-up door met the intent of Standard Review Plan, Sections 9.4.2, "Spent Fuel Pool Area Ventilation System," and 15.7.4, "Radiological Consequences of Fuel Handling Accidents," and was acceptable as such. The licensee proposes to remove "roll-up" from the description of the drumming room door. The drumming room door has been replaced by a door having a different design, although the function of the door remains the same. The replacement door provides a ventilation barrier as required by the analysis used in support of the previous amendment request. In addition, the name of the door is proposed to change to "south door of the auxiliary building crane bay" because it more accurately describes the door's current use. The staff finds the proposed changes to be acceptable, as the changes do not constitute a reduction in safety and clarify the TS.

C. Proposed Change to Figure 5.6-3

TS 5.6.1.1.c.3 includes equations for equivalent reactivity criteria for Region 2 and Region 3 in the spent fuel storage racks. The equations are also graphically depicted in Figure 5.6-3. Either the equations or the graph can be used to verify that fuel is stored in the appropriate region. The licensee proposes to delete Figure 5.6-3, as the information is redundant to the equations provided in TS 5.6.1.1.c.3.

The deletion does not alter the fuel storage requirements TSs and does not constitute a reduction in safety. Therefore, the staff finds the proposed change to be acceptable. The change is merely proposed to reduce unnecessary information in the TSs.

D. Proposed Change to Reference in TS 6.3.1

Current TS 6.3.1, "Facility Staff Qualifications," includes a requirement that the operations superintendent must hold or have held a senior operator license as specified in TS 6.2.2.h. The reference to 6.2.2.h is an administrative error made in previous amendments. Section 6.2.2.h does not exist. The proposed change provides a clear reference to the correct TS Section 6.2.2.g for the operations superintendent qualifications and does not change the current TS requirements. Therefore, the staff finds the change to be acceptable.

E. Proposed Deletion of Obsolete Notes

The following paragraphs state the proposed TS deletion and justification:

Current Surveillance Extensions Unit 1, TS 4.0.6 and 4.0.7

Unit 1, TS 4.0.6 states that "Amendments 100, 107 and 108 grant extensions for certain surveillances required to be performed on or before July 31, 1987, and until the end of the Cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1987 refueling outage."

TS 4.0.7 states that "Amendment 121 granted extensions for certain surveillances required to be performed on or before April 1, 1989, until the end of the Cycle 10-11 refueling outage. For these specific surveillances under this extension, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1989 refueling outage."

Current Surveillance Extensions Unit 2, TS 4.0.6, 4.0.7, 4.0.8, 4.0.9

For Unit 2, TS 4.0.6 states that "Amendment 78 granted extensions for certain surveillances required to be performed on or before March 31, 1986, until the end of the Cycle 5-6 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1986 refueling outage."

TS 4.0.7 states that "Amendments 97 and 99 granted extensions for certain surveillances required to be performed on or before July 1, 1988, until the end of the Cycle 6-7 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillances date during the Unit 2 1988 refueling outage."

TS 4.0.8 states that "By specific reference to this section, those surveillances which must be performed on or before August 13, 1994, and are designed as 18-month or 36-month surveillances (or required as outage-related surveillances under the provisions of Specification 4.0.5) may be delayed until the end of the Cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1994 refueling outage."

TS 4.0.9 states that "By specific reference to this section, those surveillances which must be performed on or before September 7, 1994, are designated as 18-month surveillances, so it may be delayed until just prior to core reload in the Unit 2 Cycle 9-10 refueling outage."

Justification for Deletion of Unit 1 TS 4.0.6, 4.0.7 and Unit 2 TS 4.0.6, 4.0.7, 4.0.8, and 4.0.9

Unit 1 TS 4.0.6 and TS 4.0.7, and Unit 2 TS 4.0.6, 4.0.7, 4.0.8, and 4.0.9 extensions were granted to accommodate scheduled work at the time and have been proposed to be deleted because they are no longer applicable. The references to Unit 1 TS 4.0.6 and 4.0.7 that indicated when the provision was applicable were deleted in previous amendments. The existing TS notes are no longer applicable and serve no function. For Unit 2, TSs 4.0.8 and 4.0.9 are proposed to be deleted because they no longer apply. References to TS 4.0.8 are also deleted from the following: Table 4.3-1, functional Units 7 through 11; Table 4.3-2, functional Units 1.d, 4.d, and 6.d; Table 4.3-6A, instruments 5 through 8; Table 4.3-10, items 2,

3, 11, 15, and 16; surveillance requirement (SR) 4.4.6.1.b; SR 4.4.11.1.d; SR 4.5.3.1; SR 4.6.2.2c; SR 4.6.3.1.2; SR 4.6.5.9; SR 4.7.3.1.b; SR 4.7.4.1.b; SR 4.7.5.1.e.2.a; SR 4.7.5.1.e.2.b; SR 4.7.7.1.a; and SR 4.8.1.2. Additionally, references to TS 4.0.9 are proposed to be deleted from SR 4.8.2.3.2.d and SR 4.8.2.4.2. The proposed changes are acceptable because all extensions mentioned above pertained to past refueling outages and are no longer applicable since the corresponding refueling outages have been completed. Deletion of the TSs do not eliminate any requirements. Therefore, the staff finds the proposed changes acceptable.

Surveillance Extensions Notes Unit 1 and 2, TS 4.4.12.1 and 4.4.12.2

For Unit 1 and 2, notes for SRs in 4.4.12.1 state: "Surveillance requirements to demonstrate the operability of each Reactor Vessel head vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification, and after the appropriate Plant Procedures have been written."

TS note 4.4.12.2 states: "Surveillance requirements to demonstrate the operability of each Pressurizer steam space vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification and after the appropriate Plant Procedures have been written."

Justification for Deletion of Notes for Unit 1 and Unit 2, TS 4.4.12.1 and 4.4.12.2

For Unit 1 and Unit 2, the notes for SRs 4.4.12.1 and 4.4.12.2 are proposed to be deleted. Plant Procedures 01-OHP-4030.STP.56 and 02-OHP-4030.STP.56 were developed to perform the surveillance. The surveillances are performed routinely as required and the exception allowed in the footnote is no longer applicable; therefore, the deletion of the notes for SRs in TS 4.4.12.1 and 4.4.12.2 are acceptable. The surveillance notes were written for a one-time relief extension and the relief extension time periods have expired. Deletion of the surveillance notes do not eliminate any requirements; therefore, the staff finds the proposed changes acceptable.

Note to Table 4.4-2, Unit 2

Table 4.4-2 and TS 3.3.3.1 include a note: "This Technical Specification will not be effective until after the 1982 refueling outage."

Justification for Deletion of Note to Table 4.4-2, Unit 2

For Unit 2, the notes for TS 3.3.3.1 and Table 4.4-2 are deleted. The provision has expired and it is no longer required to be included in the TSs. The staff finds the proposed change acceptable.

The exceptions that were granted in the above paragraphs have all expired and are no longer applicable. The changes do not represent a reduction in safety and deletion of the notes or TSs do not eliminate any requirements of the TSs. Therefore, the staff finds the proposed changes acceptable.

F. Proposed Changes to TS SR 4.7.3.1

Unit 1, TS 4.7.3.1

Unit 1 TS SR 4.7.3.1.d states: "At least once per 18 months during shutdown, by verifying that the cross-tie valves can cycle full travel." The licensee proposes to change the TS to "At least once per 18 months during shutdown, by verifying that the unit cross-tie valves can cycle full travel."

The clarification is proposed to indicate that the cross-tie valves are the unit cross-tie valves. The change is consistent with the equivalent Unit 2 requirement and is not intended to affect which valves are included in the surveillance. The staff finds the proposed change to be acceptable, as it does not eliminate or alter the TS requirement and allows for a clear reference in the TS.

Unit 2, TS 4.7.3.1

Unit 2 TS SR 4.7.3.1 reads: "At least two component cooling water loops shall be demonstrated OPERABLE," and has two requirements, 4.7.3.1.a and 4.7.3.1.b, to demonstrate operability of the component cooling water loops. A third requirement is included in SR 4.7.3.2. This requirement supports demonstrating that the system is operable. For Unit 2, the licensee proposes to add a new SR, 4.7.1.3.c ("By verifying pump performance pursuant to Specification 4.0.5"), to demonstrate operability by verifying pump performance pursuant to TS 4.0.5, and to renumber TS 4.7.3.2 as 4.7.3.1.d for consistency with Unit 1 TSs.

This change is consistent with the corresponding surveillance for Unit 1. In Amendment No. 164 to DPR-58 and Amendment No. 149 to DPR-74, the Nuclear Regulatory Commission approved changes requiring that all safety-related pumps in the TSs be tested at a frequency specified in TS 4.0.5. TS 4.0.5 states that safety-related pumps shall be tested in accordance with ASME Code, Section XI, unless written relief has been granted. The proposed changes are consistent with the changes approved in those amendments. There is no reduction in safety and deletion of the SRs do not eliminate or alter the TS requirement. Therefore, the staff finds the proposed change acceptable.

G. Proposed Change to Degraded Bus Voltage Instrumentation

Current Unit 1 and Unit 2 Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation," provides various instrumentation requirements. Functional Unit 8.b of Table 3.3-3 provides instrumentation requirements for the 4 kV bus degraded voltage (loss of power). On the 4 kV bus, there are three channels per bus total, with two channels per bus being required for operation in Modes 1, 2, 3, and 4. Two channels per bus are required to trip. These requirements provide assurance that the actuation will occur when required. The three channels are installed on buses T11A and T11D for Unit 1 and on buses T21A and T21D for Unit 2. The licensee proposes to add clarifying information to Table 3.3-3, to reflect the configuration of the instrumentation used to detect degraded voltage.

The configuration of the instrumentation used to detect degraded voltage may be unclear in Table 3.3-3. By design, the instrumentation is not installed on buses T11B or T11C for Unit 1 or on buses T21B or T21C for Unit 2. The licensee proposes to add references to the buses

with the appropriate instrumentation. This is similar to functional Unit 6.b, which has instrumentation requirements for 4 kV bus loss of voltage. The design and function of Unit 6.b has been reviewed and approved in the Safety Evaluation for Amendments No. 39 for Unit 1 and Amendment No. 22 for Unit 2. The proposed change is intended to clarify functional unit 8.b by indicating that the instrumentation is installed only on buses T11A and T11D. The change does not represent a reduction in safety, and allows for clarity in the TS. Therefore, the staff finds the proposed change acceptable.

H. Proposed Corrections to Typographical Errors

The licensee proposes to correct two typographical errors that were introduced in Amendment No. 131 to DPR-74 for Unit 2. These errors were to portions of the text that were not affected by the amendment. The proposed changes restore the text as it was issued in the previous amendment (Amendment 78). The valve number for line d of SR 4.5.2.a is changed from IMP-262 to IMO-262. Furthermore, the word "otherwise" is corrected in SR 4.5.2.b from the incorrect spelling of "otherswise."

Additionally, corrections are proposed to change "once" to "one" and "with" to "when" in TS 3.7.3.1 for Unit 1 and "in" to "to" in TS 3.7.3.1 for Unit 2.

These changes are acceptable since the changes are editorial and do not impact the requirements. The changes are intended to provide clarification and better direction to the operators. There is no reduction in safety by this change, therefore, the staff finds the proposed changes acceptable.

3.0 SUMMARY

The licensee has proposed changes to make several administrative changes to TSs for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed changes include: (1) revising boron sampling requirements in mode 6; (2) deleting a reference to obsolete equipment in a footnote, (3) deleting a redundant figure; (4) correcting a reference to another requirement; (5) deleting obsolete notes; (6) adding to SRs; (7) clarifying instrumentation configuration; and (8) correcting typographical errors. These changes are proposed to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors. The proposed amendment does not cause changes to accident initiators or precursors, or to the accident analyses, and does not involve a significant reduction of safety.

Based on the above evaluation, the staff finds that the proposed changes to the TSs are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the propose issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 50.35, an environmental assessment and finding of no significant impact have been prepared and published in the *Federal Register* on March 28, 2000 (65 FR 16421). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Contributor: John Stang
Kimberly Leigh

Date: March 31, 2000

50-315
3/22/2000

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DONALD C. COOK UNITS 1 AND 2 - ENVIRONMENTAL ASSESSMENT REGARDING AD MINISTRATIVE AMENDMENT (TAC NOS. MA4922 AND MA4923)

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ML003694163
NRR-042

March 22, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK UNITS 1 AND 2 - ENVIRONMENTAL ASSESSMENT REGARDING ADMINISTRATIVE AMENDMENT (TAC NOS. MA4922 AND MA4923)

Dear Mr. Powers:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your application for amendment dated December 3, 1998. The proposed amendment would make administrative and editorial changes to several Technical Specifications (TSs) to remove obsolete information, provide consistency between Unit 1 and 2 TSs, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors.

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

cc w/encl: See next page

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ML003694163
NRR-042

March 22, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK UNITS 1 AND 2 - ENVIRONMENTAL ASSESSMENT REGARDING ADMINISTRATIVE AMENDMENT (TAC NOS. MA4922 AND MA4923)

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 22, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK UNITS 1 AND 2 - ENVIRONMENTAL ASSESSMENT REGARDING ADMINISTRATIVE AMENDMENT (TAC NOS. MA4922 AND MA4923)

Dear Mr. Powers:

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

cc w/encl: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Sherry Kamke, Acting
Environmental Review Coordinator
U. S. EPA Region 4
61 Forsyth Street, SW
Atlanta, GA 30303-3104

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

DONALD C. COOK UNIT 1 AND 2

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an amendment to Facility Operating License No. DPR-58 and No. DPR-74, issued to Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook Nuclear Plant, Units 1 and 2, located in Berrien County, Michigan.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

The proposed action would make administrative and editorial changes to several Technical Specifications (TSs). The proposed changes include: (1) revising boron sampling requirements in mode 6; (2) deleting a reference to obsolete equipment in a footnote; (3) deleting a redundant figure; (4) correcting a reference to another requirement; (5) deleting obsolete notes; (6) adding to surveillance requirements; (7) clarifying instrumentation configuration; and (8) correcting typographical errors.

The proposed action is in accordance with the licensee's application for amendment dated December 3, 1998.

The Need for the Proposed Action:

These proposed changes are needed to remove obsolete information, provide consistency between Unit 1 and Unit 2 TSs, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors.

Environmental Impacts of the Proposed Action:

The Commission has completed its evaluation of the proposed action and concludes that the administrative and editorial changes do not impact any requirements. The proposed action does not modify the facility or affect the manner in which the facility is operated.

The proposed action will not increase the probability or consequences of accidents, no changes are being made in the types of any 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 non-radiological impacts, the proposed action does not involve 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 Commission concludes that there are no significant environmental impacts associated with the proposed action.

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:

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Donald C. Cook Nuclear Plant.

Agencies and Persons Consulted:

In accordance with its stated policy, on March 2, 2000, the staff consulted with the Michigan State official, Mr. David Minnaar of the Michigan 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 Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission 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 3, 1998, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http:\\www.nrc.gov (the Electronic Reading Room)

Dated at Rockville, Maryland, this 22d day of March 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

50-315
3/15/2000

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March 15, 2000

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1 /RA/
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA4929 AND MA4930)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: December 3, 1998

Brief description of amendments: The amendments incorporate the Distribution Ignition System requirements into the Unit 1 and Unit 2 Technical Specifications.

Date of issuance: March 15, 2000

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 242 and 223

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: January 26, 2000 (65 FR 4279)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 2000

No significant hazards consideration comments received: No.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 15, 2000

MORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA4929 AND MA4930)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

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Cook, Units 1 and 2, Amendments 242 and 223, Incorporate Distribution Ignition System Requirements into TSs, MA4929 AND MA4930

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March 14, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MA4929 AND MA4930)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 242 to Facility Operating License No. DPR-58 and Amendment No. 223 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 3, 1998. The amendments incorporate the Distribution Ignition System requirements into the Unit 1 and Unit 2 TSs.

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 242 to DPR-58
2. Amendment No. 223 to DPR-74
3. Safety Evaluation

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March 14, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MA4929 AND MA4930)

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Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 242 to DPR-58
2. Amendment No. 223 to DPR-74
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 15, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MA4929 AND MA4930)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 242 to Facility Operating License No. DPR-58 and Amendment No. 223 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 3, 1998. The amendments incorporate the Distribution Ignition System requirements into the Unit 1 and Unit 2 TSs.

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 242 to DPR-58
2. Amendment No. 223 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

Donald C. Cook Nuclear Plant, Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 242
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 December 3, 1998, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 242 , are hereby incorporated in the 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 15, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 242

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
VII	VII
3/4 6-24	3/4 6-24
3/4 6-25	3/4 6-25
B 3/4 6-3	B 3/4 6-3
	B 3/5 6-3a

SECTION PAGE

ELECTRIC HYDROGEN RECOMBINERS - W

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 18 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to $\geq$ 700°F within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.)

3. Verifying during a recombiner system functional test that the heater sheath temperature increases to $\geq$ 1200°F within 5 hours and is maintained for at least 4 hours.

4. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

DISTRIBUTED IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Both trains of the Distributed Ignition System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one train of the Distributed Ignition System inoperable:

a. Restore the inoperable train to OPERABLE status within 7 days, or

b. Perform surveillance requirement 4.6.4.3a once per 7 days on the OPERABLE train until the inoperable train is restored to OPERABLE status.

With no OPERABLE hydrogen igniter in one containment region, restore one hydrogen igniter in the affected containment region to OPERABLE status within 7 days, or be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each train of the Distributed Ignition System shall be demonstrated OPERABLE:

a. Once per 92 days by energizing the supply breakers and verifying that at least 34 of 35 igniters are energized.

b. Once per 92 days, by verifying at least one hydrogen igniter is OPERABLE in each containment region.

c. Once per 18 months by verifying the temperature of each igniter is a minimum 1700°F.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

Surveillance Requirement 4.6.2.2.d is performed by verifying a water flow rate ≥ 20 gpm and ≤ 50 gpm from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure ≥ 255 psig.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of containment purge and exhaust valves and locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing a qualified individual, who is in constant communication with control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

3/4.6.4 COMBUSTIBLE GAS CONTROL

Hydrogen Analyzers and Recombiners

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: 1) zirconium-water reactions; 2) radiolytic decomposition of water; and 3) corrosion of metals within containment.

The acceptance criterion of 10,000 ohms is based on the test being performed with the heater element at an ambient temperature, but can be conservatively applied when the heater element is at a temperature above ambient.

3/4.6.4. COMBUSTIBLE GAS CONTROL (continued)

Distributed Ignition System (DIS)

The DIS permits controlled burning of the excessive hydrogen generated during degraded core LOCAs postulated by 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors." The postulated amount of hydrogen is equivalent to that generated from the reaction of 75% of the fuel cladding with water. Controlled burning at low hydrogen concentrations precludes containment damage that could result from random ignition at high concentrations. An extensive program of testing and analysis has demonstrated that a system of strategically placed hydrogen igniters (the DIS) can be relied upon for controlled burns of the hydrogen gas postulated for degraded cores. Furthermore, it has been shown that this can be accomplished at combustion temperatures and pressures that will not challenge the integrity of the containment structure or the OPERABILITY of containment equipment necessary to shutdown (and maintain shutdown) the reactor.

The hydrogen igniters are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen concentration resulting from a DBA can be maintained less than the flammability limit using the hydrogen recombiners.

The DIS consists of two independent trains of 35 igniters located throughout containment. The igniters in each train are further divided into six groups per train powered from different phases of two separate three phase transformers. It is the transformer phase that uniquely defines a group.

Operation in MODES 1 and 2 with both trains available ensures the capability for controlled burning of hydrogen gas inside containment during degraded core LOCA events.

In MODES 3 and 4 both the hydrogen production rate and the total hydrogen production after a LOCA would be significantly less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the DIS is low. Therefore the DIS is not required in MODES 3 and 4.

In MODES 5 and 6, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the DIS is not required to be OPERABLE in MODES 5 and 6.

The 7 day Completion Time for restoration of an inoperable DIS train in MODES 1 or 2 is based on the low probability of occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding and the low probability of failure of the OPERABLE DIS train. This justification also applies to the 7 day Completion Time allowed for redundant igniters being inoperable in the same containment region. For this case there would also be ignition capability from adjacent containment regions by flame propagation to the region with no OPERABLE igniters.

Confidence in system OPERABILITY is demonstrated by surveillance testing. Since many igniters are inaccessible at power, surveillance testing in MODE 1 is limited to measurement of igniter current when the DIS is energized by groups. Measured currents are compared with baseline data for the group.

Igniter temperature measurement for all igniters can only be performed during shutdown and is performed every 18 months. This testing energizes all igniters and confirms the ability of each igniter to obtain a surface temperature of at least 1700°F. This temperature is conservatively above the temperature necessary to ignite hydrogen mixtures at concentrations near the lower flammability limit. Test experience indicates that individual igniter failures are generally total failures and do not involve the inability to reach the required temperature when an igniter is drawing normal amperage. This observed failure mode provides reasonable confidence that an igniter failing to reach the required temperature would also be detected by reduced group current measurements during the MODE 1 surveillances. Therefore the 18 month frequency for actual temperature measurements is acceptable.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
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 December 3, 1998, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 223 , are hereby incorporated in the 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 15, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 223

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
VII	VII
XII	XII
	3/4 6-34a
B 3/4 6-4	B 3/4 6-4
	B 3/4 6-4a

DISTRIBUTED IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Both trains of the Distributed Ignition System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one train of the Distributed Ignition System inoperable:

a. Restore the inoperable train to OPERABLE status within 7 days, or

b. Perform surveillance requirement 4.6.4.3a once per 7 days on the OPERABLE train until the inoperable train is restored to OPERABLE status.

With no OPERABLE hydrogen igniter in one containment region, restore one hydrogen igniter in the affected containment region to OPERABLE status within 7 days, or be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each train of the Distributed Ignition System shall be demonstrated OPERABLE:

a. Once per 92 days by energizing the supply breakers and verifying that at least 34 of 35 igniters are energized.

b. Once per 92 days, by verifying at least one hydrogen igniter is OPERABLE in each containment region.

c. Once per 18 months by verifying the temperature of each igniter is a minimum 1700°F.

3/4.6.4 COMBUSTIBLE GAS CONTROL

Hydrogen Analyzers and Recombiners

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: 1) zirconium-water reactions; 2) radiolytic decomposition of water; and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The acceptance criterion of 10,000 ohms is based on the test being performed with the heater element at an ambient temperature, but can be conservatively applied when the heater element is at a temperature above ambient.

Distributed Ignition System (DIS)

The DIS permits controlled burning of the excessive hydrogen generated during degraded core LOCAs postulated by 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors." The postulated amount of hydrogen is equivalent to that generated from the reaction of 75% of the fuel cladding with water. Controlled burning at low hydrogen concentrations precludes containment damage that could result from random ignition at high concentrations. An extensive program of testing and analysis has demonstrated that a system of strategically placed hydrogen igniters (the DIS) can be relied upon for controlled burns of the hydrogen gas postulated for degraded cores. Furthermore, it has been shown that this can be accomplished at combustion temperatures and pressures that will not challenge the integrity of the containment structure or the OPERABILITY of containment equipment necessary to shutdown (and maintain shutdown) the reactor.

The hydrogen igniters are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen concentration resulting from a DBA can be maintained less than the flammability limit using the hydrogen recombiners.

The DIS consists of two independent trains of 35 igniters located throughout containment. The igniters in each train are further divided into six groups per train powered from different phases of two separate three phase transformers. It is the transformer phase that uniquely defines a group.

Operation in MODES 1 and 2 with both trains available ensures the capability for controlled burning of hydrogen gas inside containment during degraded core LOCA events.

In MODES 3 and 4 both the hydrogen production rate and the total hydrogen production after a LOCA would be significantly less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the DIS is low. Therefore the DIS is not required in MODES 3 and 4.

In MODES 5 and 6, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the DIS is not required to be OPERABLE in MODES 5 and 6.

The 7 day Completion Time for restoration of an inoperable DIS train in MODES 1 or 2 is based on the low probability of occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding and the low probability of failure of the OPERABLE DIS train. This justification also applies to the 7 day Completion Time allowed for redundant igniters being inoperable in the same containment region. For this case there would also be ignition capability from adjacent containment regions by flame propagation to the region with no OPERABLE igniters.

3/4.6.4 COMBUSTIBLE GAS CONTROL (continued)

Confidence in system OPERABILITY is demonstrated by surveillance testing. Since many igniters are inaccessible at power, surveillance testing in MODE 1 is limited to measurement of igniter current when the DIS is energized by groups. Measured currents are compared with baseline data for the group.

Igniter temperature measurement for all igniters can only be performed during shutdown and is performed every 18 months. This testing energizes all igniters and confirms the ability of each igniter to obtain a surface temperature of at least 1700°F. This temperature is conservatively above the temperature necessary to ignite hydrogen mixtures at concentrations near the lower flammability limit. Test experience indicates that individual igniter failures are generally total failures and do not involve the inability to reach the required temperature when an igniter is drawing normal amperage. This observed failure mode provides reasonable confidence that an igniter failing to reach the required temperature would also be detected by reduced group current measurements during the MODE 1 surveillances. Therefore the 18 month frequency for actual temperature measurements is acceptable.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA, 4) contain sufficient water to maintain adequate sump inventory, and 5) result in a post-LOCA sump pH within the allowed range. These conditions are consistent with the assumptions used in the accident analyses.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a design basis accident and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators.

Over the course of a fuel cycle, sublimation reduces the weight of ice in the ice condenser. For the ice condenser to be considered OPERABLE, the minimum as-found ice weight of 1144 pounds per ice basket, for those ice baskets selected for weighing per the surveillance requirements, must be present at the end of a fuel cycle. An instrument measurement error allowance is included in the required minimum ice basket weight. To account for loss due to sublimation, a conservative average ice bed sublimation of 10% over an eighteen-month period is used. The beginning-of-cycle, or as-left ice basket weight, is adjusted accordingly to assure the LCO limit will be met at the end of each fuel cycle.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the ice bed temperature monitoring system ensures that the capability is available for monitoring the ice temperature. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 242 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated December 3, 1998, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would incorporate limiting conditions of operation, modes of applicability and surveillance requirements for the Distribution Ignition System (DIS) into Units 1 and 2 TSs.

2.0 EVALUATION

10 CFR 50.44, "Standard for Combustible Gas Control System in Light-Water-Cooled Power Reactors," requires the design and installation of systems to mitigate and control the concentration of combustible gas inside containment following a design basis loss of coolant accident (LOCA). To comply with the requirements of 10 CFR 50.44, the licensee has installed a distributed ignition system (DIS). An extensive program of testing and analysis has demonstrated that a system of strategically placed igniters can be relied upon for controlled burns of hydrogen postulated following a design basis accident. The DIS at D. C. Cook Nuclear Plant consists of two independent trains of resistance heating elements (igniters). Each DIS train has 35 igniters. The DIS has been selected by the licensee to comply with 10 CFR 50.44 to mitigate the consequences of the hydrogen generation following a design basis accident. The licensee made several previous submittals to the NRC concerning the design, analysis, and testing of the DIS. The NRC's review and approval of the DIS was documented in a safety evaluation dated December 17, 1981. As such, the system is considered significant to the protection of the public health and safety and meets Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D), for inclusion into the plant TSs.

The proposed amendment adds limiting conditions for operation, applicable modes, surveillance requirements and associated bases for the DIS. The TS requirements proposed by the licensee incorporate the requirements of the improved TSs (NUREG-1431, Revision 1, "Standard Technical Specifications, Westinghouse Plants"). The addition of the DIS system to the TSs will provide additional assurance of the system availability and maintain a greater margin of safety for containment integrity following design basis accidents.

3.0 SUMMARY

Based on the above evaluation, the staff finds that the DIS installed to comply with the requirements of 10 CFR 50.44 meets the requirements of 10 CFR 50.36 for inclusion into the TSs. The proposed TSs have incorporated the requirements of NUREG-1431 for operability and surveillance of the DIS. Therefore, the staff finds the proposed changes to the TSs 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

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (65 FR 4279). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Stang

Date: March 15, 2000

50-315
3/1/2000

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Docket: 05000315
Docket: 05000316

CP-1

March 1, 2000

Template NRR-058

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA7756 AND MA7757)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 241 to Facility Operating License No. DPR-58 and Amendment No. 222 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated December 22, 1999.

The amendments delete Technical Specification (TS) 5.4.2, "Reactor Coolant System Volume," regarding the reactor coolant system (RCS) volume information. This information is not required to be in the TS for compliance with 10 CFR 50.36(c)(4). Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analyses Report (UFSAR), and any changes to the information are controlled in accordance with 10 CFR 50.59.

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

NRC FILE CENTER COPY

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 241 to DPR-58
2. Amendment No. 222 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

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March 1, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA7756 AND MA7757)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 241 to Facility Operating License No. DPR-58 and Amendment No. 222 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated December 22, 1999.

The amendments delete Technical Specification (TS) 5.4.2, "Reactor Coolant System Volume," regarding the reactor coolant system (RCS) volume information. This information is not required to be in the TS for compliance with 10 CFR 50.36(c)(4). Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analyses Report (UFSAR), and any changes to the information are controlled in accordance with 10 CFR 50.59.

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 241 to DPR-58
2. Amendment No. 222 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 1, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
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Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 241 to DPR-58
2. Amendment No. 222 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 241
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 December 22, 1999, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 241, are hereby incorporated in the 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 within 30 days. In addition, the licensee shall include the relocated information to the Updated Final Safety Analysis Report as described in the licensee's application dated December 22, 1999, and evaluated in the staff's safety evaluation dated March 1 , 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 241

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
5-5	5-5

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY – SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water.

b. A nominal 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
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 December 22, 1999, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 222, are hereby incorporated in the 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 within 30 days. In addition, the licensee shall include the relocated information to the Updated Final Safety Analysis Report as described in the licensee's application dated December 22, 1999, and evaluated in the staff's safety evaluation dated March 1, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 222

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
5-5	5-5

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.

5.6 FUEL STORAGE

CRITICALITY – SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 8.97-inch center-to-center distance between fuel assemblies, placed in the storage racks.

c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

   1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

   2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MTU, or fuel of other enrichments with equivalent reactivity.

   3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MTU, or fuel of other enrichments with equivalent reactivity.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 241 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated December 22, 1999, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would delete TS 5.4.2, "Reactor Coolant System Volume," regarding the reactor coolant system (RCS) volume information. Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analyses Report (UFSAR), and any changes to the information are controlled in accordance with 10 CFR 50.59.

2.0 EVALUATION

The nominal RCS volumes currently contained in Unit 1 and Unit 2 TS 5.4.2 do not reflect the actual RCS volumes that will exist when the units are restarted. For Unit 1, replacement of the steam generators during the current outage will result in a small change (less than 2 percent) to total RCS volume. For Unit 2, TS 5.4.2 was not updated to reflect similar small changes to actual RCS volume after replacement of steam generators in 1988. Therefore, the TS 5.4.2 values for RCS volume need to be revised.

The UFSAR includes values for total RCS volume and RCS component and piping volumes that are more detailed and complete than the approximate RCS volumes listed in Unit 1 and Unit 2 TS 5.4.2. These more detailed values are used as design inputs to the actual UFSAR Chapter 14 accident analyses, and include values for RCS volume at previously evaluated steam generator tube plugging limits. Therefore, TS 5.4.2 is redundant to the UFSAR.

10 CFR 50.36(c)(4) governs the contents of Technical Specification (TS) Section 5.0, "Design Features." 10 CFR 50.36(c)(4) states, "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section." Reactor coolant system volume information

does not describe a material of construction for the RCS, nor does it specify a required geometric arrangement for the RCS.

As stated in 10 CFR 50.36(c)(2)(ii)(B), the TS limiting conditions for operation must be established for "process variables, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." TS Section 3/4.4, "Reactor Coolant System," includes the limiting conditions for operation related to the RCS, and includes information either limiting changes to, or derived from, RCS volume. For example, TS 3/4.4.4, "Pressurizer," specifies the minimum allowable water volume in the pressurizer in Modes 1, 2, and 3. In addition, TS 3/4.4.5, "Steam Generators," requires reporting of the number of steam generator tubes plugged following each inservice inspection of the steam generator tubes to the NRC. Therefore, the most significant variables related to RCS volume are already covered in these TS sections, meeting the intent of 10 CFR 50.36(c)(4).

Changes to the actual RCS volume can result from physical modifications to RCS components, changes to procedures affecting pressurizer pressures and levels, or by plugging of steam generator tubes. Changes to the facility and procedures are required to be evaluated in accordance with 10 CFR 50.59, which ensures that changes to RCS volume as a result of physical modifications and procedure changes are evaluated for impact on the plant accident analyses. Steam generator tube plugging limits are evaluated in the UFSAR to ensure acceptability of the limits on the plant accident analyses.

Since detailed RCS information already exists in the UFSAR, and any method by which the RCS volume could be changed is required to be evaluated in accordance with 10 CFR 50.59, then including this information in the TS is not necessary to ensure that a significant effect on safety does not occur. In addition, since TS Section 3/4.4 already includes the limiting conditions for operation related to the RCS, and includes information either limiting changes to, or derived from, RCS volume, then including RCS volume in TS Section 5.0 is not required as allowed by 10 CFR 50.46(c)(4).

The original TS were developed prior to the most recent guidance provided in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants.". NUREG-1431 does not include RCS volume information in TS Section 4, "Design Features," as this information does not meet the criteria for inclusion in the TS, and is not considered necessary for compliance with 10 CFR 50.36(c)(4).

The proposed change to remove this information from TS does not affect any accident initiators or precursors. Elimination of the RCS volume information from the TS does not change the methods for plant operation or actions to be taken in the event of an accident. The deletion of the RCS volume information from the TS does not change the methods of plant operation or modify plant systems, structures, or components. No new methods of plant operation are created. As such, the proposed change does not affect any accident initiators or precursors or create new accident initiators or precursors. The deletion of the RCS volume information from the TS does not affect safety limits or limiting safety system settings. Plant operational parameters are not affected. The proposed change does not modify the quantity of radioactive material available for release in the event of an accident. As such, the proposed change will not affect any previous safety margin assumptions or conditions. The actual volume of the RCS is not affected by the change, only the location of the text describing the volume. More detailed and complete RCS component and piping volume information is included in the UFSAR, and any changes to that information would be evaluated prior to implementation in accordance with 10 CFR 50.59.

The licensee proposes to make administrative changes to the format of the Unit 1 and Unit 2 TS pages in an ongoing effort to improve their appearance. The changes include addition of "5.0 DESIGN FEATURES" to the header, addition of "Page" to the footer, deletion of "NO." from the footer, addition of separating lines at the bottom of the header and the top of the footer, and continuous underlining for the titles of TS Sections 5.5 and 5.6. The staff finds that the proposed administrative changes do not represent a reduction in safety or alter the TS requirements. The administrative changes are intended to maintain consistency and enhance usability and clarity of the TS.

3.0 SUMMARY

Based on the evaluation, the staff finds that proposed TS changes do not reduce the level of safety currently maintained by the TS, is consistent with NUREG-1431, and is in accordance with 10 CFR 50.36. Therefore, the staff finds the proposed changes to the TSs are 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

Pursuant to 10 CFR 51.21, 51.32, and 50.35, an environmental assessment and finding of no significant impact have been prepared and published in the *Federal Register* on March 1 , 2000 (65 FR 11100). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Stang

Date: March 1, 2000

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March 1, 2000

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MEMORANDUM TO: Biweekly Notice Coordinator

/RA/

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA7756 AND MA7757)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: December 22, 1999

Brief description of amendments: The amendments delete Technical Specification 5.4.2, "Reactor Coolant System Volume," regarding the reactor coolant system (RCS) volume information. Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analyses Report (UFSAR), and any changes to the information are controlled in accordance with 10 CFR 50.59.

Date of issuance: March 1, 2000

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 241 and 222

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: January 13, 2000 (65 FR 2199)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No.

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March 1, 2000

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/RA/

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Project Directorate III
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March 1, 2000

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA7756 AND MA7757)

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Date of issuance: March 1, 2000

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No significant hazards consideration comments received: No.

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA7756 AND MA7757)

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Date of application for amendments: December 22, 1999

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Date of issuance:

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Amendment Nos.: 241 and 222

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The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated

No significant hazards consideration comments received: No.

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Docket: 05000316

CP-1

February 25, 2000

ML003685785
Template NRR-042

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNIT 1 AND 2 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT REGARDING REACTOR COOLANT SYSTEM VOLUME INFORMATION (TAC NOS. MA7756 AND MA7757)

Dear Mr. Powers:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your application for amendments dated December 22, 1999. The proposed would delete Technical Specification (TS) 5.4.2, "Reactor Coolant System Volume," regarding the reactor coolant system (RCS) volume information. This information is not required to be in the TS for compliance with 10 CFR 50.36(c)(4). Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analysis Report, and any changes to the information are controlled in accordance with 10 CFR 50.59.

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

Sincerely,
/RA/
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

cc w/encl: See next page

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February 25, 2000

ML0036857785
Template NRR-042

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNIT 1 AND 2 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT REGARDING REACTOR COOLANT SYSTEM VOLUME INFORMATION (TAC NOS. MA7756 AND MA7757)

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Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 25, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNIT 1 AND 2 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT REGARDING REACTOR COOLANT SYSTEM VOLUME INFORMATION (TAC NOS. MA7756 AND MA7757)

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

cc w/encl: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Sherry Kamke, Acting
Environmental Review Coordinator
U.S. EPA Region 4
61 Forsyth Street, SW
Atlanta, GA 30303-3104

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

DONALD C. COOK, UNITS 1 AND 2

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of amendments to Facility Operating License No. DPR-58 and No. DPR-74, issued to the Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook Nuclear Plant (D. C. Cook), Units 1 and 2, located in Berrien County, Michigan.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

The proposed action would delete Technical Specification (TS) 5.4.2, "Reactor Coolant System Volume," regarding the reactor coolant system (RCS) volume information. This information is not required to be in the TS for compliance with 10 CFR 50.36(c)(4). Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analysis Report and any changes to the information are controlled in accordance with 10 CFR 50.59. In addition, format changes are proposed to TS page 5-5 for both Unit 1 and Unit 2.

The proposed action is in accordance with the licensee's application for amendment dated December 22, 1999.

The Need for the Proposed Action:

The proposed action is necessary to correct the plant Technical Specifications. This information is not required to be in the TS for compliance with 10 CFR 50.36(c)(4) and is redundant to information contained in the D. C. Cook Updated Final Safety Analysis Report.

Environmental Impacts of the Proposed Action:

The NRC has completed its evaluation of the proposed action and concludes that the removal of the RCS volume from the TSs and the associated format changes to the TS pages do not impact any other requirements.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any 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 involve 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.

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:

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the D. C. Cook Nuclear Plant.

Agencies and Persons Consulted:

In accordance with its stated policy, on February 18, 2000, the staff consulted with the Michigan State official, Mr. David Minnaar of the Michigan 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 22, 1999, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, N.W., Washington, DC. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http:\\www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 23d day of February 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNIT 1 AND 2 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT REGARDING REACTOR COOLANT SYSTEM VOLUME INFORMATION (TAC NOS. MA7756 AND MA7757)

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

Sincerely,

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Environmental Assessment

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50-315
2/23/2000

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February 23, 2000

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA8183 AND MA8184)

Dear Mr. Powers:

Enclosed is a copy of a "Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," related to your request for license amendments dated February 18, 2000. The proposed license amendments would approve an unreviewed safety question discovered by the licensee during a 10 CFR 50.59 evaluation of modifications to the auxiliary feedwater (AFW) pump rooms to protect the equipment in the rooms from the environmental effects of a postulated high-energy line break. This will be accomplished by sealing the AFW pump rooms to ensure that the rooms do not communicate with the turbine buildings or each other.

This notice has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: Notice

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February 23, 2000

Mr. Robert P. Powers, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA8183 AND MA8184)

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Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 23, 2000

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Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. DPR-58 and DPR-74 issued to Indiana Michigan Power Company (the licensee) for operation of the Donald C. Cook Nuclear Power Plant, Units 1 and 2, located in Berrien County, Michigan.

The proposed amendments would approve an unreviewed safety question discovered by the licensee during a 10 CFR 50.59 evaluation of modifications to the auxiliary feedwater (AFW) pump rooms to protect the equipment in the rooms from the environmental effects of a postulated high-energy line break (HELB). This will be accomplished by sealing the AFW pump rooms to ensure that the rooms do not communicate with the turbine buildings or each other.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   Failures of the proposed MDAFP [motor driven auxilary feedwater pump] and TDAFP [turbine driven auxilary feedwater pump] room cooling systems during either normal operations or emergency operations cannot initiate any of the accidents previously evaluated in the UFSAR. The proposed MDAFP and TDAFP room cooling systems do not interface with the reactor coolant system, containment, or engineered safeguards features in such a way as to be a precursor or initiator for an accident previously evaluated. Therefore, the proposed modifications do not increase the probability of occurrence of an accident previously evaluated.

   The proposed MDAFP and TDAFP room cooling systems ensure protection of AFW equipment from the environmental effects of a HELB event. This ensures the AFW system is capable of performing the safety-related functions required to mitigate the effects of design basis accidents. The AFW system is required to mitigate design basis accidents that result in the loss of cooling for the reactor coolant system. These include loss of normal feedwater control, loss of all (non-emergency) alternating-current power (i.e., offsite power) to the plant auxiliaries, steam generator tube rupture, large break loss-of-coolant accidents, and small break loss-of-coolant accidents. In addition, the AFW system is required to safely shutdown the reactor following certain HELB events in the turbine buildings resulting from feedwater and main steam piping breaks and critical cracks. Since the AFW system is assured of performing its intended design function in mitigating the effects of design basis accidents by the proposed modifications, the consequences of accidents previously evaluated in the UFSAR will not be increased.

   Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Failures of the proposed MDAFP and TDAFP room cooling systems during either normal operations or emergency operations cannot initiate an accident. The proposed MDAFP and TDAFP room cooling systems do not interface with the reactor coolant system, containment, or engineered safeguards features in such a way as to be a precursor or initiator for an accident.

The proposed modifications to the AFW pump rooms have been designed to ensure that the train failure scenarios and design basis accident mitigation functions for AFW are preserved as described in the CNP [Cook Nuclear Plant] UFSAR. The electrical power supplies and AFW pump room cooler water sources maintain the design basis train alignments. Thus, when postulated design basis accident scenarios and single failures are applied to the proposed AFW pump room modification configurations, the AFW system remains bounded by the accident analysis presented in the UFSAR. The modifications do not impact how the AFW system will actuate and perform in response to those design basis accident scenarios that require AFW to mitigate the events.

Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed modifications to the MDAFP and TDAFP room ventilation systems do not create a reduction in the margin of safety for those systems, structures, and components required for safe shutdown or accident mitigation as previously analyzed in the UFSAR. The proposed modifications provide a different method for cooling the AFW pump rooms while ensuring environmental protection to each MDAFP and each TDAFP from the effects of postulated HELB events.

As discussed above, the proposed modifications to the AFW pump rooms have been designed to ensure that the train failure scenarios and design basis accident mitigation functions for AFW are preserved as described in the CNP UFSAR. Since the intended safety function of the AFW pump room cooling systems remains the same, margin of safety is preserved. The proposed modifications ensure the availability and reliability of the AFW pumps is maintained commensurate with the assumptions made in the UFSAR accident analyses.

Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92 are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By March 27, 2000 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington,

DC. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is

aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General

Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to David. W. Jenkins, Esq., American Electric Power, Nuclear Generation Group, One Cook Place, Bridgman, MI 49106, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(I)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated February 18, 2000, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L-Street, NW., Washington, DC. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http:www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 18th day of February 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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Subject:
D. C. Cook Units 1&2 Notice of Issuance of Amendments Spray Additive System

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

January 19, 2000 ML003653677

Template : NRR-106

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1 /RA/
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6885 AND MA6887)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: October 12, 1999

Brief description of amendments: The amendments revised the Technical Specification (TSs) Surveillance Requirement 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the associated TSs Bases. Additionally, certain administrative text format changes were made.

Date of issuance: January 19, 2000

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 240 and 221

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: November 3, 1999 (64 FR 59804)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 19, 2000.

No significant hazards consideration comments received: No.

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January 19, 2000

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1 /RA/
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6885 AND MA6887)

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Date of application for amendments: October 12, 1999

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
January 19, 2000

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6885 AND MA6887)

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No significant hazards consideration comments received: No.

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ISSUANCE OF AMENDMENTS 240 and 221 to licenses DPR-58 and DPR-74, for DONALD C. COOK, UNITS 1 AND 2 (TAC NOS.

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

V.

January 19, 2000

Template
NRR-058

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA6885 AND MA6886)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 240 to Facility Operating License No. DPR-58 and Amendment No. 221 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 12, 1999.

The amendments would revise TSs Surveillance Requirement 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the associated TSs Bases. Additionally, certain administrative text format changes are proposed.

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 240 to DPR-58
2. Amendment No. 221 to DPR-74
3. Safety Evaluation

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January 19, 2000

Mr. Robert P. Powers, Se[illegible]Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA6885 AND MA6886)

Dear Mr. Powers:

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Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 240 to DPR-58
2. Amendment No. 221 to DPR-74
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 19, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA6885 AND MA6886)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 240 to Facility Operating License No. DPR-58 and Amendment No. 221 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 12, 1999.

The amendments would revise TSs Surveillance Requirement 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the associated TSs Bases. Additionally, certain administrative text format changes are proposed.

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 240 to DPR-58
2. Amendment No. 221 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

Donald C. Cook Nuclear Plant, Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 240
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 October 12, 1999, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 240, are hereby incorporated in the 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 at the facility within 30 days. In addition, the licensee shall include the relocated information in the bases of the Technical Specifications as described in the licensee's application dated October 12, 1999, and evaluated in the staff's safety evaluation dated January 19, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: January 19, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 240

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 6-13	3/4 6-13
B3/4 6-3	B3/4 6-3

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure -- High-High signal.

d. At least once per 5 years by verifying the flow rate from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

Surveillance Requirement 4.6.2.2.d is performed by verifying a water flow rate ≥ 20 gpm and ≤ 50 gpm from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure ≥ 255 psig.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of containment purge and exhaust valves and locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing a qualified individual, who is in constant communication with control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: 1) zirconium-water reactions; 2) radiolytic decomposition of water; and 3) corrosion of metals within containment.

The acceptance criterion of 10,000 ohms is based on the test being performed with the heater element at an ambient temperature, but can be conservatively applied when the heater element is at a temperature above ambient.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 221
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 October 12, 1999, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 221, are hereby incorporated in the 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 at the facility within 30 days. In addition, the licensee shall include the relocated information in the bases of the Technical Specifications as described in the licensee's application dated October 12, 1999, and evaluated in the staff's safety evaluation dated January 19, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: January 19, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 221

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 6-12	3/4 6-12
B3/4 6-3	B3/4 6-3

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.†

d. At least once per 5 years by verifying the flow rate from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation.

†The provisions of Technical Specification 4.0.8 are applicable.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge location or other physical characteristics.

Surveillance Requirement 4.6.2.2.d is performed by verifying a water flow rate $\geq$ 20 gpm and $\leq$ 50 gpm from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure $\geq$ 255 psig.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of containment purge and exhaust valves and locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing a qualified individual, who is in constant communication with control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 240 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 221 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated October 12, 1999, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would revise TS Surveillance Requirement (SR) 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the associated TSs Bases. Additionally, the proposed amendments will make administrative text format changes to the TSs.

2.0 EVALUATION

2.1 Proposed Changes to Unit 1 and 2 TS Surveillance Requirement 4.6.2.2.d, and TS Bases 3/4 4.6.2.2

Current TSs SR 4.6.2.2.d includes details regarding the required flow rate and test parameters for the spray additive system (details of the acceptance criteria and test parameter for the flow rate verification surveillance). The current Unit 1 and 2 TS SR 4.6.2.2.d states that "At least once per 5 years by verifying a water flow rate of at least 20 gallons per minute (gpm) (greater than or equal to 20 gpm) but not to exceed 50 gpm (less than or equal to 50 gpm) from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure greater than or equal to 255 psig." The licensee proposes to change the TSs SR to state "At least once per 5 years by verifying the flow rate from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation." The licensee states that the test parameters and the acceptance criteria (i.e., a flow rate between 20 gpm and 50 gpm with a pump discharge pressure of 250 psig) in TSs SR 4.6.2.2.d are proposed to be relocated to the TS Bases 3/4 4.6.2.2. Therefore, the licensee proposes to add the following sentence to Unit 1 and 2 TS Bases 3/4.6.2.2; "SR 4.6.2.2.d is performed by verifying a water flow rate of greater or equal to 20 gpm and less than or equal to 50 gpm from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure greater or equal to 255 psig." The above details currently contained in SR

4.6.2.2.d are not necessary to ensure the operability of the spray additive system. The current TS requirements contained in the Limiting Condition for Operation (LCO) 3.6.22, "Spray Additive System," and the proposed SR are adequate to ensure the spray additive system is operable and can perform its intended function.

TS 4.6.2.2.d contain the containment spray additive system eductor testing parameters and acceptance limits. The limiting conditions for operation for the Containment Spray Additive System specify the system shall be operable. These testing parameters specify the condition at which the testing is performed and the acceptance limits are the acceptance criteria for the spray eductor testing performed to satisfy the surveillance requirements in TS 4.6.2.2.d. This surveillance ensures that the spray eductor's performance is consistent with the assumption for the safety analyses performed for design basis accidents and transients. The changes involve only the relocation of the details associated with the containment spray additive system's eductor testing parameters and acceptance limits but retain the surveillance requirement to perform containment spray additive system eductor testing. The TS Bases will now contain the eductor testing parameters and acceptance limits for the required spray eductor surveillance.

Although the testing parameters and acceptance limits are relocated from the TS to the TS Bases, the licensee must continue to evaluate any changes to testing requirements in accordance with 10 CFR 50.59. Should the licensee's determination conclude that an unreviewed safety question is involved, due to either (1) an increase in the probability or consequences of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change.

The staff has reviewed the testing parameters and acceptance limits associated with the containment spray pump proposed to be relocated from the TS to the TS bases against the criteria of 10 CFR 50.36(c)(2)(ii) and determined that none of the criteria applies as discussed below:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The surveillance requirement acceptance criteria and the test parameter are not instrumentation and do not affect instrumentation. Therefore, they are not instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The surveillance requirement acceptance criteria and the test parameter are not process variables, design features, or operating restrictions. The surveillance requirement acceptance criteria and the test parameter do not affect process variables, design features, or operating restrictions. Therefore, they are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or

transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The surveillance requirement acceptance criteria and the test parameter are not structures, systems, or components. The retained surveillance requirement still ensures the containment spray eductors are capable of performing their safety functions to mitigate design basis accidents. Therefore, they are not structures, systems, or components that are part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The surveillance requirement acceptance criteria and the test parameter are not structures, systems, or components. Therefore, they are not structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Existing TS requirements which fall within or satisfy any of the above criteria must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

The design details do not meet the criteria of 10 CFR 50.36(c)(2)(ii) as being required to be included in the TS, and therefore the staff finds the testing parameters and acceptance limits associated with the containment spray pumps may be relocated from the TS to owner controlled documents. These details will be relocated to the TS bases and all changes will be controlled pursuant to 10 CFR 50.59.

The relocation of the above details from the SR to the TS Bases does not eliminate current TS requirements associated with the required testing of the spray additive system, and will not affect the system operability. In addition, the relocation of the details associated with the acceptance criteria and test parameters to the associated TS Bases is consistent with NUREG-1431 "Standard Technical Specifications, Westinghouse Plants, Specifications," Revision 1.

The NRC staff finds that the proposed changes do not constitute a reduction in safety and do not alter the requirement of TS 4.6.6.2.d. The proposed changes are intended to allow for a more complete and accurate testing of the containment spray pump, and timely revision of the parameters upon any future changes in the analyses and calculations associated with the spray additive system. This is consistent with NUREG-1431. Therefore, the staff finds the proposed changes acceptable.

2.2 Proposed Editorial Changes to Unit 1 TS Page 3/4 6-13, Unit 1 TS Bases Page B 3/4 6-3, Unit 2 TS Page 3/4 6-12, and Unit 2 TS Bases Page B 3/4 6-3

The licensee is proposing certain format changes to Unit 1 TS page 3/4 6-13, Unit 1 TS Bases page B 3/4 6-3, Unit 2 TS page 3/4 6-12, and Unit 2 TS Bases page B 3/4 6-3 to correct minor differences in margins and text spacing due to variations in word processing and reprographic technologies. In addition, there are also specific format changes affecting the Unit 2 Bases page B 3/4 6-3. These specific changes include (1) the use of a different font which also results in altered spacing of the text on the page and content for each line of text, (2) the use of horizontal bars to separate the footer and header from the body of the page, (3) the addition of numerical annotation (i.e., 3/4 and 3/4.6 in the header text lines), (4) the removal of underlining from the two lines of header text, (5) the reversal of sequence and deletion of a blank line between the two lines of header text, (6) the removal of spaces immediately preceding and following the hyphen in the footer text, "COOK NUCLEAR PLANT-UNIT 2," (7) the addition of the word "Page" prior to the page number, and (8) the removal of the word "NO." following the word "AMENDMENT" and prior to the historical and current amendment numbers.

The staff finds that the proposed editorial changes do not represent a reduction in safety or alter any requirement. The editorial changes are intended to maintain consistency and enhance usability and clarity of the TS. Therefore, the proposed changes are acceptable.

3.0 SUMMARY

The proposed amendment would revise TS SR 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the appropriate TS Bases. Additionally, the proposed amendment allows for administrative text format changes. The proposed amendment does not cause changes to accident initiators or precursors, or to the accident analyses, and does not involve a significant reduction of safety.

Based on the above evaluation, the staff finds that the proposed changes to the TSs are 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

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 59804). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: K. Leigh
J. Stang

Date: January 19, 2000

1/13/2000

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Docket: 05000315
Docket: 05000316

CP-1

ML003680992
Template = NRR-106

January 13, 2000

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1 (Original Signed By:)
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (REPEAT OF INDIVIDUAL NOTICE) (TAC NOS. MA7756 AND MA7757)

Indiana Michigan Power Company, Docket, Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: December 22, 1999

Brief description of amendments: The amendments would delete the Donald C. Cook (D.C. Cook), Unit 1 and 2, Technical Specification (TS) 5.4.2, "Reactor Coolant System Volume," because the information regarding the reactor coolant system (RCS) is not required by TS Section 5.0, "Design Features," for compliance with 10 CFR 50.36 (c)(4). Changes to the RCS volume information are included in the D.C. Cook Updated Final Safety Analyses Report, and are controlled in accordance with 10 CFR 50.59.

Date of publication of individual notice in FEDERAL REGISTER: January 13, 1999 (65 FR 2199)

Expiration date of individual notice: February 14, 2000

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Subject:
DONALD C. COOK - CORRECTION TO AMENDMENTS

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

January 12, 2000

Template = NRR-056

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK - CORRECTION TO AMENDMENTS
(TAC NOS. MA6473 AND MA6474)

The U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 236 to Facility Operating License No. DPR-58 and Amendment No. 218 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. Page 2 of the amendments and Page 13 of the Safety Evaluation (SE) were inadvertently dated December 28, 1999, and should have read December 23, 1999. Please replace these pages with the enclosed corrected copy.

We are sorry for any inconvenience this may have caused.

Sincerely,

Original Signed By:

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: As stated

cc w/encls: See next page

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cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

2. Accordingly, the license is amended to authorize revision of the Updated Final Safety Analysis (UFSAR) and Emergency Operating Procedures (EOPs) as set forth in the application for amendment by the licensee, dated September 17, 1999, and as supplemented November 10, 1999, and November 19, 1999, and as evaluated in the staff Safety Evaluation attached to this amendment. The licensee shall update the UFSAR and change the EOPs to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis loss-of-coolant accident as authorized by this license amendment and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications Bases

Date of Issuance: December 23, 1999

2. Accordingly, the license is amended to authorize revision of the Updated Final Safety Analysis (UFSAR) and Emergency Operating Procedures (EOPs) as set forth in the application for amendment by the licensee, dated September 17, 1999, and as supplemented November 10, 1999 and November 19, 1999, and as evaluated in the staff Safety Evaluation attached to this amendment. The licensee shall update the UFSAR and change the EOPs to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis loss-of-coolant accident as authorized by this license amendment and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications Bases

Date of Issuance: December 23, 1999

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 56531). Accordingly, the amendments meet 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 amendments.

7.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: M. Mitchell
J. Rajan
M. Chatterton

Date: December 23, 1999

1/7/2000

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REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATIO N OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED N O SIGNIFICANT HAZARDS CONSIDERATION DETERMINATiON, AND OPPORTUNIT FOR A HEARING (TAC NOS MA4929 AND MA4930)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 7, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Jr., Senior Project Manager, Section 1 CFlyon for
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING (TAC NOS. MA4929 AND MA4930)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: December 3, 1998

Description of amendment requests: The proposed amendments would add a new Technical Specification (T/S) and associated Bases for the distributed ignition system (DIS). The proposed change incorporates the technical requirements of NUREG-1431, Revision 1, "Standard Technical Specifications, Westinghouse Plants."

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated. The amendments involve new requirements for the T/Ss and do not delete any existing requirements.

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1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated. The amendments involve new requirements for the T/Ss and do not delete any existing requirements.

2. The proposed amendment will not create the possibility of a new or different kind of accident previously evaluated.

   The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety.

   The T/S being proposed for the DIS is consistent with [the] design and operation as previously reviewed and approved, and therefore, does not involve a significant reduction in a margin of safety. Compliance with the proposed T/S will provide additional assurance of system availability to maintain a margin of safety for containment integrity during degraded core events.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

<u>Attorney for licensee</u>: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

<u>NRC Section Chief</u>: Claudia M. Craig

2. The proposed amendment will not create the possibility of a new or different kind of accident previously evaluated.

   The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety.

   The T/S being proposed for the DIS is consistent with [the] design and operation as previously reviewed and approved, and therefore, does not involve a significant reduction in a margin of safety. Compliance with the proposed T/S will provide additional assurance of system availability to maintain a margin of safety for containment integrity during degraded core events.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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2. The proposed amendment will not create the possibility of a new or different kind of accident previously evaluated.

   The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety.

   The T/S being proposed for the DIS is consistent with [the] design and operation as previously reviewed and approved, and therefore, does not involve a significant reduction in a margin of safety. Compliance with the proposed T/S will provide additional assurance of system availability to maintain a margin of safety for containment integrity during degraded core events.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: Claudia M. Craig

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Subject:
REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA7041 AND MA704 2)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

MEMORANDUM TO: [illegible]eekly Notice Coordinator January [illegible] 2000

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA7041 AND MA7042)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: November 3, 1999

Brief description of amendments: The amendments allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

Date of issuance: January 6, 2000

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 239 and 220

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: December 1, 1999 (64 FR 67335)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 6, 2000.

No significant hazards consideration comments received: No

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 6, 2000

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA7041 AND MA7042)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: November 3, 1999

Brief description of amendments: The amendments allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

Date of issuance: January 6, 2000

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 239 and 220

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: December 1, 1999 (64 FR 67335)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 6, 2000.

No significant hazards consideration comments received: No.

1/6/2000

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Subject:
ISSUANCE OF AMENDMENTS 239 and 220 to licenses DPR-58 and DPR-74, for COOK, UNITS 1 AND 2, RE: FUEL ROD ZIRLO CLADDING AND INTEGRAL FUEL BURNABLE ABSORBER REQUIREMENTS.

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

January 6, 2000

Template No. 058

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2, RE: FUEL ROD ZIRLO CLADDING AND INTEGRAL FUEL BURNABLE ABSORBER REQUIREMENTS (TAC NOS. MA7041 AND MA7042)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 239 to Facility Operating License No. DPR-58 and Amendment No. 220 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated November 3, 1999.

The amendments allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

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

Sincerely,
Original Signed By
Carl F. Lyon
for John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 239 to DPR-58
2. Amendment No. 220 to DPR-74
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 6, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2, RE: FUEL ROD ZIRLO CLADDING AND INTEGRAL FUEL BURNABLE ABSORBER REQUIREMENTS (TAC NOS. MA7041 AND MA7042)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 239 to Facility Operating License No. DPR-58 and Amendment No. 220 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated November 3, 1999.

The amendments allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

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

Sincerely,

CFLyon for

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 239 to DPR-58
2. Amendment No. 220 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 239
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 November 3, 1999, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 239 , are hereby incorporated in the 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 within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: January 6, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 239

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
5-4	5-4
5-6	5-6
5-8	5-8
-	5-8a

5.2 CONTAINMENT (Continued)

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 5.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4 or ZIRLO, except that limited substitutions of zirconium alloy or stainless steel filler rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analysis to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 4.95 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.6 FUEL STORAGE (Continued)

1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations and graphically depicted in Figure 5.6-3.

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

5.6 FUEL STORAGE (Continued)

CRITICALITY - NEW FUEL

5.6.2 The new fuel storage racks are designed and shall be maintained with:

a. Westinghouse fuel assemblies having either a maximum enrichment of 4.55 weight % U-235, or an enrichment between 4.55 and 4.95 weight % U-235 with greater than or equal to the minimum number of integral fuel burnable absorber pins as shown on Figure 5.6-4(interpolation of the Boron-10 loading between 1.0X and 1.5X and between 1.5X and 2.0X is acceptable);

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR;

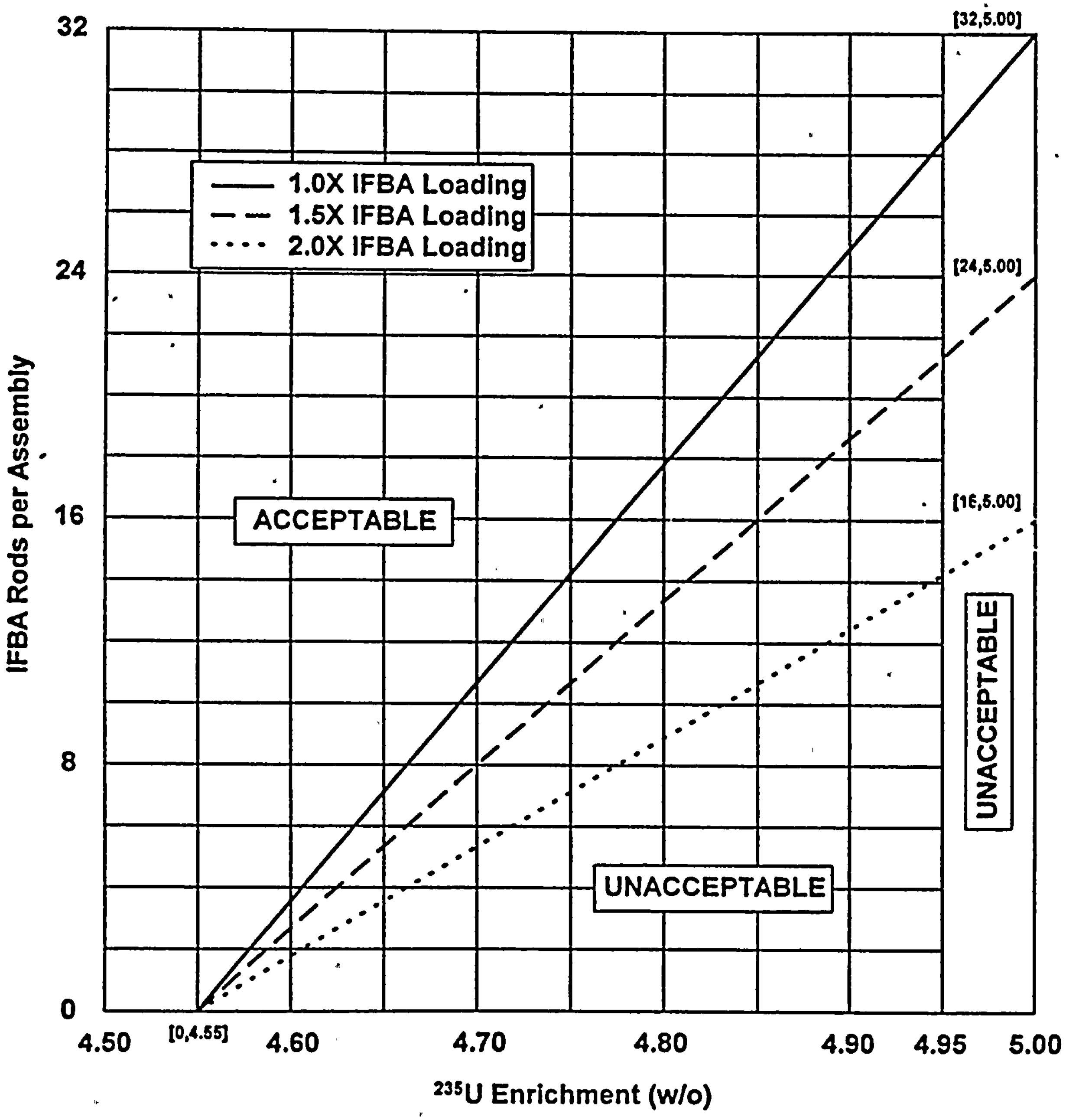
c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

Figure 5.6-4: New Fuel Storage Rack Integral Fuel Burnable Absorber (IFBA) Requirements



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
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, 1999, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 220 , are hereby incorporated in the 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 within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: January 6, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
5-4	5-4
5-6	5-6
5-9	5-9
-	5-9a

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO, except that limited substitutions of zirconium alloy or stainless steel filler rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.3 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and may be nominally enriched up to 4.95 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

5.6 FUEL STORAGE (Continued)

CRITICALITY - SPENT FUEL (Continued)

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations and graphically depicted in Figure 5.6-3.

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

5.6 FUEL STORAGE (Continued)

CRITICALITY - NEW FUEL

5.6.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having either a maximum enrichment of 4.55 weight % U-235, or an enrichment between 4.55 and 4.95 weight % U-235 with the minimum number of integral fuel burnable absorber pins as shown on Figure 5.6-4 (interpolation of the Boron-10 loading between 1.0X and 1.5X and between 1.5X and 2.0X is acceptable);

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.7 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

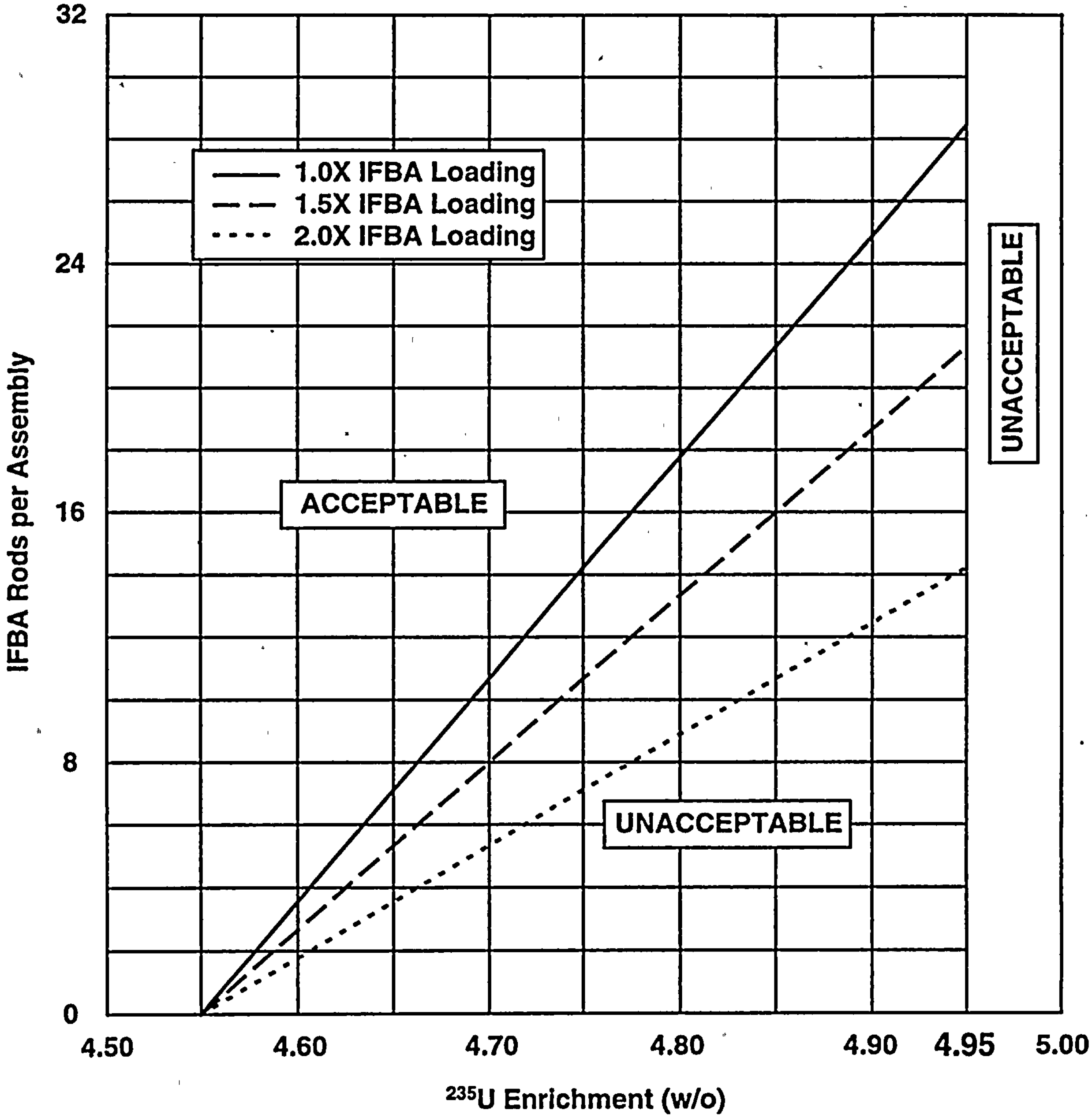
DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3613 fuel assemblies.

Figure 5.6-4: New Fuel Storage Rack Integral Fuel Burnable Absorber (IFBA) Requirements



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 239 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated November 3, 1999, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The proposed amendments would allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

2.0 EVALUATION

2.1 USE OF FUEL RODS WITH ZIRLO CLADDING

TS 5.3.1 requires, in part, that each fuel assembly shall consist of fuel rods clad with Zircaloy-4. Limited substitutions of zirconium alloy or stainless steel filler rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analysis to comply with all fuel safety design bases. The licensee proposes to change TS 5.3.1 to allow fuel rods to be clad with either Zircaloy-4 or ZIRLO. The licensee is transitioning to a more advanced Westinghouse fuel assembly design consisting of the VANTAGE 5 fuel assembly design currently used at CNP with the addition of ZIRLO cladding. The new fuel assembly design has become the standard fuel design. It is used at many other Westinghouse plants and provides improved corrosion resistance, enhanced fuel reliability, and the capability to support future increased discharge burnups.

The use of ZIRLO cladding in Westinghouse fuel was described in Westinghouse Topical Report WCAP-12610, "VANTAGE+ Fuel Assembly Reference Core Report," and was approved by the staff for irradiation up to 60,000 MWD/MTU rod average burnup in a safety evaluation transmitted by letter from A. Thadani (NRC) to S. Tritch (Westinghouse) dated July 1, 1991. The safety evaluation concluded that:

a. The mechanical design bases and limits for ZIRLO clad fuel assembly design are the same as those for the previously licensed Zircaloy-4 clad fuel assembly design, except those specified for clad corrosion which are improved.

b. The neutronic evaluations have shown that ZIRLO clad fuel nuclear design bases are satisfied and that key safety parameter limits are applicable. The nuclear design models and methods accurately describe the behavior of ZIRLO clad fuel.

c. The thermal and hydraulic design bases for ZIRLO clad fuel is unchanged from those of fuel clad with Zircaloy-4.

d. The methods and computer codes used in the analysis of the non-LOCA (loss-of-coolant accident) licensing-basis events are valid for ZIRLO clad fuel, and all licensing-basis criteria are met.

e. The large-break LOCA evaluation model was adapted (without effecting model parameters as approved consistent with Appendix K of 10 CFR Part 50) only to reflect the behavior of the ZIRLO clad material during a LOCA. Consequently, the revised evaluation model satisfies 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

In a safety evaluation transmitted by letter from A. Thadani (NRC) to S. Tritch (Westinghouse) dated October 9, 1991, for WCAP-12610, Appendices F and G, the NRC concluded that the LOCA analyses and methods used demonstrated conformance with the criteria given in 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The evaluation stated that its conclusions were based upon the close similarity between the material properties of the ZIRLO alloy of zirconium to those of other zirconium materials that have been previously licensed for use as cladding material. Based on this similarity, the NRC staff found that it is appropriately conservative to apply the criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, when reviewing VANTAGE+ (ZIRLO) fuel applications, including WCAP-12610, Appendices F and G.

The change from Zircaloy-4 to ZIRLO is consistent with 10 CFR 50.44, 10 CFR 50.46, and NUREG-1431, Rev.1, "Standard Technical Specifications - Westinghouse Plants," which includes ZIRLO as an acceptable cladding material. TS 5.3.1 requires, in part, that "Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases." TS 5.3.1 requires the licensee to justify the use of ZIRLO by cycle-specific reload analyses in accordance with NRC-approved applications of fuel rod configurations.

The licensee proposes to use fuel rods with ZIRLO cladding in order to take advantage of improvements in fuel clad corrosion margin and fuel integrity. Due to the similarities in hydraulic, mechanical, and thermal characteristics of the Zircaloy-4 and ZIRLO clad fuel, and since the licensee's proposal is consistent with NUREG-1431, the regulations, and references the staff-approved WCAP-12610, the staff concludes that the use of ZIRLO clad fuel at D.C. Cook is acceptable.

2.2 METHODOLOGY FOR DETERMINING IFBA REQUIREMENTS

The TS design requirements for the new fuel storage racks are intended to ensure that adequate reactivity margin is maintained to prevent an inadvertent criticality. Reactivity margin is maintained by controlling maximum enrichment and spacing of fuel assemblies in the new fuel storage racks. In addition, the use of IFBA is necessary for Westinghouse fuel assemblies with higher base reactivity (high enrichments) to maintain the required reactivity margin.

The current TS requirement for determining the amount of IFBA present in each stored Westinghouse fuel assembly for reactivity control employs the K-infinity (or K_∞) methodology. This methodology and an additional methodology called reactivity equivalencing, based on use of a CNP site-specific IFBA-enrichment curve for Westinghouse fuel assemblies, were both described in CDB-95-175, "Criticality Analysis of the Donald C. Cook Nuclear Plant New Fuel Storage Vault with Credit for Integral Fuel Burnable Absorbers," previously reviewed and approved by the NRC staff for Amendment Nos. 213 and 198 for CNP, Units 1 and 2, respectively, dated February 27, 1997.

The K-infinity methodology uses the reactor core configuration rather than a site-specific new fuel storage rack configuration. A review of this methodology was recently performed by Westinghouse and documented in Nuclear Safety Advisory Letter (NSAL) 99-003, dated February 26, 1999. This review determined that the K-infinity methodology could lead to IFBA requirements that are nonconservative compared to those required by the methodology involving use of an IFBA-enrichment curve as described in CDB-95-175.

The licensee proposes to revise TS 5.6.2.a and add a new TS Figure 5.6-4, "New Fuel Storage Rack Integral Fuel Burnable Absorber (IFBA) Requirements," to specify maximum enrichments and IFBA requirements for Westinghouse fuel in the new fuel storage racks. Specifically, TS 5.6.2.a is proposed to state that the new fuel storage racks are designed and shall be maintained with "Westinghouse fuel assemblies having either a maximum enrichment of 4.55 weight % U-235, or an enrichment between 4.55 and 4.95 weight % U-235 with greater than or equal to the minimum number of integral fuel burnable absorber pins as shown on Figure 5.6-4 (interpolation of the Boron-10 loading between 1.0X and 1.5X and between 1.5X and 2.0X is acceptable)." The new TS Figure 5.6-4 specifically covers IFBA requirements between 4.55 and 4.95 weight % U-235. The licensee proposes to delete the footnote for TS Table 5.6-1 since it is superseded by the proposed change to TS 5.6.2.a.

New TS Figure 5.6-4 is based on the reactivity equivalencing calculational method described in CDB-95-175, which was previously reviewed and approved by the NRC for Amendment Nos. 213 and 198. Use of this method ensures the required reactivity margin is maintained for storage of Westinghouse fuel assemblies in the new fuel storage racks. Therefore, the staff considers the proposed method for determining IFBA requirements and resulting TS changes acceptable.

2.3 DELETION OF SPECIFIED STORED FUEL ASSEMBLY TYPES

TS 5.6.1.2 and TS Table 5.6-1 list the specific fuel assembly types allowed in the spent fuel storage racks and the specific fuel assembly types allowed in the new fuel storage racks. These include Westinghouse and Exxon/ANF fuel designs.

The design and operational requirements for the spent fuel storage racks and new fuel storage racks are intended to ensure that adequate reactivity margin is maintained to prevent an inadvertent criticality. Reactivity margin is maintained by controlling maximum enrichment, overall reactivity of the stored fuel assemblies, and spacing of fuel assemblies in the spent fuel storage racks and new fuel storage racks. The listing of specific fuel assembly types and their maximum nominal enrichment illustrates the different specific fuel assembly designs that have been determined to meet the design requirements for storage to ensure the reactivity margin requirements of the TS are maintained.

The licensee proposes to delete the specific fuel assembly types from the TS, specifically TS 5.6.1.2 and TS Table 5.6-1, to eliminate the need to revise the TS in the future for changes in specific fuel assembly types that otherwise do not affect TS requirements or require NRC review and approval under 10 CFR 50.59. The current criticality requirements specified in TS 5.6.1.1 and TS 5.6.2 for the spent fuel storage racks and new fuel storage racks remain unchanged.

Identifying specific fuel assembly types is not necessary in the TS, because the maximum enrichment and criticality requirements in TS 5.6.1.1 and TS 5.6.2 ensure the safety of stored fuel assemblies. In addition, changes in specific fuel assembly types must be justified by cycle-specific reload analyses in accordance with 10 CFR 50.59. This proposed change is consistent with both NUREG-0452, Rev.4, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," and NUREG-1431, Rev.1, "Standard Technical Specifications - Westinghouse Plants." Therefore, the staff considers the proposed deletion of specific fuel assembly types acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 67335). Accordingly, the amendments meet 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 amendments.

5.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Lyon

Date: January 6, 2000

12/28/99

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From: Stefanie Fountain

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Subject:
REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF A MENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6473 AND MA6474)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

07-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 28, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6473 AND MA6474)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: September 17, 1999, as supplemented November 10 and 19, 1999

Brief description of amendments: The amendments would approve the licensee's revision of the Updated Final Safety Analysis Report and Emergency Operating Procedures to use methodology to credit the negative reactivity provided by insertion of the rod cluster control assemblies (RCCAs) into the reactor core following any design basis loss-of-coolant accident, during realignment from a cold leg recirculation to a hot leg recirculation configuration. This change to the licensing basis, when evaluated by the licensee in accordance with 10 CFR 59.59, resulted in an unreviewed safety question that requires prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90 prior to implementation. The amendments also change the Bases for TS Section 3/4.5.5, Refueling Water Storage Tank.

Date of issuance: December 28, 1999

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 236 and 218

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

DFX2

<u>Date of initial notice in FEDERAL REGISTER</u>: October 20, 1999 (64 FR 56531)

The licensee's letters of November 10 and 19, 1999, provided additional information that did not change scope of the application or the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 28, 1999.

No significant hazards consideration comments received: No.

Date of initial notice in FEDERAL REGISTER: October 20, 1999 (64 FR 56531)

The licensee's letters of November 10 and 19, 1999, provided additional information that did not change scope of the application or the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 28, 1999

No significant hazards consideration comments received: No.

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December 23, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MA7049 AND MA7050)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-58 and Amendment No. 219 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendment consists of changes to Appendix A, Technical Specifications (TSs), in response to your application dated November 5, 1999.

The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect the monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

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

Sincerely,

ORIGINAL SIGNED BY:
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 237 to DPR-58
2. Amendment No. 219 to DPR-74
3. Safety Evaluation

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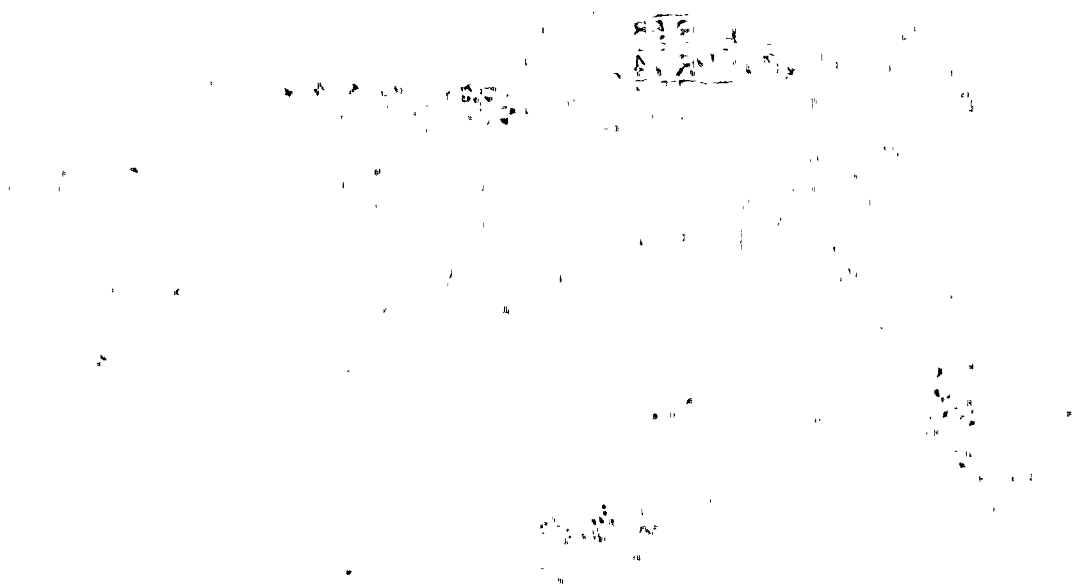

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MA7049 AND MA7050)

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The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect the monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

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Sincerely,

John F. Stang, Project Manager
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. to DPR-58
2. Amendment No. to DPR-74
3. Safety Evaluation Report

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Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MA7049 AND MA7050)

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The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect the monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

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Sincerely,

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. to DPR-58
2. Amendment No. to DPR-74
3. Safety Evaluation

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Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MA7049 AND MA7050)

Dear Mr. Powers:

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The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect the monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

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Sincerely,

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. to DPR-58
2. Amendment No. to DPR-74
3. Safety Evaluation

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December 23, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MA7049 AND MA7050)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-58 and Amendment No. 219 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendment consists of changes to Appendix A, Technical Specifications (TSs), in response to your application dated November 5, 1999.

The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect the monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

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Sincerely,

ORIGINAL SIGNED BY:
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
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2. Amendment No. 219 to DPR-74
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 23, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MA7049 AND MA7050)

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The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect the monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

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Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 237 to DPR-58
2. Amendment No. 219 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (NRC) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 5, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amendment (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 Commissions regulations and all applicable requirements have been satisfied.

ML003670812

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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 237, are hereby incorporated in the 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 23, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 237

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 5-1	3/4 5-1
3/4 5-2	3/4 5-2
B 3/4 5-1	B 3/4 5-1

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 921 and 971 cubic feet,

c. A boron concentration between 2400 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 585 and 658 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, due to boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least Mode 3 within the next 6 hours and reduce reactor coolant system pressure to less than or equal to 1000 psig within the following 6 hours.

b. With one accumulator inoperable for reasons other than boron concentration not within limits, restore the accumulator to OPERABLE status within 1 hour, or be in at least Mode 3 within the next 6 hours and reduce reactor coolant system pressure to less than or equal to 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Reactor Coolant System Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and, for the affected accumulator(s), within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume (that is not the result of addition from the refueling water storage tank) by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig, by verifying that power is removed from each accumulator isolation valve operator.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Standard 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required. Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is greater than 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor-operated isolation valve.

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour completion time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The completion time minimizes the potential for exposure of the plant to a LOCA under these conditions.

If the accumulator cannot be returned to OPERABLE status within the associated completion time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to $\leq$ 1000 psig within 12 hours. The

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 219
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (NRC) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 5, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amendment (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 Commissions 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 219, are hereby incorporated in the 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 23, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 219

TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 5-1	3/4 5-1
3/4 5-2	3/4 5-2
B 3/4 5-1	B 3/4 5-1

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 921 and 971 cubic feet,

c. A boron concentration between 2400 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 585 and 658 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable due to boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least Mode 3 within the next 6 hours and reduce reactor coolant system pressure to less than or equal to 1000 psig within the following 6 hours.

b. With one accumulator inoperable for reasons other than boron concentration not within limits, restore the accumulator to OPERABLE status within 1 hour, or be in at least Mode 3 within the next 6 hours and reduce reactor coolant system pressure to less than or equal to 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Reactor Coolant System Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b.. At least once per 31 days and, for the affected accumulator(s), within 6 hours after each solution volume increase greater than or equal to 1% of tank volume (that is not the result of addition from the refueling water storage tank) by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that power is removed from each accumulator isolation valve operator.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Standard 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required. Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is greater than 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve.

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour completion time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The completion time minimizes the potential for exposure of the plant to a LOCA under these conditions.

If the accumulator cannot be returned to OPERABLE status within the associated completion time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 219 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated November 5, 1999, the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2 (DC Cook). The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect monitoring pressure data in the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is physically removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

2.0 EVALUATION

2.1 Proposed Changes to Unit 1 and 2 TS Requirement 3.5.1, Action "a" and "b"

Unit 1 and 2 TS 3.5.1, Action "a" and "b," describe credible plant operation limits under conditions of one inoperable accumulator. Unit 1 and 2 TS 3.5.1, Action "a," currently states "With one accumulator inoperable, due to boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least MODE 3 within the next 6 hours and reduce pressurizer pressure to less than or equal to 1000 psig within the following 6 hours." The licensee proposes to replace the word "pressurizer" with the phrase "reactor coolant system." The licensee proposes the TS to read as "With one accumulator inoperable, due to boron concentration not within limits, restore boron concentration to within limits within 72 hours or be in at least MODE 3 within the next 6 hours and reduce *reactor coolant system* pressure to less than or equal to 1000 psig within the following 6 hours."

Unit 1 and 2 TS 3.5.1, Action "b," currently states "With one accumulator inoperable for reasons other than boron concentration not within limits, restore the accumulator to OPERABLE status within 1 hour, or be in at least MODE 3 within the next 6 hours and reduce pressurizer pressure

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to less than or equal to 1000 psig within the following 6 hours." The licensee proposes to also replace the word "pressurizer" with the phrase "reactor coolant system." The licensee proposes the TS to read as "With one accumulator inoperable for reasons other than boron concentration not within limits, restore the accumulator to OPERABLE status within 1 hour, or be in at least MODE 3 within the next 6 hours and reduce *reactor coolant system* pressure to less than or equal to 1000 psig within the following 6 hours."

The licensee also proposes to modify a footnote that references the applicability of Unit 1 and 2 TS 3.5.1 to MODES 1, 2, and 3. The footnote currently states "Pressurizer Pressure above 1000 psig." Again, the licensee proposes to replace the word "pressurizer" with "reactor coolant system." The footnote is proposed to read as "*Reactor Coolant System* Pressure above 1000 psig."

The purpose of Unit 1 and 2 TS 3.5.1 is to ensure that the accumulators are operated under the correct operating conditions. Accumulators are large tanks filled with borated water that inject coolant in to the Reactor Coolant System (RCS) when RCS conditions are less than 600 psig (i.e., during a loss-of-coolant accident at low pressures). Action "a" and "b" require reducing pressurizer pressure to less than or equal to 1000 psig within 6 hours if an accumulator becomes inoperable. The purpose of the Unit 1 and 2 footnote is to clarify the mode applicability of the TS.

The licensee proposes to change the word "pressurizer" to the phrase "reactor coolant system" in TS 3.5.1, Actions "a" and "b." The change is intended to provide consistency with the plant design. The licensee's present pressurizer pressure instrumentation is calibrated to read at standard operating conditions in the range of 1700 - 2500 psig. When the pressurizer pressure falls below 1700 psig, the instrument scale can no longer show the true reading. The pressure of 1500 psig would be off the instrument's scale. There is no other pressurizer pressure instrumentation calibrated for lower pressure levels. Therefore, the licensee proposes to use the RCS pressure indicators, which are calibrated to a range of 0 - 5000 psig, to meet the conditions of TS 3.5.1. The licensee states that "RCS pressure and pressurizer pressure instrumentation measure a similar parameter in the primary coolant system. Since the RCS is a closed loop fluid system (the pressurizer is connected to the RCS- emphasis added), pressure instruments should indicate approximately the same value. There is no significant difference between the instrument reading because they are corrected for range, height, and accuracy. There is no significant change in the margin of pressure between when the accumulators are required to be aligned at 1000 psig and the upper limit for nitrogen cover-pressure of 658 psig specified in TS 3.5.1.d. Since there is no wide-range pressurizer pressure instrumentation and the pressurizer pressure narrow-range instruments are calibrated for a 1700-2500 psig range, RCS pressure indicators are used for TS SR 4.5.1.c. Using RCS pressure wide-range indicators is acceptable because they have a calibrated range of 0-5000 psig, which provides a more accurate indication of RCS pressure at the TS applicability requirement of 1000 psig."

The NRC staff finds that the proposed change does not constitute a reduction in safety and does not alter the requirement of TS 3.5.1, Action "a" or "b." Monitoring the RCS pressure at low pressure conditions is equivalent to monitoring the pressurizer pressure at low pressure conditions. The proposed change is intended to provide consistency with plant design and therefore allow clarity and consistency in the TS. Furthermore, the change is consistent with

NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, and similar changes have been made at the Byron and Braidwood nuclear facilities. Therefore, the staff finds the proposed change acceptable.

2.2 Proposed Changes to Unit 1 and 2 Surveillance Requirement 4.5.1.c

Unit 1 and 2 TS Surveillance Requirement (SR) 4.5.1.c currently states "At least once per 31 days when the RCS pressure is above 2000 psig, by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit." The licensee proposes to replace the phrase "is disconnected by removal of the breaker from the circuit" with "is removed from each accumulator isolation valve." The licensee proposes the TS to read as *"At least once per 31 days when RCS pressure is above 2000 psig, by verifying that power to the isolation valve operator is removed from each accumulator isolation valve operator."*

The licensee is proposing this change to reflect the actual design of the plant. Unit 1 TS SR 4.5.1.c was revised on March 30, 1976, to reflect guidance in NUREG-0452, "Standard Technical Specifications - Westinghouse Pressurized Water Reactors." However, the licensee has determined that the past procedure did not support literal compliance with the SR in "removal of the breaker from the circuit." The past procedure involved molded-case circuit breakers (MCCB), which supply power to the accumulator isolation valve operators, being placed in the "OFF" position, so that the associated accumulator isolation valves could not operate. However, the MCCB were not designed for ready physical removal, and it was decided that the past procedure could therefore not meet the TS. Additionally, physical removal of the breaker from the circuit in the MCCB is difficult and dangerous. Therefore, the licensee proposes to delete the words "by removal of the breaker from the circuit" in order to allow for power to be removed from the isolation valve operator without physical removal of the circuit breaker. The necessary protection against a single active failure is provided with the control power being removed from the accumulator isolation valve motor-operator. Removal of the control power will ensure that an active failure will not result in the inadvertent actuation of the accumulators. Thus, physical removal of the breaker from the circuit is unnecessary. In addition, the proposed change to the surveillance requirement provides clear description on what is an acceptable method for removing power from the accumulator isolation valves. Stating the requirement in this manner satisfies the Bases for the TS while reflecting the actual plant design, which precluded ready physical removal of the breaker.

The staff finds that the proposed changes do not represent a reduction in safety or alter any requirement. Power removed from the accumulator isolation valve motor-operator does provide adequate assurance that there will be no undetected closure of an accumulator motor-operated isolation valve operate. The proposed change is consistent with NUREG-143, and the Byron and Braidwood nuclear facilities have amended their TS, as approved by the NRC, in similar fashion. Therefore, the proposed changes are acceptable.

2.3 Proposed Changes to Unit 1 and 2 TS Bases 3/4.5.1

The third paragraph in the Unit 1 and 2 TS Bases 3/4.5.1 currently states "The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279- 1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required." The licensee proposes to spell out the word "Standard." The licensee also proposes to add the sentence "Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is greater than 2000 psig ensure that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve," to the end of the paragraph. The TS bases paragraph is proposed to read as "The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE *Standard* 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required. *Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is greater than 2000 psig, ensure that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve.*

The sixth paragraph of Unit 1 and 2 TS Bases 3/4.5.1 currently states "If the accumulator cannot be returned to OPERABLE status within the associated completion time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to less than or equal to 1000 psig within 12 hours." The licensee proposes replace the word "pressurizer" with the acronym "RCS." The licensee proposes the TS Bases to read as "If the accumulator cannot be returned to OPERABLE status within the associated completion time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and *RCS* pressure reduced to less than or equal to 1000 psig within 12 hours."

The purpose of TS Bases 3/4.5.1 is to detail and outline operability bases for accumulators. The licensee proposes to replace the word "pressurizer" with the acronym "RCS" (reactor coolant system) to provide consistency with the TS and plant design. The licensee proposes to add the text to the third paragraph to reflect guidance provided in NUREG - 1431 for SR 4.5.1.c.

The NRC staff finds that the proposed change does not constitute a reduction in safety and does not alter any requirement. Monitoring the RCS pressure at low pressure conditions is equivalent to monitoring the pressurizer pressure at low pressure conditions, and therefore substitution of the acronym "RCS" for "pressurizer" is acceptable. The proposed change is intended to provide consistency with plant design and therefore allow clarity and consistency in the TS. Furthermore, the change is consistent with NUREG - 1431 and similar changes have been made at the Byron and Braidwood nuclear facilities. Adding guidance to the TS from NUREG-1431 is intended to benefit and clarify the SR for the reader. Therefore, the staff finds the proposed changes acceptable.

2.4 <u>Proposed Unit 1 and 2 Administrative Changes to TS Pages 3/4 5-1, 3/4 5-2, B 3/4 5-1</u>

The licensee proposes to make administrative changes to the format of Unit 1 and 2 TS pages 3/4 5-1, 3/4 5-2, and B 3/4 5-1 in an ongoing effort to improve their appearance. The changes include adding "3/4 LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS" to the header, adding the acronym "ECCS" in the header, adding "Page" in

the footer, and deleting "NO." in the footer. For the TS Bases pages, these changes include rearranging the order of the text in the header, deleting "NO." in the footer, and adding "Page" in the footer.

The staff finds that the proposed administrative changes do not represent a reduction in safety or alter the TS requirements. The administrative changes are intended to maintain consistency and enhance usability and clarity of the TS. Therefore, the staff finds the proposed changes are acceptable.

3.0 <u>SUMMARY</u>

The proposed amendment would revise Unit 1 and 2 TS 3.5.1, Action "a" and "b," to reflect monitoring pressure data from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS SR 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. Furthermore, the licensee proposes to make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1. The proposed amendment does not cause changes to accident initiators or precursors, or to the accident analyses, and does not involve a significant reduction of safety.

Based on the above evaluation, the staff finds that the proposed changes to the TS are acceptable.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulatio ns, the Michigan State official was notified of the propose issuance of the amendments. The State official had no comments.

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 65735). Accordingly, the amendments meet 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 amendments.

6.0 <u>CONCLUSION</u>

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public

Principal Contributors: John Stang
Kimberly Leigh

Date: December 23, 1999

DATED: December 23, 1999

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-58, DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-74, DONALD C. COOK NUCLEAR PLANT, UNIT 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 23, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA6473 AND MA6474)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 236 to Facility Operating License No. DPR-58 and Amendment No. 218 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments would approve revisions to the Updated Final Safety Analysis Report (UFSAR) and Emergency Operating Procedures (EOPs) based on your analysis of an unreviewed safety question in response to your application dated September 17, 1999. By letter dated October 26, 1999, the NRC staff issued a request for additional information (RAI). By letters dated November 10, 1999, and November 19, 1999, the licensee responded to the RAI.

The proposed amendments would approve the licensee's use of methodology to credit the negative reactivity provided by insertion of the rod cluster control assemblies (RCCAs) into the reactor core following any design basis loss-of-coolant accident, during realignment from a cold leg recirculation to a hot leg recirculation configuration. This change to the licensing basis, when evaluated by the licensee in accordance with 10 CFR 59.59, resulted in an unreviewed safety question that requires prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90 prior to implementation. The proposed amendment also changes the Bases for TS Section 3/4.5.5, Refueling Water Storage Tank.

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

Sincerely,

/S/ Original signed by

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 236 to DPR-58
2. Amendment No. 218 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 236 to DPR-58
2. Amendment No. 218 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 236
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 September 17, 1999, as supplemented November 10, 1999, and November 19, 1999, 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.

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2. Accordingly, the license is amended to authorize revision of the Updated Final Safety Analysis (UFSAR) and Emergency Operating Procedures (EOPs) as set forth in the application for amendment by the licensee, dated September 17, 1999, and as supplemented November 10, 1999, and November 19, 1999, and as evaluated in the staff Safety Evaluation attached to this amendment. The licensee shall update the UFSAR and change the EOPs to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis loss-of-coolant accident as authorized by this license amendment and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications Bases

Date of Issuance: December 28, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 236

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
B 3/4 5-3	B 3/4 5-3

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. Consistent with the applicable LOCA analyses, the limits on RWST minimum volume and boron concentration ensure that 1) when combined with water from melted ice, the RCS, and the accumulators, sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) with the exception for the hot leg switchover subcriticality analysis following a cold leg break that incorporates control rod insertion, the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

At the time hot leg switchover is performed, there is the potential following a cold leg LOCA that boron-diluted liquid from the containment sump will displace the boron-concentrated liquid in the core. To compensate for this momentary reduction of boron in the core, control rod insertion has been credited after a cold leg LOCA to provide negative reactivity necessary to assure core subcriticality.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

The ECCS and containment integrity analyses assumed a maximum RWST water temperature above 100°F. Maintaining RWST water temperature at or below 100°F ensures the containment spray system will provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig, and that containment cooling will be maintained following a LOCA or steam line rupture inside containment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 218
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 17, 1999, as supplemented November 10, 1999, and November 19, 1999, 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 to authorize revision of the Updated Final Safety Analysis (UFSAR) and Emergency Operating Procedures (EOPs) as set forth in the application for amendment by the licensee, dated September 17, 1999, and as supplemented November 10, 1999 and November 19, 1999, and as evaluated in the staff Safety Evaluation attached to this amendment. The licensee shall update the UFSAR and change the EOPs to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis loss-of-coolant accident as authorized by this license amendment and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications Bases

Date of Issuance: December 28, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 218

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
B 3/4 5-3	B 3/4 5-3

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or a steam line rupture. Consistent with the applicable LOCA analyses, the limits on RWST minimum volume and boron concentration ensure that 1) when combined with water from melted ice, the RCS, and the accumulators, sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) with the exception for the hot leg switchover subcriticality analysis following a cold leg break that incorporates control rod insertion, the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

At the time hot leg switchover is performed, there is the potential following a cold leg LOCA that boron-diluted liquid from the containment sump will displace the boron-concentrated liquid in the core. To compensate for this momentary reduction of boron in the core, control rod insertion has been credited after a cold leg LOCA to provide negative reactivity necessary to assure core subcriticality.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

The ECCS and containment integrity analyses assumed a maximum RWST water temperature above 100°F. Maintaining RWST water temperature at or below 100°F ensures the containment spray system will provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig, and that containment cooling will be maintained following a LOCA or steam line rupture inside containment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 236 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 218 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated September 17, 1999, the Indiana Michigan Power Company (the licensee) requested approval to make changes to the Updated Final Safety Analysis Report (UFSAR) and applicable emergency operating procedures (EOPs) to credit the negative reactivity provided by insertion of the rod cluster control assemblies (RCCAs) following any design basis Loss-of-Coolant Accident (LOCA). The use of the methodology and associated changes to the UFSAR and EOPs, when evaluated by the licensee in accordance with 10 CFR 59.59, resulted in an unreviewed safety question that requires prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90 prior to implementation. The proposed amendment would also change the Bases for Technical Specification (T/S) 3/4.5.5, "Refueling Water Storage Tank (RWST)," which is affected by the application of the methodology.

By letter dated October 26, 1999, the NRC staff made a request for additional information (RAI) concerning the licensee's leak before break (LBB) analysis. By letters dated November 10, 1999, and November 19, 1999, the licensee provided the requested information. The information contained in the November 10 and November 19, 1999, letters supplemented the September 17, 1999, application and did not change the Commission's preliminary significant hazards determination.

2.0 EVALUATION

The concern addressed by taking credit for the negative reactivity provided by insertion of the RCCAs pertains to the post LOCA dilution of boron in the containment sump liquid due to the boron concentrating in the reactor vessel. Following a LOCA the potential exists for the reactor coolant collected in the recirculation sump to decrease in boron concentration to a value at, or below, the critical boron concentration for the reactor core. When the emergency core cooling system (ECCS) alignment is switched from cold leg injection alignment to a hot leg injection alignment following a LOCA, the potential exists for the reactor core to return to criticality during the relatively short time frame when the realignment first takes place. This occurs during the switchover. After the switchover is accomplished and the ECCS is aligned to the hot leg recirculation configuration, the core is provided with a back-flushing flow that aids in

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re-establishing an evenly distributed boric acid concentration within the reactor vessel. The switchover subcriticality analysis conservatively assumes that the diluted boron sump liquid completely displaces the more highly borated liquid in the vessel during the transition from cold leg recirculation to hot leg recirculation. It is this conservative assumption that results in the postulated return to criticality at the time of the switchover. Therefore, taking credit for insertion of the RCCAs provides a significant source of negative reactivity that can be used to offset the conservative assumption to ignore mixing, and can be used to demonstrate post-LOCA subcriticality at the time the switchover is performed.

The licensee's justification for use of the control rod insertion methodology following a LOCA is the ability of the rod cluster control assemblies inserting into the reactor core following design basis LOCAs. The licensee analyzed the following reactor coolant system (RCS) breaks.

60 in^2 Accumulator Line Break

98 in^2 Pressurizer Surge Line Break

144 in^2 Reactor Vessel Inlet Nozzle Break

144 in^2 Reactor Vessel Outlet Nozzle Break

594 in^2 Reactor Coolant loop Outlet Nozzle

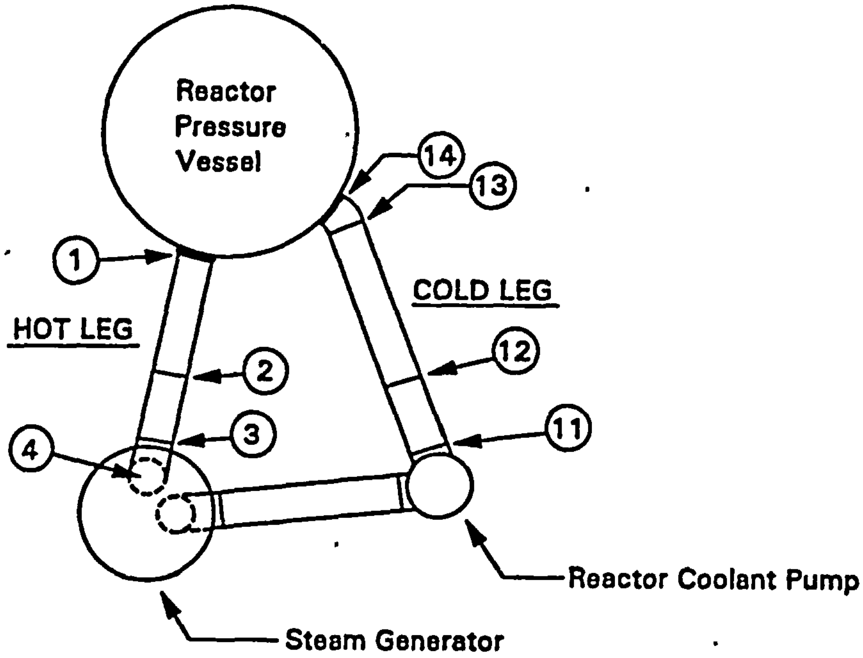
2.1 Main Coolant Loop Break Analysis

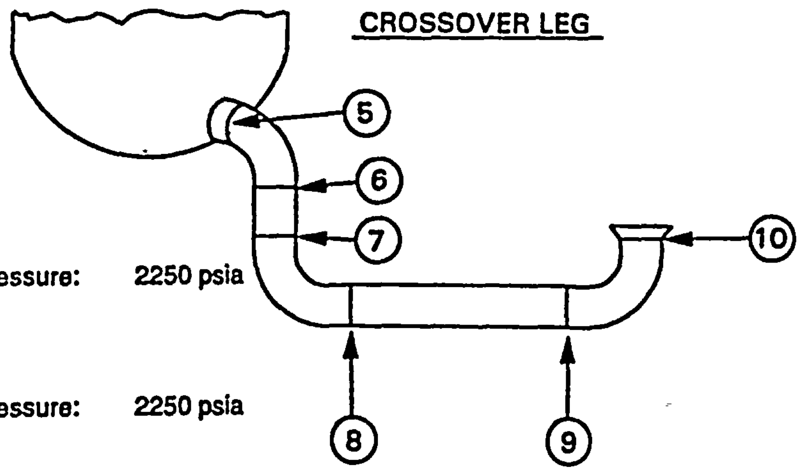
A part of the licensee's analysis concerned the of use of leak-before-break (LBB) technology to remove from consideration the dynamic effects (in this case the acoustic loads on the reactor internals generated by the depressurization associated with a "instantaneous" double-ended guillotine break (DEGB)) of a rupture of the D.C. Cook Unit 1 and 2 main coolant loops (MCLs). The NRC has previously permitted licensees to take credit for LBB piping behavior to address a similar issue, the resolution of Unresolved Safety Issue A-2 on asymmetric LOCA blowdown loads, and the licensee's submittal was consistent with the provision of Title 10 of the Code of Federal Regulations Part 50, Appendix A, General Design Criteria 4, which permits licensees to exclude the dynamic effects associated with postulated pipe ruptures from the facility's licensing basis if "analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." NRC approval of the licensee's proposed application of LBB would result in the licensee needing only to evaluate the effect on the reactor internals of the acoustic loads developed by the DEGB of reactor coolant system auxiliary lines (along with SSE loads).

The MCLs of D.C. Cook Units 1 and 2 had been previously approved for the application of LBB technology by NRC letter dated, November 22, 1985. The licensee reanalyzed the applicability of LBB to this D.C. Cook Unit 1 and 2 MCL as a result of changes to the D.C. Cook Unit 1 and Unit 2 reactor coolant systems due to steam generator replacement (SG) activities. The following sections address the LBB review.

2.1.1 Identification of Analyzed Piping and Piping Material Properties

The licensee's submittal identified and analyzed the following sections of high energy piping for LBB behavior verification. For each D.C. Cook Unit 1 and 2 MCL, the submittal addressed the piping from the reactor vessel to the SG (the hot leg), from the SG to the reactor coolant pumps (RCPs) (the crossover leg) and the piping from the RCPs to the pressure vessel (the cold leg). Fourteen separate locations were analyzed around on loop of the piping for each unit, and these locations are identified in Figure 1.





HOT LEG

Temperature 620°F, Pressure: 2250 psia

CROSSOVER LEG

Temperature 548°F, Pressure: 2250 psia

COLD LEG

Temperature 548°F, Pressure: 2250 psia

FIGURE 1 SCHEMATIC DIAGRAM OF D. C. COOK UNITS 1&2 PRIMARY LOOP SHOWING WELD LOCATIONS

The analyzed piping was identified as having the following material components. The main piping sections were manufactured from American Society for Mechanical Engineers (ASME) SA-351, Grade CF8M molybdenum-bearing cast stainless steel (CSS). The welds were identified as being stainless steel shielded metal arc welds (SMAWs).

For the material properties used in the LBB analysis, the licensee's analysis used Certified Materials Test Report (CMTR) data for the tensile properties and adjusted the tensile data to the temperature required for the analysis by interpolating between the ASME Code tensile properties and applying an equivalent ratio to the actual tensile property data. For determining the fracture toughness properties of the CSS pieces, the licensee's analysis evaluated the effect of thermal aging, as required by NRC staff guidance on LBB evaluations.

2.1.2 General Aspects of the Licensee's LBB Analysis

In this analysis, the licensee sought to reaffirm the LBB behavior of the subject piping considering changes made to the piping and supports as a result of SG replacement activities. As such, the analysis directly examined the impact of the recalculated piping loads during normal operation (NOP) and SSE conditions on the critical flaw margin and leakage flaw stability criteria. The licensee's analysis made use of the Westinghouse proprietary evaluation codes for assessing the fracture mechanics behavior of the leakage flaw and critical flaw. A brief non-proprietary overview of the analysis and results is provided below.

2.1.3 Licensee Evaluation of the Main Coolant Loop Piping

The licensee's analysis in WCAP-15131, Revision 1, was initiated by an evaluation to determine if any atypical loading condition or degradation mechanism exists which could invalidate the assumptions of the LBB analysis. This evaluation included a review of pressurized water reactor operating history and the potential for water hammer events, intergranular stress corrosion cracking (IGSCC), stress corrosion cracking (SCC), and/or low or high cycle fatigue of the MCL piping. The licensee concluded that IGSCC and SCC were extremely unlikely based upon primary water chemistry control and monitoring. Water hammer events are unlikely due to pressurized-water reactor (PWR) operational characteristics which preclude voiding in these normally filled lines. Finally, low and high cycle fatigue is addressed by piping designs to meet ASME Code requirements and vibrational monitoring systems. In summary, the licensee determined that these loading conditions and degradation mechanisms had an extremely low probability of occurrence which did not affect the ability of the subject piping to be qualified for LBB.

Next, the licensee determined the appropriate loading conditions to be used for the analysis. The licensee calculated the total piping stress at the fourteen locations identified in Figure 1 from the algebraic sum of the forces and moments due to deadweight, thermal expansion, and pressure loads during normal 100 percent power operation. These loads, herein called the "NOP" loads, along with the corresponding pipe dimensions at the fourteen locations, are given in Table 1. The licensee then calculated the total piping stress at the fourteen locations from the absolute sum of the forces and moments due to deadweight, thermal expansion, pressure loads during normal 100 percent power operation, and SSE inertial and anchor motions. These loads, herein called the "NOP+SSE" loads, for each location are given in Table 2. Based on an evaluation of these loads, the licensee determined that the limiting locations for the LBB analysis would be locations 1, 10 , and 11, as shown in Figure 1.

Then the leakage flaw size at each of these limiting locations was determined. The leakage flaw size was determined by applying the NOP loads to a postulated through-wall flaw and determining what size flaw would provide 10 times the leakage detectable by the D.C. Cook containment leakage monitoring system. This safety factor of 10 on the detectable leakage is included in the NRC guidance on LBB evaluations to account for uncertainties in the thermohydraulic calculations for the fluid flow through the crack. Since the D.C. Cook containment leakage monitoring system is capable of detecting 1 gallon per minute (gpm) of leakage (in the course of 4 hours), the leakage size flaw is 10 gpm. The leakage size flaw for locations 1, 10, and 11 are shown in column 2 of Table 3.

The licensee then determined the critical flaw size at each location. The NOP+SSE loads were applied to a postulated through-wall flaw, and the minimum flaw size which failed under the NOP+SSE loads was defined as the critical flaw size. In order to demonstrate that this piping met the margins on flaw size required by NUREG-1061, Volume 3, and DSRP 3.6.3, the critical flaw size at each location must be twice the length of the leakage flaw size. The licensee's analysis demonstrated this in two ways. First, the critical flaw size was determined by using a limit load analysis methodology. This assumes that the piping fails by plastic collapse when the net section of the piping as a stress level equals the flow stress of the material. The second method to demonstrate that a margin of two on flaw size was achieved involved an analysis using a J-integral approach to assessing fracture behavior. For this analysis, the licensee analyzed a flaw equivalent to twice the length of the leakage size flaw and demonstrated that it did not propagate unstably to failure under NOP+SSE loadings. This ensured that a margin of two was achieved without directly determining the critical flaw size. The results of the licensee's analysis are given in columns 3 and 4 of Table 3.

A final criteria that must be evaluated to demonstrate the LBB qualification of the piping is to show that the leakage size flaw is stable under loads potentially greater than the NOP+SSE load combination. If the NOP+SSE loads are summed algebraically, then they should be multiplied my a factor of $\sqrt{2}$ and the leakage flaw should still be found to be stable. In this evaluation, since the licensee chose to sum the NOP+SSE loads absolutely, no additional multiplier is required. Therefore, its demonstration that a flaw twice the size of the leakage flaw (i.e. the critical flaw) was stable also demonstrates that the leakage flaw will be stable under these loads.

2.1.4 <u>Leak Before Break Staff Summary</u>

Based on the information provided by the licensee regarding the materials comprising the D.C. Cook Unit 1 and 2 MCL piping and the loads under NOP and SSE conditions, the staff independently assessed the compliance of this system with the LBB criteria established in NUREG-1061, Volume 3. The staff has concluded that the analyses submitted by the licensee, along with additional information submitted regarding the torsional moments at each analysis location, were sufficient to demonstrate that LBB behavior would be expected from the subject piping following the installation of the replacement SGs. The staff's evaluation, which follows the guidance of NUREG-1061, Volume 3, is provided below.

2.1.5 Identification of Analyzed Piping and Piping Material Properties

The staff examined the list of materials identified for the MCL piping and concluded that it would be necessary to evaluate the material properties of both the CSS piping segments and the associated SMAWs at the limiting locations because of their susceptibility to thermal aging. NUREG-1061, Volume 3, specifies particular aspects which should be considered when developing materials property data for LBB analyses. First, data from the testing of the plant-specific piping materials is preferred. However, in the absence of such data, more generic data from the testing of samples having the same material specification may be used. More specifically, it was noted in Appendix A of the NUREG that "[m]aterial resistance to ductile crack extension should be based on a reasonable lower-bound estimate of the material's J-resistance curve," while section 5.2 of the NUREG stated that the materials data should include,"appropriate toughness and tensile data, long-term effects such as thermal aging and other limitations."

The staff noted that although tensile test data had been provided by the licensee for the cast stainless steel heats in the D.C. Cook Unit 1 and 2 MCLs, this data did not account for thermal aging effects. Likewise, no heat-specific fracture toughness data for the D.C. Cook materials was provided in the aged condition. The staff's evaluation assumed that conservatively high amounts of δ-ferrite were present (greater than 15 percent, based on the highest value cited for a specific heat, 39344-2, of 22.92 percent) in the CSS pieces at the limiting locations. Results from work at Argonne National Laboratory (References 1 and 2), sponsored by the NRC, were used as the basis for developing generic J-R and stress-strain curves for the CSS material. The CSS material properties parameters used for the staff's evaluation are given in Table 5. Materials property parameters for the evaluation of the aged stainless steel SMAWs were also developed based on work by Argonne National Laboratory (Reference 3) and are given in Table 6. These generic material properties representations for CSS and SMAWs are consistent with those chosen by the staff in previous LBB reviews.

2.1.6 General Aspects of the Staff's LBB Analysis

The staff's analysis was performed in accordance with the guidance provided in NUREG-1061, Volume 3. Based on the information submitted by the licensee, the staff determined the critical flaw size at the bounding location for the MCL using the codes compiled in the NRC's Pipe Fracture Encyclopedia (Reference 4). For the purposes of the staff's evaluation, the critical location was defined by those locations at which materials with low postulated fracture toughness existed in combination with high ratios of SSE-to-NOP stresses. This was because high SSE stresses tend to reduce the allowable critical flaw size while low NOP stresses increase the size of the leakage flaw required to produce 10 gpm of leakage. In particular, when evaluating the critical flaw in thermally-aged CSS base materials, the staff used the LBB.ENG2 code developed by Brust and Gilles (Reference 5). When evaluating SS SMAWs, the staff used the LBB.ENG3 code developed by Battelle (Reference 5) for the express purpose to determine if a substantial difference in the tensile properties of the weld and base metal was expected.

The staff then compared the critical flaw at the bounding location to the leakage flaw which provided 10 gpm of leakage under NOP conditions to determine whether the margin of 2 defined in NUREG-1061, Volume 3, was achieved. The leakage flaw size calculation was carried out

using the Pipe Crack Evaluation Program (PICEP, Revision 1) analytic code developed by the Electric Power Research Institute. The 10 gpm value was defined by noting that the D.C. Cook Unit 1 and 2 containment leakage detection systems would be able to detect a 1 gpm leak in the course of one hour and a factor of 10 is applied to this 1 gpm detection capability to account for thermohydraulic uncertainties in calculating the leakage through small cracks.

2.1.7 Staff Evaluation of the D.C. Cook Unit 1 and 2 Main Coolant Loop

First, the staff examined the licensee's evaluation regarding atypical loading conditions or degradation mechanisms which could invalidate the assumptions of the LBB analysis. The staff concurred that the evaluation of pressurized water reactor operating history and the potential for water hammer events, intergranular stress corrosion cracking (IGSCC), stress corrosion cracking (SCC), and/or low or high cycle fatigue of the MCL piping was appropriate. The staff agreed with the licensee's conclusion that IGSCC and SCC were extremely unlikely based upon primary water chemistry control and monitoring. The staff also agreed that water hammer events are unlikely due to PWR operational characteristics and that low and high cycle fatigue is addressed by piping designs that meet ASME Code requirements and vibrational monitoring systems. In summary, the staff concurred with licensee's determination that these loading conditions and degradation mechanisms had the extremely low probability of occurrence which did not affect the ability of the subject piping to be qualified for LBB.

The staff's evaluation then examined the loadings submitted by the licensee. It was noted that the summation methodology utilized by the licensee was not completely consistent with the guidance provided by the staff in NUREG-1061 Volume 3 or Draft Standard Review Plan (DSRP) 3.6.3 for determining loads for LBB analyses. The inconsistency in the licensee's analysis was that the licensee did not include the torsional moments (as directed to in NUREG-1061 Volume 3). However, the licensee subsequently provided those moment components for each piping location in a letter dated November 19, 1999, so that information was available for the staff's evaluation. Based on the staff's evaluation of the loadings supplied by the licensee, the staff concluded that the limiting locations for the MCL piping evaluation would be location 1 (at the hot leg nozzle connection to the reactor vessel), location 5 (at the SG outlet nozzle), location 10 (at suction nozzle of the RCP) and location 11 (at the discharge nozzle of the RCP) as shown in Figure 1.

At each location, the staff evaluated the critical and leakage flaw sizes for the CSS material and the associated SMAW weld. The material properties assumed by the staff for the CSS and SMAW materials are shown in Tables 4 and 5 and the loads used in the staff's leakage flaw and critical flaw analysis (which include the torsional moments) are shown in Table 6. The staff applied the PICEP code using the nominal piping dimensions and a crack surface roughness of $\varepsilon = 0.0003$ inches. This procedure calculated a 10 gpm leakage flaw size for each material. The critical flaw size determined by using the LBB.ENG2 or LBB.ENG3 code as appropriate. The ratio of the critical-to-leakage-flaw size was then determined for comparison to the recommended margin of 2 in NUREG-1061, Volume 3. These results are summarized in Table 7. Since the margin of 2 on the crack sizes was achieved for each location, the leakage flaw was also shown to be stable given the absolute summation of the NOP and SSE loads (as calculated by the licensee plus the torsional loads included in the staff's analysis), and meets the margin on loading recommended by NUREG-1061, Volume 3.

2.1.8 Main Coolant Loop Break Summary

Based on the information and analysis supplied by the licensee, the staff was able to independently assess the LBB status of the D.C. Cook Unit 1 and 2 MCL piping. The staff has concluded that it has been demonstrated that the LBB behavior of the MCL is covered by the analysis submitted by the licensee and the independent evaluation by the staff presented in this SE. Furthermore, the licensee should be permitted to credit this conclusion for eliminating the dynamic effects associated with the postulated rupture of these sections of piping from the D.C. Cook Unit 1 and 2 facility licensing basis, consistent with the provisions of 10 CFR 50, Appendix A, General Design Criteria 4, including credit for removing from consideration the acoustic loads that would be generated by a DEGB of the MCL when evaluating the ability to insert the RCCAs.

Table 4: Parameters used in Staff Evaluation of Aged D.C. Cook CSS Piping

Parameter	Value
Young's Modulus	25500 ksi
Yield Strength	32.8 ksi
Ultimate Tensile Strength	78.8 ksi
Sigma-zero	32.2 ksi
Epsilon-zero	0.00129
Ramberg-Osgood Alpha	1.276
Ramberg-Osgood n	6.6
C	2599 in-lb / in^2
n	0.31

Note: $J = C(\Delta a)^n$

Table 5: Parameters used in Staff Evaluation of Aged D.C. Cook SS Piping Welds

Parameter	Value
Young's Modulus	25000 ksi
Yield Strength	49.4 ksi
Ultimate Tensile Strength	61.4 ksi
Sigma-zero	35.0 ksi
Epsilon-zero	0.00125
Ramberg-Osgood Alpha	9.0
Ramberg-Osgood n	9.8
A	228 in-lbs./in^2
C	476 in-lbs./in^2
n	0.643

Note: $J = A + C(\Delta a)^n$

Table 6: Loads Used in the Staff's Evaluation of Locations 1, 5, 10, and 11

	Location 1	Location 5	Location 10	Location 11
Normal Ops. Axial (Including Pressure)	1529 kips	1664 kips	1796 kips	1372 kips
Normal Ops. Bending	28495 in-kips	4557 in-kips	7313 in-kips	5116 in-kips
NOP + SSE Axial (Including Pressure)	1766 kips	1891 kips	1866 kips	1492 kips
NOP + SSE Bending	30033 in-kips	10790 in-kips	16837 in-kips	14347 in-kips

Table 7: Results of the Staff's Evaluation for Locations 1, 5, 10, and 11

Location	Leakage Flaw Size	Critical Flaw Size	Margin
1 - CSS	5.8 inches	28.9 inches	5.0
1 - SMAW	5.9 inches	30 inches	5.1
5 - CSS	9.8 inches	> 42 inches	> 4.3
5 - SMAW	9.9 inches	> 45 inches	> 4.5
10 - CSS	8.9 inches	> 42 inches	> 4.7
10 -SMAW	9.0 inches	> 45 inches	> 5
11 - CSS	8.1 inches	> 35 inches	> 4.3
11 - SMAW	8.2 inches	> 37 inches	> 4.5

2.2 OTHER RCS BREAK ANALYSIS

The relevant pipe breaks in the branch lines that were considered in the analysis are the 60-inch^2 accumulator line break and the 98-inch^2 pressurizer surge line break. These breaks are not covered by the application of the LBB technology at D. C. Cook. The licensee used the MULTIFLEX 3.0 computer code to calculate the blowdown loads on the reactor vessel and the reactor vessel internals, including the guide tubes and core barrel. MULTIFLEX has previously been used by Westinghouse to calculate blowdown loads. The version of MULTIFLEX used for this analysis is considered by Westinghouse as an improved version that was developed specifically for the Westinghouse Owner's Group Baffle Barrel Bolt Program (BBBP). Westinghouse stated that previous BBBP analyses performed using this version of the code were accepted by the NRC.

The NRC did not perform a detailed review of the MULTIFLEX 3.0 Code, but found the general methodology reasonable and acceptable. A conservative and previously accepted 1 millisecond break opening time was assumed. Therefore, these calculations of LOCA blowdown loads are judged to be conservative. Also, the blowdown loadings have been determined using a methodology which has been previously accepted by the NRC.

2.3 CONTROL ROD INSERTION

The ability to insert the control rods is a function of the guide tube's deflection during a LOCA transient. As the amount of deflection increases, control rod insertion time will first increase due to increased resistance and at sufficient deflection, control rod insertion will be precluded. Since the guide tube is a complex structure, and the motion of control rods are dependent on the amount of friction between the two components, it is difficult to determine control rod insertion through analytical means. For this reason, guide tube scram tests have been performed by Westinghouse in the past to experimentally determine the limits of control rod insertability. Guide tube scram tests have been performed on 96"-17x17, 150"-15x15 guide tubes, (References 6 and 7). Full-size guide tubes, with rod control clusters, were mechanically loaded at discrete elevations to simulate flow loads experienced during a postulated LOCA transient. The insertion for the control rods as a function of the guide tube deflection, which in turn is a function of the applied mechanical loads, were recorded during the tests. The allowable load is then determined as the limiting applied mechanical load corresponding to the guide tube's permanent loss of function. Westinghouse determined the total LOCA loads by combining the inertial acceleration and acoustic loads calculated by MULTIFLEX with the hydraulic cross flow loads, i.e., drag loads, which were estimated based upon scale model tests and plant strain measurements, together with information from the MULTIFLEX and other hydraulic calculations. A dynamic load factor was applied to account for the transient nature of the drag loading. This total LOCA load was added using the square root sum of squares (SRSS) method to the peak safe shutdown earthquake (seismic) load to obtain the total load. The staff finds the methodology used by Westinghouse to calculate the combined peak guide tube loading to be reasonable and consistent with industry practice. In most respects, this methodology is similar to NRC-accepted methodology for assuring control rod insertion during faulted LOCA and seismic conditions in other applications (Reference 8). Therefore, the staff finds it acceptable.

Westinghouse compared the calculated combined peak loads to the allowable values. Due to the differences in fuel assemblies between Unit 1 (15 x 15) and Unit 2 (17 x 17), the allowable loads and the peak combined loads differ between the units. For both units, the calculated peak combined load showed considerable margin to the allowable. Therefore, the maximum guide tube deflection which occurs under the limiting analyzed break size noted above plus deflection from seismic loading will not prevent the control rods from inserting.

2.4 FUEL ASSEMBLY GRID LOADING CONSIDERATIONS

The general analytical procedure for evaluating fuel assembly transient response to seismic and LOCA transients was provided schematically by the licensee, outlining the main steps in the analytical sequence. Forcing functions for the reactor internals model are based on postulated LOCA and seismic conditions. The hydraulic forces and loop mechanical loads resulting from a postulated LOCA pipe rupture are prescribed at appropriate locations of the reactor pressure vessel (RPV) model. For the seismic analysis, the plant-specific design acceleration spectra are specified, based upon the plant site characteristics. For the current analysis, the synthesized seismic time histories are calculated from the D. C. Cook plant-specific acceleration response spectra envelope. These spectra are for the containment buildings at the appropriate elevation and the design-basis damping. Both the LOCA and seismic time histories are applied to the RPV system model. The core plate motions from the dynamic analysis of this model are obtained and are then input to the reactor core model.

The limiting LOCA and seismic grid impact loads for 15x15 and 17x17 assembly cores have been summarized. The maximum grid loads obtained from SSE and LOCA loading analyses, were combined as required using the SRSS method. The results of the seismic and LOCA analyses of the maximum impact forces for the 15x15 and 17x17 structural grids are compared to allowable grid distortion loads. These allowable grid loads are experimentally established as the 95-percent confidence level on the mean from the distribution of grid distortion data at normal plant operating temperature. Acceptability of the fuel assembly grid performance for RCCAs control rod insertion is verified by demonstrating that no grid deformation occurs in assemblies directly beneath control rod locations. For both Units 1 and 2, no fuel assembly grid distortion was calculated and thus control rod insertion will not be impeded under limiting break plus seismic loadings.

2.5 REACTIVITY CONTROL

The negative reactivity associated with the RCCAs being available will provide adequate negative reactivity to ensure that following a design-basis LOCA, the realignment from a cold leg recirculation configuration to a hot leg recirculation configuration will not result in core re-criticality. The amount of negative reactivity available is verified every fuel cycle and is shown to be sufficient to prevent re-criticality.

3.0 SUMMARY

Based on the review of the structural analysis methodology and results as discussed above, the staff finds that, for both Units 1 and 2, the maximum fuel assembly guide tube deflections which occur during limiting LOCA breaks plus seismic loadings will not prevent the control rod from inserting. In addition, no fuel assembly grid distortion will occur and thus control rod insertion will not be impeded for these loads. Therefore the staff finds that it is acceptable for the licensee to revise the UFSAR and EOPs to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis LOCA.

4.0 REFERENCES

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4. Pipe Fracture Encyclopedia, produced on CD-ROM by Battelle-Columbus Laboratory for the U.S. Nuclear Regulatory Commission, 1997.

5. Brust, F.W., et. al., "Assessment of Short Through-Wall Circumferential Cracks in Pipes," NUREG/CR-6235, BMI-2179.

6. "Scram Deflection Test Report 17x17 Guide Tubes, 96 Inch and 150 Inch," WCAP 9251, December 1977

7. "Summary Report on PGE Scrammability Test," CE-RD-233, February 17, 1969

8. D. Beaumont, et. al, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," WCAP 9401-P-A, August 1981, Westinghouse Report

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 56531). Accordingly, the amendments meet 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 amendments.

7.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: M. Mitchell
J. Rajan
M. Chatterton

Date: December 28, 1999

12/23/99

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Subject:
Memorandum requesting publication of notice of issuance of amendments to licenses DPR-58 and DPR-74, in biweekly FR notice. Amendments would approve the licensee's revision of the UFSAR & Emergency Operating Procedures.

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6473 AND MA6474)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: September 17, 1999, as supplemented November 10 and 19, 1999

Brief description of amendments: The amendments would approve the licensee's revision of the Updated Final Safety Analysis Report and Emergency Operating Procedures to use methodology to credit the negative reactivity provided by insertion of the rod cluster control assemblies (RCCAs) into the reactor core following any design basis loss-of-coolant accident, during realignment from a cold leg recirculation to a hot leg recirculation configuration. This change to the licensing basis, when evaluated by the licensee in accordance with 10 CFR 59.59, resulted in an unreviewed safety question that requires prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90 prior to implementation. The amendments also change the Bases for TS Section 3/4.5.5, Refueling Water Storage Tank.

Date of issuance:

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment Nos.:

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

DF02

Date of initial notice in FEDERAL REGISTER: October 20, 1999 (64 FR 56531)

The licensee's letters of November 10 and 19, 1999, provided additional information that did not change scope of the application or the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated

No significant hazards consideration comments received: No.

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Date of initial notice in FEDERAL REGISTER: October 20, 1999 (64 FR 56531)

The licensee's letters of November 10 and 19, 1999, provided additional information that did not change scope of the application or the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated

No significant hazards consideration comments received: No.

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REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC NOS. MA7049 AND 7050)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 23, 1999

MEMORANDUM TO: BiWeekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC NOS. MA7049 AND 7050)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: November 5, 1999

Brief description of amendment: The amendments would revise Unit 1 and 2 Technical Specification (TS) 3.5.1, Action "a" and "b," to reflect the monitoring of pressure from the Reactor Coolant System instead of the pressurizer. The amendment would also revise Unit 1 and 2 TS Surveillance Requirement 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is physically removed from the circuit. Furthermore, the amendment would make administrative changes to Unit 1 and 2 TS Bases 3/4.5.1.

Date of issuance: December 23, 1999

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment No.: 237 and 219

Facility Operating License Nos. DPR-58 and DPR 74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: November 23, 1999 (64 FR 65735)

DFX2

0036705277

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 23, 1999.

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 23, 1999.

No significant hazards consideration comments received: No.

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Docket: 05000315, Notes: N/A

CP-1

December 22, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE REPAIR CRITERIA AND SLEEVING METHODOLOGIES (TAC NO. MA6389)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 238 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 17, 1999.

The amendment removes the voltage-based repair criteria, F* repair criteria, and sleeving methodologies from the Unit 1 TSs and clarifies the Bases sections accordingly.

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

Sincerely,

Original signed by:

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 238 to DPR-58
2. Safety Evaluation

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December 22, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE REPAIR CRITERIA AND SLEEVING METHODOLOGIES (TAC NO. MA6389)

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Sincerely,

Original signed by:

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 238 to DPR-58
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 22, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE REPAIR CRITERIA AND SLEEVING METHODOLOGIES (TAC NO. MA6389)

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Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 238 to DPR-58
2. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 238
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 August 17, 1999, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 238 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 22, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 238

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-11a	-
3/4 4-11b	-
3/4 4-12	3/4 4-12
3/4 4-14	3/4 4-14
B 3/4 4-2a	B 3/4 4-2a
B 3/4 4-2b	-
B 3/4 4-2b	-
B 3/4 4-3	B 3/4 4-3
B 3/4 4-5	B 3/4 4-5

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All tubes that previously had detectable wall penetrations (greater than or equal to 20%) that have not been plugged.

* This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions for Operation for Specification 3.4.1.3 are maintained.

SURVEILLANCE REQUIREMENTS (continued)

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for the samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

SURVEILLANCE REQUIREMENTS (continued)

Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality or replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. Percent Degradation means the amount of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40 percent or more of the nominal tube wall thickness shall be plugged prior to returning the steam generator to service.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Inspection determines the condition of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes, and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 6.9.1	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

S=3(N/n)% N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system. The allowable primary-to-secondary leak rate is 150 gallons per day per steam generator. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Leakage in excess of this limit will require plant shutdown and an inspection, during which the leaking tubes will be located and plugged. A steam generator while undergoing crevice flushing in Mode 4 is available for decay heat removal and is operable/operating upon reinstatement of auxiliary or main feed flow control and steam control.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit which is defined in Specification 4.4.5.4.a.6. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitations provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The limitation on seal line resistance ensures that the seal line resistance is greater than or equal to the resistance assumed in the minimum safeguards LOCA analysis. This analysis assumes that all of the flow that is diverted from the boron injection line to the seal injection line is unavailable for core cooling.

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) will minimize the potential for a large leakage event during steam line break under LOCA conditions. This operating leakage limit will ensure the calculated offsite doses will remain within 10 percent of the 10 CFR 100 requirements and that control room habitability continues to meet GDC-19. Leakage in the intact loops is limited to 150 gpd.

Also, the 150 gpd leakage limit incorporated into this specification is more restrictive than the standard operating leakage limit and is intended to provide an additional margin. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected and the plant shut down in a timely manner.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Cook Nuclear Plant site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 238 TO 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 dated August 17, 1999, the Indiana Michigan Power Company (IM, or the licensee) requested an amendment to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant (D.C. Cook), Unit 1. The proposed amendment would remove the voltage-based repair criteria, F* repair criteria, and sleeving methodologies from the Unit 1 TSs and clarify the Bases sections accordingly. The changes are proposed due to the planned replacement of the Unit 1 steam generators (SGs) during the current outage.

2.0 EVALUATION

2.1 Background

IM plans to replace the Unit 1 SGs during the current outage prior to Unit 1 restart. The current surveillance requirements for sample selection, inspection frequency, acceptance criteria, repair methods, and required reports were specifically developed for application to the degraded Westinghouse Model 51 SGs (OSGs) installed in Unit 1. These requirements were developed, in part, to permit tubes to remain in service that were experiencing various tube degradation mechanisms. After replacement of the OSGs, the current voltage-based repair criteria, F* repair criteria, and sleeving methods will no longer be applicable due to material and design changes incorporated into the replacement steam generators (RSGs), which were manufactured by Babcock and Wilcox.

The analyses performed to support application of the voltage-based and F* repair criteria were specifically based on the OSGs. Following replacement of the OSGs, these analyses will not apply to the RSGs. In addition, the currently approved Westinghouse mechanical, Westinghouse laser-welded, and Combustion Engineering leak-tight welded sleeving processes will no longer be applicable to Unit 1. These sleeving processes were developed specifically for Westinghouse SG materials and design and are, therefore, not applicable to the Babcock and Wilcox RSGs.

IM proposes to revise TS 3/4.4.5, TS Bases 3/4.4.5, TS Bases 3/4.4.6.2, and TS Bases 3/4.4.8. These revisions would remove the TS Unit 1 modifications made to address the various SG tube degradation mechanisms that have occurred on the Unit 1 OSGs. Incorporation of these proposed changes will return the Unit 1 TSs to the original licensing bases, except for the operational leakage limits, which are consistent with NUREG-0452, Rev. 4, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," and the Unit 2 TSs.

2.2 Evaluation of Proposed Changes

The proposed TS revisions reflect the significant design differences between the OSGs and the RSGs. The SG tube repair criteria and the associated surveillance and reporting requirements are identified as interim plugging criteria. These repair criteria were required to address a form of SG tube degradation in the OSG known as outer diameter stress corrosion cracking (ODSCC). For Unit 1 operation after SG replacement, the voltage-based and F* requirements will no longer be applicable due to the design differences between the OSGs and the RSGs. In addition, the sleeving methodologies referenced in the Unit 1 TSs are specific to the Westinghouse Model 51 SGs and have not been analyzed or approved for use by the NRC for use with the RSGs. The removal of the interim plugging criteria and the associated surveillance and reporting requirements required no independent staff analysis since the RSGs do not have the same type of SG tube support structures as the OSGs, and, therefore, the SG tube repair criteria for the ODSCC flaws that occurred in the OSGs are not applicable to the RSGs. Accordingly, the RSGs will not be subject to the relatively large end-of-cycle SG tube leakage that could occur under postulated accident conditions. Similarly, the proposed removal of the SG tube alternate repair criteria for flaws occurring within the OSG tubesheets (i.e., the F* repair criteria) and the proposed removal of the sleeving methodologies also required no independent staff analysis for the same reason. The various SG tube repair criteria and the associated surveillance and reporting requirements are not required to ensure the safe operation of the RSGs due to the significant design differences between the OSGs and the RSGs.

The TS acceptance limits will be based on through-wall criteria that require tubes to be plugged when imperfections exceed the plugging limit of 40 percent of the nominal tube wall thickness. The proposed program for periodic inservice inspection of the RSGs monitors the integrity of the SG tubing to provide reasonable assurance that there is sufficient time to take proper and timely corrective action if any tube degradation is present. The proposed program is consistent with NUREG-0452 and was the basis for the original TSs issued for Unit 1. The purpose of the TS plugging limit, in conjunction with surveillance and maintenance programs, is to provide reasonable assurance that the SG tubes accepted for continued service will retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions. Although D.C. Cook is not a General Design Criteria (GDC) plant, IM has determined that the RSG design is consistent with GDC 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A. Compatibility with these GDCs supports the application of TS acceptance limits based on through-wall criteria.

The licensee proposes to revise TS Surveillance Requirement (TSSR) 4.4.5.2, "Steam Generator Tube Sample and Selection," to remove reference to previous defects or imperfections repaired by sleeving and to revise TSSR 4.4.5.2.b.1 and TSSR 4.4.5.2 to remove reference to sleeving. The licensee proposes to delete TSSRs 4.4.5.2.b.4, 4.4.5.2.c, 4.4.5.2.e, and 4.4.5.2.f, and to renumber TSSR 4.4.5.2.d. These proposed changes reflect the removal

of sleeving methodologies and repair criteria applicable to the OSGs that are not applicable to the RSGs, and are acceptable.

The licensee proposes to revise TSSR 4.4.5.3, "Inspection Frequencies," to add a requirement for SG inservice inspection after 6 effective full power months but within 24 calendar months after SG replacement. The revision is acceptable in that it applies a requirement on the inspection frequency of the first inservice inspection of the RSGs which is identical to that of the OSGs.

The licensee proposes several changes to TSSR 4.4.5.4, "Acceptance Criteria." The licensee proposes to change TSSR 4.4.5.4.a items 1, 2, 3, 6, 7, and 8 to remove references to sleeving and to change TSSR 4.4.5.4.a items 5 and 6 to remove references to repair limits. The licensee proposes to revise TSSR 4.4.5.4.a.5 to define a defect in terms of the plugging limit, to revise TSSR 4.4.5.4.a.6 to remove discussion of the applicability of F* tubes and sleeves, and to revise TSSR 4.4.5.4.a.8 to remove reference to the interim plugging limit. The licensee proposes to delete TSSR 4.4.5.4.a items 9, 10, 11, 12, and 13, and TSSR 4.4.5.4.c. The licensee proposes to revise TSSR 4.4.5.4.b to remove reference to sleeves and repair limits. These proposed changes reflect the removal of sleeving methodologies and repair criteria applicable to the OSGs that are not applicable to the RSGs, and are acceptable.

The licensee proposes several changes to TSSR 4.4.5.5, "Reports." The licensee proposes to revise TSSR 4.4.5.5.a, TSSR 4.4.5.5.b.1, and TSSR 4.4.5.5.b.3 to remove references to sleeving, and to delete TSSR 4.4.5.5.d. These required reports are not applicable to the RSGs since the RSGs are not subject to the same forms of SG tube degradation in the tubesheet area that have occurred in the OSG; therefore, the changes are acceptable.

The licensee proposes to revise TS Table 4.4-2 to remove references to sleeving. These proposed changes reflect the removal of sleeving methodologies applicable to the OSGs that are not applicable to the RSGs, and are acceptable.

The licensee proposes to revise TS Bases 3/4.4.5, "Steam Generators Tube Integrity," to remove references to repair of defective tubes, repair limits, and sleeving, and to remove details on voltage-based repair limits and approved methodologies for sleeving. These proposed changes reflect the removal of sleeving methodologies and repair criteria applicable to the OSGs that are not applicable to the RSGs, and are acceptable.

The licensee proposes to revise TS Bases 3/4.4.6.2, "Operational Leakage," to remove references to crack growth and expected primary-to-secondary leakage during a main steamline break accident. Under the interim plugging criteria, a leak rate of 8.4 gpm was determined to be the upper limit for primary-to-secondary leakage in a faulted SG. This leakage, combined with the maximum of 150 gpd allowed leakage from each nonfaulted SG, was determined to limit the offsite dose to 10 percent of the 10 CFR Part 100 limits. Following replacement of the SGs, the leakage is limited to 150 gpd from each SG. The 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected tube leak and minimizes the potential for excessive leakage or tube burst in the event of main steamline break or loss-of-coolant accident conditions, and therefore, is acceptable.

The licensee proposes to revise TS Bases 3/4.4.8, "Specific Activity," to remove the discussion of offsite dose following a main steamline break with a primary-to-secondary leak rate of 120 gpm. The discussion was added to TS Bases 3/4.4.5 and 3/4.4.8 by Amendment No. 166, dated July 29, 1992. Subsequently, Amendment No. 178, dated March 15, 1994, imposed a limit of 12.6 gpm (later revised to 8.4 gpm) SG leakage for the main steamline break accident analysis to limit offsite doses to 10 percent of the 10 CFR Part 100 limits. Amendment No. 178 removed the reference to the 120 gpm evaluation from Bases 3/4.4.5, but inadvertently did not revise Bases 3/4.4.8. Since the discussion regarding the 120 gpm SG leak rate does not apply to the RSGs, the revision is acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The 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 (64 FR 54375). 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.

5.0 CONCLUSION

The 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 Contributor: F. Lyon

Date: December 22, 1999

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Docket: 05000315, Notes: N/A

CP-1

December 22, 19

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC NO. MA6389)

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: August 17, 1999

Brief description of amendment: The amendment removes the steam generator voltage-based repair criteria, F* repair criteria, and sleeving methodologies from the Unit 1 Technical Specifications and clarifies the Bases sections accordingly.

Date of issuance: December 22, 1999

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 238

Facility Operating License No. DPR-58: Amendment revises the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 6, 1999 (64 FR 54375)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1999.

No significant hazards consideration comments received: No.

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Docket: 05000315, Notes: N/A

CP-1

December 20, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGE, "SEALED SOURCE CONTAMINATION," AND ITS ASSOCIATED BASES TO ADDRESS TESTING REQUIREMENTS FOR FISSION DETECTORS (TAC NO. MA4920)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specification (TS) 3/4.7.7 in response to your application dated December 3, 1998.

The amendment would revise TS 3/4.7.7, "Sealed Source Contamination," and its associated bases to address testing requirements for fission detectors.

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

Sincerely,

Original signed by:

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 235 to DPR-58
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 20, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGE, "SEALED SOURCE CONTAMINATION," AND ITS ASSOCIATED BASES TO ADDRESS TESTING REQUIREMENTS FOR FISSION DETECTORS (TAC NO. MA4920)

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The amendment would revise TS 3/4.7.7, "Sealed Source Contamination," and its associated bases to address testing requirements for fission detectors.

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

Sincerely,

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No.235 to DPR-58
2. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

Jeremy J. Euto, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630 CPH Mailroom
Lansing, MI 48909-8130

Gordon Arent
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

DATED: December 20, 1999

AMENDMENT NO.   TO FACILITY OPERATING LICENSE NO. DPR-58, DONALD C. COOK NUCLEAR PLANT, UNIT 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 235
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 December 3, 1998, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 235 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 20, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 235

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 7-26	3/4 7-26
3/4 7-27	3/4 7-27
B 3/4 7-5	B 3/4 7-5
B 3/4 7-5a	B 3/4 7-5a

3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.7.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and:

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.7.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources and fission detectors previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials.

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.7.7.1.3 Reports - A Special Report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of ≥ 0.005 microcuries of removable contamination.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The Unit 1 control room emergency ventilation system operates automatically on a Safety Injection Signal from either Unit 1 or Unit 2. The automatic start from Unit 2 is only available when the Unit 2 ESF actuation system is active in modes 1 through 4 in Unit 2.

The control room ventilation system normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that adequate cooling is provided for ECCS equipment and that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the ESF ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.8 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.235TO 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 letter dated December 3, 1998, the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1. The proposed amendment would revise TS 3/4.7.7, "Sealed Source Contamination," and its associated bases to address testing requirements for fission detectors. The proposed changes would provide consistency with the Unit 2 TS requirements and NUREG-0452, "Standard Technical Specifications." The purpose of the requirement for leak testing is to limit the amount of removable contamination that is available for intake and to ensure that occupational dose limits are not exceeded.

2.0 EVALUATION

The licensee proposes to revise Unit 1 TS 3/4.7.7 related to sealed source contamination testing. The TS change would specifically address testing requirements for fission detectors to make them consistent with the Unit 2 TS requirements and with NUREG-0452, "Standard Technical Specifications." Specifically, Surveillance Requirement 4.7.7.1.2.a, "Sources in Use," is revised to exclude fission detectors previously subjected to core flux. Sections 4.7.7.1.2.b and c, "Stored sources not in use," and "Startup sources," would be revised to specifically include fission detectors. These changes would require fission detectors to be tested prior to use or transfer to another licensee unless the detector was tested during the previous 6 months. Similar to startup sources, testing of fission detectors would be required 31 days prior to being subjected to core flux and following repair or maintenance of a source. Since fission detectors currently in use are subjected to a core flux even when the reactor is shut down, the TS changes do not introduce any new requirements for testing unless a new fission detector is used. Section 4.7.7.1.3 would be revised to replace the word "detection" with "detector." In addition, Bases pages B 3/4 7-5 and 7-5a would be revised to delete wording related to obsolete regulations and to use more generic wording.

Identification of inconsistencies between the language found in the TSs for Units 1 and 2 regarding testing of fission detectors prompted the licensee to propose changes to the Unit 1 TSs based on NUREG-0452, "Standard Technical Specifications." Although the Unit 1 TSs do not specifically address testing of fission detectors, testing of fission detectors for both units is currently in conformance with 10 CFR 70.39c, utilizing an activity limit of .005 µCi of removable contamination detected from a dry wipe test. Although 10 CFR 70.39c is based on transfer of sealed sources containing plutonium, there are no specific sealed source leakage limits for uranium. Plutonium and uranium are both alpha emitters that pose an inhalation and ingestion hazard. The activity limit for sealed source leakage based on plutonium will ensure that total body and individual organ dose limits will not be exceeded for uranium intake. Since uranium-235 has a half-life of 7.13×10^8 years and uranium-238 has a half-life of 4.5×10^9 years, the activities of these long-lived radionuclides will not change significantly within 6 months. Therefore, the proposed changes to the Unit 1 TSs provide assurance that leakage of fission detectors will be discovered prior to exposure of a worker and that removable surface contamination will not pose a radiological health hazard due to leaking of uranium from the detector. The proposed changes to the TS are acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 (64 FR 43773). Accordingly, this 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 this amendment.

5.0 CONCLUSION

The 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Sochor

Date: December 20, 1999

December 20, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI. 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGE, "SEALED SOURCE CONTAMINATION," AND ITS ASSOCIATED BASES TO ADDRESS TESTING REQUIREMENTS FOR FISSION DETECTORS (TAC NO. MA4920)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specification (TS) 3/4.7.7 in response to your application dated December 3, 1998.

The amendment would revise TS 3/4.7.7, "Sealed Source Contamination," and its associated bases to address testing requirements for fission detectors.

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

Sincerely,

Original signed by:

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 235 to DPR-58
2. Safety Evaluation

cc w/encls: See next page

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Body:

Docket: 05000315, Notes: N/A

CP-1

December 20, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE DPR-58 (TAC NO. MA4920)

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: December 3, 1998

Brief description of amendment: This amendment revised the Technical Specifications for sealed source leakage testing to specifically address testing requirements for fission detectors.

Date of issuance: December 20, 1999

Effective date: December 20, 1999 with full implementation within 45 days

Amendment No.: 235

Facility Operating License No. DPR-58: Amendment revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: August 11, 1999 (64 FR 43773)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085

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December 20, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE DPR-58 (TAC NO. MA4920)

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: December 3, 1998

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 20, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE DPR-58 (TAC NO. MA4920)

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook

Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: December 3, 1998

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No significant hazards consideration comments received: No.

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50-315/316

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 13, 1999

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: ISSUANCE OF AMENDMENTS - DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA6766 AND MA6767)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-58 and Amendment No. 217 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 1, 1999, as supplemented November 19, 1999.

The amendments involve the resolution of an unreviewed safety question related to certain small-break loss-of-coolant accident scenarios for which there may not be sufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system and containment spray system pumps during and following switchover to cold leg recirculation. Resolution of this issue consists of a combination of physical plant modifications, new analyses of containment recirculation sump inventory, and resultant changes to the accident analyses to ensure sufficient water inventory in the containment recirculation sump. The amendments would also change the TSs dealing with the refueling water storage tank inventory and temperature, the required amount of ice in each ice basket in the containment, and the delay to start the containment air recirculation/hydrogen skimmer fans.

The proposed TSs involving the removal of the word "Each" in Sections 3.6.5.1.d and 4.6.5.1.b.2 will be evaluated and issued in separate correspondence.

The supporting documentation for the proposed TSs also addresses issue number 1 in the Confirmatory Action Letter issued by the NRC on September 19, 1997. Final resolution of this issue will be reported in a future inspection report.

Amendment No. 216 for Unit 1 and Amendment No. 200 for Unit 2 contain TS pages that are affected by the enclosed Amendment Nos. 234 and 217. Since Amendment Nos. 216 and 200 may not be implemented until December 31, 2000, the NRC is issuing two sets of TS pages with the enclosed amendments. The first set should be inserted when Amendment Nos. 234 and

PDR ADOCK

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217 are implemented. The second set replaces the Amendment Nos. 216 and 200 pages that are affected by Amendment Nos. 234 and 217 and should be inserted into Amendment Nos. 216 and 200.

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

Sincerely,

John F. Stang

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 234 to DPR-58
2. Amendment No. 217 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 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

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
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

Drinking Water and Radiological Protection Division
Michigan Department of Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

217 are implemented. The second set replaces the Amendment Nos. 216 and 200 pages that are affected by Amendment Nos. 234 and 217 and should be inserted into Amendment Nos. 216 and 200.

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

Sincerely,

Original signed by:

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 234 to DPR-58
2. Amendment No. 217 to DPR-74
3. Safety Evaluation

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

Sincerely,

Original signed by:

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 234 to DPR-58
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 234
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 October 1, 1999, as supplemented November 19, 1999, 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 Facility Operating License No. DPR-58 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234, are hereby incorporated in the 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 13, 1999

ATTACHMENT 1 TO LICENSE AMENDMENT NO. 234

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 1-16	3/4 1-16
3/4 3-21b	3/4 3-21b
3/4 3-26b	3/4 3-26b
3/4 3-33b	3/4 3-33b
3/4 5-11	3/4 5-11
3/4 6-26	3/4 6-26
3/4 6-27	3/4 6-27
3/4 6-35	3/4 6-35
B3/4 5-3	B3/4 5-3
B3/4 6-4	B3/4 6-4

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 375,500 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. MANUAL					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	2/train	1/train	2/train	1, 2, 3, 4	18
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
d. Steam Line Isolation	2/steam line (1 per train)	2/steam line (1 per train)	2/operating steam line (1 per train)	1, 2, 3	20
e. Containment Air Recirculation Fan	1/train	1/train	1/train	1, 2, 3, 4	18
10. CONTAINMENT AIR RECIRCULATION FAN					
a. Manual	------------------------------- See Functional Unit 9 -------------------------------				
b. Automatic Actuation Logic	2	1	2	1, 2, 3	13
c. Containment Pressure - High	3	2	2	1, 2, 3	14*

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Manual		
a. Safety Injection (ECCS)	N.A.	N.A.
Feedwater Isolation	N.A.	N.A.
Reactor Trip (SI)	N.A.	N.A.
Containment Isolation - Phase "A"	N.A.	N.A.
Containment Purge and Exhaust Isolation	N.A.	N.A.
Auxiliary Feedwater Pumps	N.A.	N.A.
Essential Service Water System	N.A.	N.A.
b. Containment Spray	N.A.	N.A.
Containment Isolation - Phase "B"	N.A.	N.A.
Containment Purge and Exhaust Isolation	N.A.	N.A.
c. Containment Isolation - Phase "A"	N.A.	N.A.
Containment Purge and Exhaust Isolation	N.A.	N.A.
d. Steam Line Isolation	N.A.	N.A.
e. Containment Air Recirculation Fan	N.A.	N.A.
10. CONTAINMENT AIR RECIRCULATION FAN		
a. Manual	---------- See Functional Unit 9 ----------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure - High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
9. Manual					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	N.A.	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation	N.A.	N.A.	N.A.	R	1, 2, 3, 4
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	N.A.	N.A.	N.A.	R	1, 2, 3, 4
d. Steam Line Isolation	N.A.	N.A.	Q	R	1, 2, 3
e. Containment Air Recirculation Fan	N.A.	N.A.	N.A.	R	1, 2, 3, 4
10. CONTAINMENT AIR RECIRCULATION FAN					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure - High	S	R	M(3)	N.A.	1, 2, 3

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 375,500 gallons of borated water.

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 70°F and a maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.6.5 ICE CONDENSER

ICE BED

LIMITING CONDITION FOR OPERATION

3.6.5.1 The ice bed shall be OPERABLE with:

a. The stored ice having boron concentration of at least 1800 ppm (the boron being in the form of sodium tetraborate), and a pH of 9.0 to 9.5 at 25°C,

b. Flow channels through the ice condenser,

c. A maximum ice bed temperature of ≤ 27°F,

d. Each ice basket containing at least 1144 lbs of ice (end-of-cycle), and

e. 1944 ice baskets.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ice bed inoperable, restore the ice bed to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 The ice condenser shall be determined OPERABLE:

a. At least once per 12 hours by using the ice bed temperature monitoring system to verify that the maximum ice bed temperature is ≤ 27°F.

b. At least once per 18 months by:

1. Chemical analyses which verify that at least 9 representative samples of stored ice have a boron concentration of at least 1800 ppm (the boron being in the form of sodium tetraborate), and a pH of 9.0 to 9.5 at 25°C.

2. Weighing a representative sample of at least 144 ice baskets and verifying that each ice basket contains at least 1144 lbs of ice (end-of-cycle). The representative sample shall include 6 baskets from each of the 24 ice condenser bays and

SURVEILLANCE REQUIREMENTS (Continued)

shall be constituted of one basket each from Radial Rows 1, 2, 4, 6, 8 and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1144 pounds of ice (end-of-cycle), a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1144 pounds/basket (end-of-cycle) at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - bays 1 through 8, Group 2 - bays 9 through 16, and Group 3 - bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8 and 9 in each group shall not be less than 1144 pounds/basket (end-of-cycle) at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,222,000 pounds (end-of-cycle).

3. Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on the top deck floor grating, on the intermediate deck and on flow passages between ice baskets and past lattice frames is restricted to a nominal thickness of 3/8 inches. If one flow passage per bay is found to have an accumulation of frost or ice greater than this thickness, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 18 months by verifying, by a visual inspection, each ice condenser bay, that the accumulation of frost or ice on the lower inlet plenum support structures and turning vanes is restricted to a nominal thickness of 3/8 inches. An accumulation of frost and ice greater than this thickness is evidence of abnormal degradation of the ice condenser.

d. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each 1/3 of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

CONTAINMENT AIR RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent containment air recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment air recirculation system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each containment air recirculation system shall be demonstrated OPERABLE at least once per 3 months on a STAGGERED TEST BASIS by:

a. Verifying that the return air fan starts on an auto-start signal after a 120 ± 12 seconds delay, the motor operated valve in the suction line to the containment's lower compartment opens when the return air fan starts, and the return air fan operates for at least 15 minutes (applicable in MODES 1, 2, and 3 only),

b. Verifying that with the return air fan discharge backdraft damper locked closed and the fan motor energized, the static pressure between the fan discharge and the backdraft damper is ≥ 4.0 inches, water gauge,

c. Verifying that with the fan off, the return air fan damper opens when a force of ≤ 11 lbs is applied to the counterweight, and

d. Verifying that the return air fan can be manually started from the control room, and the motor operated valve in the suction line to the containment's lower compartment opens when the return air fan starts.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. Consistent with the applicable LOCA analyses, the limits on RWST minimum volume and boron concentration ensure that 1) when combined with water from melted ice, the RCS, and the accumulators, sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

The ECCS and containment integrity analyses assumed a maximum RWST water temperature above 100°F. Maintaining RWST water temperature at or below 100°F ensures the containment spray system will provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig, and that containment cooling will be maintained following a LOCA or steam line rupture inside containment.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA, 4) contain sufficient water to maintain adequate sump inventory, and 5) result in a post-LOCA sump pH within the allowed range. These conditions are consistent with the assumptions used in the accident analyses.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a design basis accident and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators.

Over the course of a fuel cycle, sublimation reduces the weight of ice in the ice condenser. For the ice condenser to be considered OPERABLE, the minimum as-found ice weight of 1144 pounds per ice basket, for those ice baskets selected for weighing per the surveillance requirements, must be present at the end of a fuel cycle. An instrument measurement error allowance is included in the required minimum ice basket weight. To account for loss due to sublimation, a conservative average ice bed sublimation of 10% over an eighteen-month period is used. The beginning-of-cycle, or as-left ice basket weight, is adjusted accordingly to assure the LCO limit will be met at the end of each fuel cycle.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the ice bed temperature monitoring system ensures that the capability is available for monitoring the ice temperature. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 234

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of Amendment No. 216 with the attached revised page. The page replaces the Amendment No. 216 pages that is affected by the issuance of the enclosed Amendment No. 234. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 1-16	3/4 1-16

BORATED WATER SOURCES - OPERATIONS

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum usable borated water volume of 8,500 gallons,*

2. Between 6,550 and 6,990 ppm of boron, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 375,500 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

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* Not required when borated water is injected into the RCS to meet SHUTDOWN MARGIN requirements of MODES 3 and 4.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 217
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 October 1, 1999, as supplemented November 19, 1999, 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 Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 217, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of it's date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 13, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 217

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 1-16	3/4 1-16
3/4 3-20	3/4 3-20
3/4 3-20a	3/4 3-20a
3/4 3-25b	3/4 3-25b
3/4 3-32	3/4 3-32
3/4 5-11	3/4 5-11
3/4 6-35	3/4 6-35
3/4 6-36	3/4 6-36
3/4 6-44	3/4 6-44
B3/4 5-3	B3/4 5-3
B3/4 6-4	B3/4 6-4

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum usable borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

a. The refueling water storage tank with:

1. A minimum contained borated water volume of 375,500 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 Kv Bus Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
9. MANUAL					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	2/train	1/train	2/train	1, 2, 3, 4	18
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
d. Steam Line Isolation	2/steam line (1 per train)	2/steam line (1 per train)	2/operating steam line (1 per train)	1, 2, 3	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
e. Containment Air Recirculation Fan	1/train	1/train	1/train	1, 2, 3, 4	18
10. CONTAINMENT AIR RECIRCULATION FAN					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	2	1	2	1, 2, 3	13
c. Containment Pressure - High	3	2	2	1, 2, 3	14*

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Manual		
a. Safety Injection (ECCS)	N.A.	N.A.
Feedwater Isolation	N.A.	N.A.
Reactor Trip (SI)	N.A.	N.A.
Containment Isolation - Phase "A"	N.A.	N.A.
Containment Purge and Exhaust Isolation	N.A.	N.A.
Auxiliary Feedwater Pumps	N.A.	N.A.
Essential Service Water System	N.A.	N.A.
b. Containment Spray	N.A.	N.A.
Containment Isolation - Phase "B"	N.A.	N.A.
Containment Purge and Exhaust Isolation	N.A.	N.A.
c. Containment Isolation - Phase "A"	N.A.	N.A.
Containment Purge and Exhaust Isolation	N.A.	N.A.
d. Steam Line Isolation	N.A.	N.A.
e. Containment Air Recirculation Fan	N.A.	N.A.
10. CONTAINMENT AIR RECIRCULATION FAN		
a. Manual	------------------------------ See Functional Unit 9	---------------------------
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure - High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	S	R	M	N.A.	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	N.A.	1, 2, 3
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3, 4
b. 4 kv Bus Degraded Voltage	S	R	M	N.A.	1, 2, 3, 4
9. Manual					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	N.A.	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation	N.A.	N.A.	N.A.	R	1, 2, 3, 4
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	N.A.	N.A.	N.A.	R	1, 2, 3, 4
d. Steam Line Isolation	N.A.	N.A.	Q	R	1, 2, 3
e. Containment Air Recirculation Fan	N.A.	N.A.	N.A.	R	1, 2, 3, 4
10. CONTAINMENT AIR RECIRCULATION FAN					
a. Manual	------ See Functional Unit 9 ------				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure - High	S	R	M(3)	N.A.	1, 2, 3

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 375,500 gallons of borated water.

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 70°F and a maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.6.5 ICE CONDENSER

ICE BED

LIMITING CONDITION FOR OPERATION

3.6.5.1 The ice bed shall be OPERABLE with:

a. The stored ice having boron concentration of at least 1800 ppm (the boron being in the form of sodium tetraborate), and a pH of 9.0 to 9.5 at 25°C,

b. Flow channels through the ice condenser,

c. A maximum ice bed temperature of ≤ 27°F,

d. Each ice basket containing at least 1144 lbs of ice (end-of-cycle), and

e. 1944 ice baskets.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ice bed inoperable, restore the ice bed to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 The ice condenser shall be determined OPERABLE:

a. At least once per 12 hours by using the ice bed temperature monitoring system to verify that the maximum ice bed temperature is ≤ 27°F.

b. At least once per 18 months by:

1. Chemical analyses which verify that at least 9 representative samples of stored ice have a boron concentration of at least 1800 ppm (the boron being in the form of sodium tetraborate), and a pH of 9.0 to 9.5 at 25°C.

2. Weighing a representative sample of at least 144 ice baskets and verifying that each ice basket contains at least 1144 lbs of ice (end-of-cycle). The representative sample shall include 6 baskets from each of the 24 ice condenser bays and

SURVEILLANCE REQUIREMENTS (Continued)

shall be constituted of one basket each from Radial Rows 1, 2, 4, 6, 8 and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1144 pounds of ice (end-of-cycle), a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1144 pounds/basket (end-of-cycle) at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - bays 1 through 8, Group 2 - bays 9 through 16, and Group 3 - bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8 and 9 in each group shall not be less than 1144 pounds/basket (end-of-cycle) at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,222,000 pounds (end-of-cycle).

3. Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on the top deck floor grating, on the intermediate deck and on flow passages between ice baskets and past lattice frames is restricted to a nominal thickness of 3/8 inches. If one flow passage per bay is found to have an accumulation of frost or ice greater than this thickness, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 18 months by verifying, by a visual inspection, each ice condenser bay, that the accumulation of frost or ice on the lower plenum support structures and turning vanes is restricted to a nominal thickness of 3/8 inches. An accumulation of frost or ice greater than this thickness is evidence of abnormal degradation of the ice condenser.

d. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each 1/3 of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

CONTAINMENT AIR RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent containment air recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment air recirculation system inoperable, restore the inoperable system to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each containment air recirculation system shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Verifying that the return air fan starts on an auto-start signal after a 120 ± 12 seconds delay, the motor operated valve in the suction line to the containment's lower compartment opens when the return air fan starts, and the return air fan operates for at least 15 minutes (applicable in MODES 1, 2, and 3 only).

b. Verifying that with the return air fan discharge backdraft damper locked closed and the fan motor energized, the static pressure between the fan discharge and the backdraft damper is ≥ 4.0 inches, water gauge.

c. Verifying that with the fan off, the return air fan damper opens when a force of ≤ 11 lbs is applied to the counterweight.

d. Verifying that the return air fan can be manually started from the control room, and the motor operated valve in the suction line to the containment's lower compartment opens when the return air fan starts.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or a steam line rupture. Consistent with the applicable LOCA analyses, the limits on RWST minimum volume and boron concentration ensure that 1) when combined with water from melted ice, the RCS, and the accumulators, sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following a LOCA assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

The ECCS and containment integrity analyses assumed a maximum RWST water temperature above 100°F. Maintaining RWST water temperature at or below 100°F ensures the containment spray system will provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig, and that containment cooling will be maintained following a LOCA or steam line rupture inside containment.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: 1) zirconium-water reactions; 2) radiolytic decomposition of water; and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The acceptance criterion of 10,000 ohms is based on the test being performed with the heater element at an ambient temperature, but can be conservatively applied when the heater element is at a temperature above ambient.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA, 4) contain sufficient water to maintain adequate sump inventory, and 5) result in a post-LOCA sump pH within the allowed range. These conditions are consistent with the assumptions used in the accident analyses.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a design basis accident and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators.

Over the course of a fuel cycle, sublimation reduces the weight of ice in the ice condenser. For the ice condenser to be considered OPERABLE, the minimum as-found ice weight of 1144 pounds per ice basket, for those ice baskets selected for weighing per the surveillance requirements, must be present at the end of a fuel cycle. An instrument measurement error allowance is included in the required minimum ice basket weight. To account for loss due to sublimation, a conservative average ice bed sublimation of 10% over an eighteen-month period is used. The beginning-of-cycle, or as-left ice basket weight, is adjusted accordingly to assure the LCO limit will be met at the end of each fuel cycle.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the ice bed temperature monitoring system ensures that the capability is available for monitoring the ice temperature. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 217

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of Amendment No. 200 with the attached revised page. These pages replace Amendment No. 200 pages that are affected by the issuance of the enclosed Amendment No.   . The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 1-16	3/4 1-16

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 8500 gallons,*

2. Between 6,550 and 6,990 ppm of boron, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 375,500 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

*Not required when borated water is injected into the RCS to meet SHUTDOWN MARGIN requirements of MODES 3 and 4.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated October 1, 1999, as supplemented November 19, 1999, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments involve the resolution of an unreviewed safety question related to certain small-break loss-of-coolant accident (LOCA) scenarios for which there may not be sufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system (ECCS) and containment spray system pumps during and following switchover to cold leg recirculation. Resolution of this issue consists of a combination of physical plant modifications, new analyses of containment recirculation sump inventory, and resultant changes to the accident analyses to ensure sufficient water inventory in the containment recirculation sump. The amendments would also change the TSs dealing with the refueling water storage tank inventory and temperature, the required amount of ice in each ice basket in the containment, and the delay to start the containment air recirculation/hydrogen skimmer fans.

The proposed TSs involving the removal of the word "Each" in Sections 3.6.5.1.d and 4.6.5.1.b.2 will be evaluated and issued in separate correspondence.

The licensee's November 19, 1999, letter provided information inadvertently left out of the October 1, 1999, application. The November 19, 1999, letter did not alter the scope of the application or the staff's initial proposed no significant hazards determination.

2.0 BACKGROUND

Upon indications of a LOCA in a pressurized water reactor such as Donald C. Cook Nuclear Plant (DC Cook) Unit 1 or Unit 2, water is injected into the reactor vessel to make up for coolant expelled from the break and to cool the core. The source for this water is the refueling water storage tank (RWST). When the RWST water level reaches a specified set point, the water source is transferred to the containment recirculation sump and the emergency core cooling system (ECCS) and containment spray (CTS) pumps continue to supply the reactor vessel and the containment atmosphere from this source. The water inventory for the containment

recirculation sump consists of water from the RWST, the reactor vessel, the safety injection accumulators and melted ice from the ice condensers which are designed to absorb the energy released as a result of the LOCA.

On September 8, 1997, the licensee confirmed that there may not be sufficient water in the containment recirculation sump to ensure that the ECCS and CTS pumps would not entrain air by vortexing to the extent that performance of these pumps may be degraded for certain small-break LOCAs. The licensee therefore shutdown both units in compliance with TS requirement 3.0.3.

By letter dated October 21, 1997 (Reference 3), the licensee requested exigent changes to the TSs to ensure adequate water inventory in the containment recirculation sump to preclude vortexing. This was done by increasing the minimum required mass of ice in each ice condenser basket. In addition, the fraction of ice sublimation during a cycle was revised from 10% to 5%. The licensee considered the change to the sublimation fraction to be an unreviewed safety question as defined in 10 CFR 50.59. An analysis of the water inventory in the containment following a limiting small-break LOCA supported this change. An NRC safety evaluation report dated January 2, 1998 (Reference 4), approved these changes to the TSs. These changes were designated as Amendment 220 for Unit 1 and 204 for Unit 2. Both DC Cook units have been shut down since these license amendments were approved and have not operated with the revised TSs.

In a March 17, 1999, (Reference 5) letter from the licensee to the NRC, the licensee identified a potentially nonconservative assumption in the analysis supporting Amendments 220 for Unit 1 and 204 for Unit 2.

By letter dated October 1, 1999 (Reference 1), which is the subject of this review, the licensee requested changes to the TSs with an accompanying new analysis, to correct the earlier deficiencies and ensure that the ECCS and CTS pump vortexing limit is satisfied. This submittal was supplemented by additional proprietary technical information in a letter dated November 19, 1999 (Reference 2).

Attachment 6 of the licensee's October 1, 1999, submittal provides a more detailed chronology of the events which led to the need for the proposed TS changes.

The following TSs and bases changes are proposed in the licensee's October 1, 1999, letter:

(1) The available RWST water inventory will be increased from 350,000 gallons to 375,500 gallons.

(2) The maximum temperature for the RWST water will be limited to 100°F.

(3) The actuation signal for the containment air recirculation/hydrogen skimmer (CEQ) fans will be changed from containment pressure-high-high to containment pressure-high and the time delay for the CEQ fan start will be reduced from 9±1 minutes to 120±12 seconds.

(4) The weight of ice is decreased from 1333 pounds to 1144 pounds per basket. (This includes a 1% uncertainty for weight measurement.)

(5) The ice weight (both per basket and total) will be required to be within specified limits at the end of the cycle rather than at the beginning of the cycle as currently required.

(6) The sublimation fraction will be increased from 5% to 10%. This is a change to the TS Bases. The staff considers this to be an unreviewed safety question, as defined in 10 CFR 50.59. This is consistent with the licensee's treatment of a similar change in the licensee's October 21, 1997, submittal (Reference 3).

These changes are supported by analyses which demonstrate that the water level in the containment recirculation sump will remain above the vortexing limit and analyses which demonstrate that the other relevant criteria of the DC Cook licensing basis are satisfied.

The water level analyses were performed by the licensee using the Modular Accident Analysis Program (MAAP) 4.0.4 computer program. MAAP was originally developed as part of the Industry Degraded Core Rulemaking (IDCOR) program to study damaged core, primary system and containment failure scenarios. The licensee stated that this code was chosen because of its ability to model the reactor and containment systems together in an integrated way.

MAAP has not been previously approved by the NRC and has not been reviewed for acceptability for use in licensing calculations as part of this review. Instead, the staff has relied on comparisons supplied by the licensee of MAAP with experimental data and with other computer codes to demonstrate the capability of MAAP to predict ice condenser behavior. In addition, the NRC staff requested the Los Alamos National Laboratory (LANL) to perform some independent calculations to assess the licensee's results.

The licensee provided WCAP-15302, "Donald C. Cook Nuclear Plant Units 1 and 2 Modifications to the Containment Systems Westinghouse Safety Evaluation," dated September 1999 as Attachment 10 to the licensee's October 1, 1999, letter to demonstrate compliance with other criteria related to accident and transient analysis in the DC Cook licensing basis.

In addition to these TS changes, the licensee's October 1, 1999, submittal discussed several other changes that will be made to the plant. These include:

(1) Containment water level instrumentation will be improved to provide operators with more accurate level indication during the switchover from injection (pump suction from the RWST) to recirculation (pump suction from the containment recirculation sump).

(2) Penetrations will be made in a wall separating the pipe annulus region from the reactor coolant system (RCS) loop compartment to allow water to flow freely between these areas (see Figure 1 of Attachment 6 to the October 1, 1999, letter).

(3) Elbows and a vertical section of pipe will be added to the RWST to increase the RWST overflow height.

(4) The drain lines in the CEQ fan room will be rerouted and the check valves in the lines will be replaced.

As stated above, the purpose of the changes proposed by the licensee to the TSs is to ensure that the containment water level following a LOCA is high enough so that vortexing and concomitant ingestion of air by the ECCS and CTS pumps can be prevented. This level is 602 feet 10 inches, which is approximately 4 feet above the containment floor. This level was shown to be adequate by hydraulic testing sponsored by the licensee and conducted at Alden Research Laboratory. Reference 6 describes the tests that were performed and their results. These tests were performed to satisfy Unit 2 License Condition 2.c.(3)(H). By letter dated July 2, 1982, the NRC concluded that the license adequately responded to this license condition and found the sump testing performed by the licensee to be acceptable.

3.0 EVALUATION

3.1 Changes to the TSs

3.1.1 TS 3/4.1.2.8 "Borated Water Sources-Operating" TS 3/4.5.5 "Refueling Water Storage Tank"

The minimum RWST inventory requirement will be increased from 350,000 gallons to 375,500 gallons. The licensee stated that the change will be implemented by modifying the RWST overflow line. Increasing the RWST water inventory results in more water being available to the containment recirculation sump to prevent vortexing and air entrainment in the ECCS and CTS pumps. The licensee's calculation of water level in containment following LOCAs of various sizes and locations, discussed in Attachment 7 of the October 1, 1999, submittal, demonstrates that the increase in RWST water inventory (together with other changes) is sufficient to ensure that the vortexing limit of 602-feet 10-inches is met or exceeded.

The licensee has also proposed adding a requirement to the TSs to limit the maximum allowable RWST water temperature of 100°F. The licensee has performed design basis LOCA calculations as well as containment integrity (maximum containment pressure and temperature) calculations using a water temperature of 105°F. Therefore, the 100°F RWST water temperature is bounded. For the water inventory calculations, the lower RWST water temperature limit of 70°F is conservative since it increases the containment spray's ability to remove heat and thereby reduces the amount of ice that is melted during a LOCA.

The licensee performed LOCA calculations in order to demonstrate that the different conditions at switchover to recirculation due to the increased RWST inventory (water temperature, boron concentration) and the timing of the switchover do not adversely affect other safety limits such as the ECCS criteria of 10 CFR 50.46 and pH concentration of the coolant in the sump.

The licensee also reported the results of RWST drain down calculations which demonstrate that the timing of the switchover from injection to recirculation is adequate to ensure a sufficient water source for adequate flow for both the ECCS and CTS pumps. The licensee indicated that the calculations also demonstrate that adequate injection will continue during the switchover from the injection phase to the recirculation phase of the LOCA. The staff finds this to be acceptable.

3.1.2 TS Table 3.3-3 "Engineered Safety Feature Actuation System Instrumentation"
TS Table 3.3-4 "Engineered Safety Feature Actuation System Instrumentation Trip Set Points"
TS Table 4.3-2 "Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements"
TS 3/4.6.5.6, "Containment Air Recirculation Systems"

The current TSs do not specify automatic actuation of the CEQ fans and valves. Instead, they are actuated automatically with a delay time of 9±1 minutes after receipt of a containment spray automatic actuation signal (on Containment Pressure-High-High in MODES 1,2, and 3 at 2.9 psig). The licensee proposes to change the initiating signal and timing so that the CEQ fans will start and the hydrogen skimmer valves will start to open 120±12 seconds after a containment pressure-high signal (at a containment pressure equal to 1.1 psig).

The purpose of this change is to start the CEQ fans in order to increase the rate of ice melting in the containment. Increasing the rate of ice melting in containment is conservative for the water inventory calculations since it increases the steam flow into the ice condenser bays which promotes more rapid ice melting. However, it is nonconservative from the perspective of design basis peak containment pressure and temperature calculations. The licensee has performed new calculations (described in Attachment 10 of the licensee's October 1, 1999, letter) of peak containment pressure and temperature following the most conservative large break LOCA and has demonstrated that the peak containment pressure and temperatures remain below the containment design values. Therefore, the staff finds this change to the initiating signals and timing of the CEQ fans and hydrogen skimmer valves to be acceptable.

The current TSs specify that manual actuation of the CEQ fans is accomplished as part of the containment spray manual actuation requirements which are applicable in MODES 1 through 4. The proposed changes to the TS Table 3.3-4 and TS Table 4.3-2 specify that manual actuation of the CEQ fans and valves is separate from the containment spray manual actuation requirements. This is acceptable since suitable controls are provided in the control room to operate the CEQ fans independent of the containment spray controls. The surveillance requirements for manual actuation have been changed to be consistent with the new method of CEQ fan actuation.

3.1.3 TS 3/4.6.5.1 Ice Condenser

The purpose of this TS is to ensure that the amount of ice available will provide sufficient pressure suppression to maintain the peak containment pressure following a design basis accident below the containment design pressure. In addition, the water from the melting of all or part of this ice, when combined with water from the reactor coolant system, safety injection accumulators, and the RWST will be sufficient to ensure that the ECCS and CTS pumps will operate above the vortexing level criterion during recirculation following a LOCA.

The current weight of ice specified in this TS for each basket is 1333 pounds. Since there are 1944 baskets in the DC Cook design, this is a total of 2.59 million pounds of ice. This weight is the beginning-of-cycle (BOC) or "as-left" value, and includes a 1% uncertainty for weighing and a 5% factor for sublimation during the 18-month cycle which decreases the ice weight.

The licensee is proposing to reduce the minimum required weight of ice in the DC Cook ice condensers. The licensee states that the decrease in ice weight will "facilitate effective management of the ice inventory for the ice condenser and ...facilitate ice condenser maintenance."

The total ice condenser ice weight will be reduced to a nominal 2.2 million pounds or 1132 pounds of ice per basket. This value was used in the DC Cook safety analyses discussed in the October 1, 1999 letter. The licensee further proposes that this value be an end-of-cycle (EOC) value or an "as found" value. It therefore must be increased by the 1% weighing uncertainty to the proposed TS limit of 1144 pounds per basket. The sublimation factor is not included in this value since the sublimation process would be complete at the time of the surveillance.

TS Bases Section 3/4.6.5.1 discusses adjusting the ice weight or the number of baskets to be weighed depending on accumulated data from several cycles. However, since the value in the TSs cannot be changed without NRC approval, this provision is moot and the licensee proposes to eliminate it from the bases. The staff concurs.

The licensee has proposed modifying the Bases to increase the sublimation uncertainty from 5% to 10%. The staff considers this change to be an unreviewed safety question as defined in 10 CFR 50.59 which is consistent with the licensee's handling of this issue in the October 27, 1997, submittal to the NRC. Although some other ice condenser plants have larger values for the allowance for sublimation, a review of industry data shows that actual overall sublimation does not exceed the 10% value. The staff therefore finds the 10% allowance to be acceptable.

The licensee proposed changing TS 3.6.5.1.d to state that the ice bed shall be OPERABLE with:

> <u>ice baskets</u> containing at least 1144 lbs of ice (end-of-cycle)

rather than

> <u>each ice basket</u> containing at least 1144 lbs of ice (end-of-cycle)

The licensee states that the purpose of this change is to clarify that the as-found weight applies to all of the ice baskets weighed. However, a similar change to the ice condenser TSs was proposed to the NRC by TVA for Sequoyah Units 1 and 2 as the lead plant for generic changes to the ice condenser TSs and the NRC staff has questioned the implications of this wording. The proposed TSs involving the removal of the word "Each" in Sections 3.6.5.1.d and 4.6.5.1.b.2 will be evaluated and issued in separate correspondence.

For both the calculation of water inventory in the containment recirculation sump following a LOCA and the peak pressure and temperature in containment following a LOCA, it is conservative to use the minimum ice weight. The licensee has done this. Based on the results of the licensee's calculations showing that (1) the water level in the containment recirculation sump is above the vortexing limit of 602 feet 10 inches, and (2) the containment peak pressure and temperature are below the containment design limits, the staff finds the ice weight proposed by the licensee to be acceptable.

The licensee's change to an end-of-cycle surveillance of the ice weight is acceptable since it is in compliance with the definition of a limiting condition for operation defined in 10 CFR 50.36 as "the lowest functional capabilities or performance levels of equipment." The licensee will, by procedure, begin the cycle with a nominal ice weight which will be increased by the weighing uncertainty of 1% and the sublimation allowance of 10%. This does not change the process from the current procedure.

3.2 Analysis

The licensee has demonstrated that the TS changes described above are acceptable based on analyses of design basis accidents, in particular, the LOCA and the main steam line break. All relevant safety criteria of the DC Cook licensing basis will be satisfied if the reactor is operated within these limits.

3.2.1 Containment Water Level

In order to demonstrate that the ECCS and CTS pumps will operate acceptably with respect to the vortex limit, the licensee has calculated the containment water level following the most limiting LOCA and shown that this level is greater than the vortex limit of 602 feet 10 inches (see Section 2.0 of this safety evaluation).

The calculations were performed with the MAAP 4.0.4 computer program. The NRC has not reviewed the MAAP code. However, the MAAP code has been used previously by the licensee to support the October 27, 1997, request for TS changes related to ice weight. In that case, the staff approved the licensee's proposed TS changes based on comparisons between MAAP and experimental data, as well as staff calculations performed with the MELCOR computer code using input data supplied by the licensee (Reference 7). The staff used the same method in reviewing this submittal.

The MAAP code was selected by the licensee since it provides an integral calculation tool and it is not necessary to transfer data from one computer program to another. Calculations of water inventory depend on modeling of the containment as well as flows into and out of the reactor vessel, that is, the water inventory in the reactor coolant system, the amount of melted ice, the amount of water injected from the safety injection accumulators and the ECCS, the containment spray taking suction from the RWST or the recirculation sump, and the flow of water between different compartments in the containment.

The licensee compared MAAP calculations with data from three experimental studies relevant to ice condenser containments. These are described in Reference 1, Attachment 7, and Reference 2. The licensee also compared MAAP with calculations done with the licensing codes used for design basis accident analysis. This is also reported in Attachment 7 to Reference 1.

The first experimental program was the Westinghouse Waltz Mill experiments (Reference 8). Eight 36-foot long ice baskets were used to obtain data for various sized LOCAs at scaled steam delivery rates.representing a large-break LOCA blowdown, a medium-break LOCA blowdown, and a small-break LOCA blowdown. A large LOCA blowdown followed by a steam flow rate through the inlet doors representative of a post-blowdown energy release (due to decay heat) was also included. MAAP predictions of these data provide an important test of the ability of the code to predict important ice condenser phenomena such as ice melt rate and the displacement of air from the lower to the upper compartment. Of particular interest for the water inventory calculations is the small break blowdown comparison (Test F) since the limiting break for water inventory concerns is the small break. For Test F, MAAP provided reasonable agreement with the experimental results for pressure and mass of melted ice. Also of interest is the large break blowdown with decay heat steaming rate (Test K) since it provides an opportunity to model long term energy transfer in the experimental assembly. In both of these tests the comparisons with measured data were good, demonstrating that MAAP adequately modeled the energy exchange processes in the experiment.

The second experimental program was a 1991 NRC program (Reference 9) to study the behavior of aerosols in ice condensers. While the behavior of aerosols is not of concern for the DC Cook proposed TS changes, the tests also provided data from a test assembly which modeled many features of the ice condenser containment and provided useful data on ice melt and exit gas temperature. Exit gas temperature is important because it is a good indication of the exchange of energy during the tests. Some computer calculations assume a constant exit temperature based on experiment, which is satisfactory for large breaks but not suitable for smaller breaks. This program was limited since it did not model the inlet doors and was limited to behavior after blowdown. Another important feature of these experiments is that an air flow typical of one train of air recirculation fans was included in the tests. Because of the absence of inlet doors, countercurrent natural circulation occurred in tests with lower flow rates. Since this was not a concern for the DC Cook work, the licensee modeled only those tests with a higher gas flow. The higher gas flow prevented this counter current natural circulation flow by "gas flooding" (that is hold-up of what otherwise would have been a downward flow). The licensee provided comparisons with exit temperature data with a heat transfer coefficient representing natural convection and radiation as a parameter. For reasonable values of this heat transfer coefficient, the licensee predicted temperatures between the lowest and highest measured exit temperatures.

The third experimental program was an Electric Power Research Institute (EPRI) program to study the mixing of hydrogen in ice condenser containments (Reference 10). Subcompartments modeling the lower and upper compartments of an ice condenser containment were installed in a large vessel (at a scale factor of 0.3 to an ice condenser containment). This mock-up of an ice condenser containment did not contain ice. It did simulate air recirculation from the upper to the lower compartment. The experiments served as a test of the ability of the MAAP code to calculate gas and temperature distributions in different portions of the containment. The licensee's comparisons of data from these tests with MAAP showed reasonable agreement.

In addition to comparisons with experimental data, the licensee provided comparisons of MAAP with calculations performed with the LOTIC-3 (Reference 11) and NOTRUMP (Reference 12) computer programs. LOTIC-3 is a Westinghouse design basis containment code for predicting ice condenser behavior. NOTRUMP provides the mass and energy input for LOTIC-3 (that is, the mass and energy flow rates from the reactor coolant system break to the containment). Comparisons of LOTIC-3 and MAAP were made for 2-inch and 6-inch cold leg breaks. In general, the comparisons between LOTIC-3 and MAAP were as expected and the differences were explainable in terms of the differences in models between the two codes. In particular, the prediction of ice melt as a function of time was less for MAAP than for LOTIC-3. Since LOTIC-3 is used for design basis containment pressure and temperature analyses, the faster rate of ice melt in LOTIC-3 is conservative.

Table 5-1 of Attachment 7 of Reference 1 lists the break locations and sizes considered by the licensee. In all cases, the break was located in the cold leg. The location was either in the lower compartment or the reactor cavity. Section 5.1 of Reference 1 discusses the choice of these two locations. Cold leg breaks release more mass into the containment, but the energy release is less than for a hot leg or crossover leg break and consequently, the amount of melted ice is less. The staff finds the licensee's choice of break locations to be acceptable. The results of the licensee's calculations are shown in Figure 5-18 of Attachment 7, which is a plot of minimum sump level during recirculation as a function of the effective break diameter. The 1-inch break with 50% flow to the reactor cavity and 50% flow to the sump is the limiting break. (A break at the cold leg nozzle in the reactor cavity results in a flow split between the reactor cavity and the lower compartment.) The worst single failure was determined to be the loss of one CEQ fan. This results in the lowest flow of gas through the ice condenser and, therefore, the lowest rate of ice melt. On the other hand, both trains of containment spray are considered to be in operation since this minimizes the rate of ice melting.

The licensee also analyzed postulated breaks from hot standby condition, MODE 3 of the TSs. The lower range average reactor coolant system temperature of 350°F was used for the analyses since this results in minimum steam production due to flashing and therefore minimum ice melting. The break sizes analyzed were sufficient to actuate the containment sprays and CEQ fans. For some split flow breaks in MODE 3 there is insufficient inventory in the sump to remain above 602 feet 10 inches. The time below this limit is short. However, since the ECCS pump flow rates are reduced for these breaks, air entrainment in the pumps is not a problem. The licensee addressed this issue in Attachment 9 of Reference 1.

The licensee listed several conservatisms in the analyses done to determine the minimum water level in the lower compartment of the containment. These include:

(1) The volume of internal equipment in the lower compartment was neglected. The licensee estimates that if this equipment were included in the calculation sump water level, the level would be increased by approximately 2.2 inches.

(2) A maximum cool down rate of 100° F/hr was assumed following the initiation of the accident. A slower cool down rate would increase the energy discharged to the containment and, hence, more ice would melt. The licensee did not quantify the increase in ice melt which would result from a slower cool down rate.

(3) MAAP uses the assumption that the steam and water discharged from the analyzed breaks are in equilibrium. This results in the maximum water enthalpy and therefore the minimum steam mass is produced as the mixture flashes in the containment. The licensee did not quantify this effect.

(4) The licensee has assumed that the vortex limit, 602 feet 10 inches, remains constant, even as the break flow, and consequently the demand for ECCS pump flow decreases as the break size decreases. The licensee states that for the limiting break sizes, the flow is significantly less than that used in the hydraulic tests used to determine the vortex limit.

3.2.2 <u>Containment Integrity Analyses</u>

The licensee's previous analysis of record for containment integrity analyses is Reference 13. This was approved by the staff in a safety evaluation dated March 13, 1997 (Reference 14).

Section 3.4.2 of the Reference 15 lists several changes to the plant and the input assumptions to the analyses of Reference 13 which affect the containment integrity calculations. Some of these changes increased the peak containment pressure and others tended to decrease the peak containment pressure. Changes made which improve the heat removal capability of the containment deal with increasing the heat transfer from the containment. These include increasing the UA of heat exchangers and decreasing the emergency service water system temperature. The licensee performed new calculations using these assumptions. The result of these calculations is that the peak containment pressure following the most limiting design basis LOCA is calculated to be 11.6 psig. This includes a 0.1 psi pressure increase due to noncondensible hydrogen and a 0.1 psi increase due to leakage from the control air system.

The peak containment temperature following a design basis main steam line break is 324.7°F.

The staff has reviewed the changes made by the licensee to both the LOCA and main steam line break containment integrity analyses and finds them acceptable since they are consistent with planned plant operation and design or provide conservative assumptions for the analyses.

3.2.3 <u>LOCA Analysis</u>

10 CFR 50.46 requires that the containment pressure be minimized in the calculation of peak cladding temperature. The licensee examined the change in the starting of the CEQ fans following a large break LOCA, and concluded that the effect on the peak cladding temperature is negligible.

3.2.4 <u>Staff Independent Analysis</u>

The staff requested Los Alamos National Laboratory (LANL) to perform independent analyses using the MELCOR computer program (Reference 16) to model the containment response and RELAP5 (Reference 17) to model the mass and energy addition to the containment as a result of the LOCA. One-inch, 2-inch, and 6-inch breaks in the lower compartment were modeled. The input was derived from input developed by the licensee and provided to the staff during the staff review of the previous TS change to the ice mass (Reference 7). The staff calculations are documented in Reference 18. While the input for these calculations does not have a one-to-one correspondence with the input used by the licensee, the containment pressure, pool temperatures, fraction of ice remaining, the level above the lower compartment floor, and the timing of some key events such as RWST switchover and ice melt compare favorably with the containment response described in Attachment 7 to Reference 1 and Reference 19. In addition, LANL performed a "cold" calculation, meaning that the conditions were assumed for different parameters (such as lower power, lower RWST water temperature, lower RHR heat exchanger secondary side temperature) which tended to minimize the amount of ice melt. Even for this more extreme case, the calculations show that the water level criterion of 602 feet 10 inches is not violated. LANL has also pointed out that an ice melt of 25% is needed to provide sufficient water to meet the 602-foot 10-inch limit. LANL calculations show that this amount of ice melt is reached early in the transient.

The agreement with the licensee's calculations is favorable and adds confidence that the licensee's modeling is reasonable.

4.0 SUMMARY

The staff finds the proposed changes to the TSs for the Donald C. Cook Nuclear Plant, Units 1 and 2, to be acceptable. This approval is based on the licensee's analyses which show that all licensing criteria are satisfied. Analyses of water level in containment, performed with the MAAP code, are acceptable, based on comparisons of the MAAP code with relevant experimental data and approved computer codes and independent analyses performed by the LANL for the staff.

The results of other analyses demonstrate that other relevant licensing criteria (such as containment peak temperature and pressure and the criteria of 10 CFR 50.46) remain satisfied, even with some assumptions different from those used previously. The calculations were done with NRC-approved methods.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 58458). Accordingly, the amendments meet 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 amendments.

7.0 CONCLUSION

The 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter to U.S. Nuclear Regulatory Commission from R. P. Powers, Vice President, American Electric Power, Technical Specification Change Request Containment Recirculation Sump Water Inventory, October 1, 1999.

2. Letter from M.W. Rencheck, Vice President, Indiana Michigan Power Company, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Technical Specification Change Request Containment Recirculation Sump Water Inventory," November 19, 1999.

3. Letter to U.S. Nuclear Regulatory Commission from E.E. Fitzpatrick, Indiana Michigan Power Company, "Request for Exigent Technical Specification Amendment Technical Specification 3/4.6.5 Ice Weight and Surveillance Requirement and Technical Specification 3/4.5.5 Basis Refueling Water Storage Tank Change," October 21, 1997.

4. Letter to E.E. Fitzpatrick, Indiana Michigan Power Company from John B. Hickman, U.S. Nuclear Regulatory Commission, January 2, 1998.

5. Letter to U.S. Nuclear Regulatory Commission from Michael W. Rencheck, Indiana Michigan Power Company, March 17, 1999.

6. "Hydraulic Model Investigation of Vortexing and Swirl Within a Reactor Containment Recirculation Sump, Donald C. Cook Nuclear Power Station," Alden Research Laboratory, September 1978.

7. "MAAP Input Parameters and Results Used for Donald C. Cook Nuclear Power Plant Units 1 and 2 Analyses," prepared for American Electric Power Company by Fauske and Associates, Inc., Burr Ridge, Illinois, Attached to AEP Memorandum to USNRC, October 13, 1997.

8. Salvatori, R., "Final Report: Ice Condenser Full Scale Section Test at the Waltz Mill Facility," Westinghouse Proprietary Class 2 Report, WCAP-8282, February 1974.

9. Ligotke, M.W., "Ice Condenser Aerosol Test," NUREG/CR-5768, 1991.

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REQUEST FOR PUBLICATION IN BIWEEkLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6766 AND MA6767)

Body:

Docket: 05000315, Notes: N/A

Docket: 05000316, Notes: N/A

CP-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 13, 1999

MEMORANDUM TO: Biweekly Notice Coordinator

FROM: John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. MA6766 AND MA6767)

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: October 1, 1999, as supplemented November 19, 1999

Brief description of amendments: The amendments involve the resolution of an unreviewed safety question related to certain small-break loss-of-coolant accident scenarios for which there may not be sufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system and containment spray system pumps during and following switchover to cold leg recirculation. Resolution of this issue consists of a combination of physical plant modifications, new analyses of containment recirculation sump inventory, and resultant changes to the accident analyses to ensure sufficient water inventory in the containment recirculation sump. The amendments would also change the Technical Specifications dealing with the refueling water storage tank inventory and temperature, the required amount of ice in each ice basket in the containment, and the delay to start the containment air recirculation/hydrogen skimmer fans.

Date of issuance: December 13, 1999

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 234 and 217

DF42

9912350260

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 29, 1999 (64 FR 58458)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 1999

No significant hazards consideration comments received: No.

Facility Operating License Nos. DPR-58 and DPR-74: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 29, 1999 (64 FR 58458)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 1999.

No significant hazards consideration comments received: No.

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