October 10, 1995

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, OH 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: RELOCATION OF TECHNICAL SPECIFICATION TABLES OF INSTRUMENT RESPONSE TIME LIMITS (TAC NOS. M92496 AND M92497)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 202 to Facility Operating License No. DPR-58 and Amendment No. 187 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 26, 1995.

The amendments modify TS 3/4.3.1, Reactor Trip System Instrumentation, and 3/4.3.2, Engineered Safety Feature Actuation System Instrumentation, and their accompanying Bases, to relocate the tables of response time limits to the Updated Final Safety Analysis Report (UFSAR). Your request was submitted as a TS line item improvement in accordance with Generic Letter (GL) 93-08.

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

Sincerely,

ORIGINAL SIGNED BY

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 202 to DPR-58
2. Amendment No. 187 to DPR-74
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\WPDOCS\DCCOOK\C092496.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	LA:PD31	E	PM:PD31	E	LPM	E	OGC		D:PD31	E
NAME	CJamerson		JHickman		WReckley		R Bachmann		JHannon	
DATE	9/15/95		9/18/95		9/18/95		9/19/95		10/6/95	

OFFICIAL RECORD COPY

Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, Michigan 49106

Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
1 Cook Place
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington, DC 20037

Mayor, City of Bridgman
Post Office Box 366
Bridgman, Michigan 49106

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

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 48909

Mr. S. Brewer
American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43215

DATED: October 10, 1995

AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK-UNIT 1
AMENDMENT NO. 187 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK-UNIT 2

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
C. Jamerson
J. Hickman (2)
OGC
G. Hill, IRM (4)
C. Grimes, O-11F23
J. Kennedy
ACRS
W. Kropp, RIII
SEDB

cc: Plant Service list

DF01

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 202
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated May 26, 1995, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 202, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John B. Hickman

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 10, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 202

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 3-1	3/4 3-1
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
B 2-7	B 2-7
B 3/4 3-1	B 3/4 3-1

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.*

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2

Table Intentionally Deleted

TABLE 3.3-2 (Continued)

Table Intentionally Deleted

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, CHANNEL FUNCTIONAL TEST and TRIP ACTUATING DEVICE OPERATIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-5

Table Intentionally Deleted

TABLE 3.3-5 (Continued)

Table Intentionally Deleted

TABLE 3.3-5 (Continued)

Table Intentionally Deleted

TABLE 3.3-5 (Continued)

Table Intentionally Deleted

LIMITING SAFETY SYSTEM SETTINGS

BASES

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. A time delay is incorporated in each of these trips to prevent spurious reactor trips from momentary electrical power transients.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trip is an anticipatory trip which provides reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. This trip is blocked below P-7. The open/close position trip assures a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of this trip. The functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated May 26, 1995, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 187 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 10, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 187

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 3-1	3/4 3-1
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10
3/4 3-14	3/4 3-14
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
B 2-7	B 2-7
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system 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 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.*

---

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2

Table Intentionally Deleted

TABLE 3.3-2 (Continued)

Table Intentionally Deleted

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, CHANNEL FUNCTIONAL TEST and TRIP ACTUATING DEVICE OPERATIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-5

Table Intentionally Deleted

TABLE 3.3-5 (Continued)

Table Intentionally Deleted

TABLE 3.3-5 (Continued)

Table Intentionally Deleted

TABLE 3.3-5 (Continued)

Table Intentionally Deleted

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these system is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these system is consistent with the assumptions used in the accident analyses.

Protection has been provided for main feedwater system malfunctions in MODES 3 and 4. This protection is required when main feedpumps are aligned to feed steam generators in MODES 3 and 4. The availability of feedwater isolation on high-high steam generator level terminates the addition of cold water to the steam generators in any main feedwater system malfunction. The total volume that can be added to the steam generators by the main feedwater system in MODES 3 and 4 is limited by this safeguards actuation and the fact that feedwater isolation on low T_{avg} setpoint coincident with reactor trip can only be cleared above the low-low steam generator level trip setpoint.

The restrictions associated with bypassing ESF trip functions below either P-11 or P-12 provide protection against an increase in steam flow transient and are consistent with assumptions made in the safety analysis.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be OPERABLE at all times during plant operation, but they are not required for safe shutdown of the plant.

In addition, a minimum of two in containment radiation-level monitors with a maximum range of 10^7 R/hr for photon only should be OPERABLE at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible.

Table 3.3-6 is based on the following Alarm/Trip Setpoints and Measurement Ranges for each instrument listed. For the unit vent noble gas monitors, it should be noted that there is an automatic switchover from the low/mid-range channels to the high-range channel when the upper limits of the low- and mid-range channel measurement ranges are reached. In this case there is no flow to the low- and mid-range channels from the unit vent sample line. This is considered to represent proper operation of this monitor. Therefore, if automatic switchover to the high-range should occur, and the low- and mid-range detectors are capable of functioning when flow is re-established, the low- and mid-range channels should not be declared inoperable and the ACTION statement in the Technical Specification does not apply. This is also true while purging the low- and mid-range chambers following a large activity excursion prior to resumption of low-level monitoring and establishment of a new background.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 187 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated May 26, 1995, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The proposed amendments would change the TS to modify the requirements of TS 3/4.3.1 and TS 3/4.3.2 and relocate Tables 3.3-2 and 3.3-5, which provide the response time limits for the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS) instruments, from the TS to the Updated Final Safety Analysis Report (UFSAR). The licensee has stated that the next update of the UFSAR will include these tables. The NRC provided guidance to all holders of operating licenses or construction permits for nuclear power reactors on the proposed TS changes in Generic Letter 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits," dated December 29, 1993.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. These criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36, 60 FR 36953 (July 19, 1995). In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the four criteria defined by 10 CFR 50.36, for determining whether a particular matter is required to be included in the TS limiting conditions for operation, are as follows:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(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;

(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;

(4) a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As a result, 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.

3.0 EVALUATION

The licensee has proposed changes to TS 3.3.1.1 and 3.3.2.1 to remove references to Tables 3.3-2 and 3.3-5, which are being deleted from the TS. The licensee committed to relocate the tables on response time limits to the UFSAR in the next periodic update.

Tables 3.3-2 and 3.3-5 contain the values of the response time limits for the RTS and ESFAS instruments. The limiting conditions for operation for the RTS and ESFAS instrumentation specify these systems shall be operable with the response times as specified in these tables. These limits are the acceptance criteria for the response time tests performed to satisfy the surveillance requirements of TS 4.3.1 and 4.3.2 for each applicable RTS and ESFAS trip function. These surveillances ensure that the response times of the RTS and ESFAS instruments are consistent with the assumptions for the safety analyses performed for design basis accidents and transients. The changes associated with the implementation of Generic Letter 93-08 involve only the relocation of the RTS and ESFAS response time tables but retain the surveillance requirement to perform response time testing. The UFSAR will now contain the acceptance criteria for the required RTS and ESFAS response time surveillances. Because it does not alter the TS requirements to ensure that the response times of the RTS and ESFAS instruments are within their limits, the staff has concluded that relocation of these response time limit tables from the TS to UFSAR is acceptable.

The staff's determination is based on the fact that the removal of the specific response time tables does not eliminate the requirements for the licensee to ensure that the protection instrumentation is capable of performing its safety function. Although the tables containing the specific response time requirements are relocated from the TS to the UFSAR, the licensee must continue to evaluate any changes to response time 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's review concluded that 10 CFR 50.36 does not require the response time tables to be retained in technical specifications. Requirements related to the operability, applicability, and surveillance requirements, including performance of testing to ensure response times, for RTS and ESFAS systems are retained due to those systems' importance in mitigating the consequences of an accident. However, the staff determined that the inclusion of specific response time requirements for the various instrumentation channels and components addressed by Generic Letter 93-08 was not required. The response times are considered to be an operational detail related to the licensee's safety analyses which are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions of the affected instrument or component response times, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety. Further, the response time requirements do not constitute a condition or limitation on operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, in that the ability of the RTS and ESFAS systems to perform their safety functions are not adversely impacted by the relocation of the response time tables from the TS to the UFSAR.

In addition to removing the response times from the TS, the licensee is modifying the TS Bases Sections 3/4.3.1 and 3/4.3.2 to reflect these changes and has stated that the plant procedures for response time testing include acceptance criteria that reflect the RTS and ESFAS response time limits in the tables being relocated to the UFSAR. These changes are acceptable in that they merely constitute administrative changes required to implement the TS change discussed above.

These TS changes are consistent with the guidance provided in Generic Letter 93-08 and the TS requirement of 10 CFR 50.36. The staff has determined that the proposed changes to the TS for the D.C. Cook Nuclear Plant, Units 1 and 2, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Michigan official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The 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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The 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 (59 FR 629). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: William Reckley
Janet Kennedy

Date: October 10, 1995