August 22, 1990

Docket No. 50-315

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Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-58: (TAC NO. 75892)

The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated January 31, 1990.

This amendment changes Technical Specification (TS) 3/4.7.1.5.1.b, "Steam Generator Stop Valves," to require full valve closure within 8 seconds. TS Table 3.3-5, "Engineered Safety Features Response Times," has also been changed to reflect the increased closure time.

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

Timothy Colburn, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 147 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

TGC

LA/PD31:DRSP	PM/PD31:DRSP	DCD/RP (A)D/PD31:DRSP	OGC
PShuttleworth	TColburn	RPierson	
7/11/90	7/11/90	7/13/90	8/16/90

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

August 22, 1990

Docket No. 50-315

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Indiana Michigan Power Company
c/o American Electric Power Service Corporation
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See next page

Mr. Milton Alexich
Indiana Michigan Power Company

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
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 January 31, 1990, 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. 147 , 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert C. Pierson, Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 22, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 147

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 marginal lines indicating the area of change.

REMOVE	INSERT
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 7-10	3/4 7-10

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0*/13.0#
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 48.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 13.0#/23.0##
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg} --Low-Low	
a. Safety Injection (ECCS)	≤ 15.0#/25.0##
b. Reactor Trip (from SI)	≤ 5.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	≤ 20.0#/30.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 15.0#/50.0##
h. Steam Line Isolation	≤ 13.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 13.0#/23.0##
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 11.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 10.0
d. Containment Air Recirculation Fan	≤ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

PLANT SYSTEMS

STEAM GENERATOR STOP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each steam generator stop valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one steam generator stop valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one steam generator stop valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The stop valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each steam generator stop valve that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 8 seconds on any closure actuation signal while in HOT STANDBY with T_{avg} greater than or equal to 541°F during each reactor shutdown except that verification of full closure within 8 seconds need not be determined more often than once per 92 days.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.5.3 The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 when performing PHYSICS TESTS at the beginning of a cycle provided the steam generator stop valves are maintained closed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letters dated January 31, and May 14, 1990 (Refs. 1 and 2), respectively, Indiana Michigan Power Company, the licensee for operation of Donald C. Cook Nuclear Plant, proposed changes to Technical Specifications (TS) of D. C. Cook Units 1 and 2 regarding the main steam isolation response times.

Specifically, the licensee proposed to (1) change Surveillance Requirement 4.7.1.5.1 by increasing the full closure time, upon any closure actuation signal, of each steam generator stop valve from 5 to 8 seconds; and (2) change TS Table 3.3-5, "Engineered Safety Features Response Times," by increasing by 3 seconds the response times of the steam line isolation upon reaching the set points of High Steam Flow Coincident With Low-Low T_{avg} (average reactor coolant temperature), High Steam Flow Coincident With Low Steam line Pressure, and High-High Containment Pressure, respectively. In addition, the licensee in its May 14, 1990, submittal also proposed several editorial changes to Unit 2 TS Table 3.3-5. These editorial changes use the words "less than or equal to" to replace the symbol "≤," and have no consequences.

The staff evaluation of the proposed TS changes regarding the steam generator stop valve closure time and steam line isolation response times follows.

2.0 EVALUATION

In support of the proposed TS changes, the licensee has performed an evaluation of the limiting accident scenarios which rely upon closure of the steam generator stop valves, or main steam isolation valve (MSIV), in the safety analyses to ensure that the licensing basis accident analyses remain acceptable with the 3 seconds increase in the MSIV closure time and the steam line isolation response times. These accidents are evaluated in the following sections.

2.1 Main Steam Line Break Core Response Limiting Case

The analyses of main steam line break (MSLB) core response for D. C. Cook Units 1 and 2 were performed with an assumption of a steam line isolation delay time of 11 seconds. The Unit 2 MSLB analysis was documented in Appendix B to a February 6, 1990, submittal (Ref. 3) in support of Cycle 8 reload with a transition from the ANF fuel to Westinghouse 17x17 Vantage-5 fuel. The Unit 1

MSLB was previously analyzed and documented in Section 3.3.4.13 of WCAP-11902 (Ref. 4) in support of reduced temperature and pressure operation. The limiting MSLB case was found to be the double-ended rupture located upstream of the flow restrictor with off-site power available. The results of the analysis showed that the minimum departure from nucleate boiling ratio (DNBR) remained above the minimum DNBR limit of 1.45 for the W-3 critical heat flux correlation at pressure below 1000 psia and, therefore, no fuel failure was calculated.

WCAP-11902 states that the MSLB analysis for Unit 1 assumed the MSIVs to close in less than 7 seconds from receipt of actuation signal, which is 2 seconds longer than the current TS value of 5 seconds but is shorter than the proposed TS value of 8 seconds. However, Supplement 1 to WCAP-11902 (Ref. 5), which was submitted in support of a rerating program with reduced temperature and pressure operation, amended the statement and indicated that the MSLB analysis did not specifically model the MSIV closure time. Rather, the analysis modelled only the total delay time for steam isolation by assuming 11 seconds delay from the time the High Steam Flow Coincident With Low Steam Pressure set point is reached until the time the MSIVs are fully closed. Of the 11 second total delay, 8 seconds is allocated for the closure time of the MSIVs and 3 seconds will account for signal processing and electronic delay. This is consistent with current TS where the total delay time for the set point of High Steam Flow Coincident With Low Steam Pressure is 8 seconds, with a stop valve closure time of 5 seconds and a 3 second signal processing and electronic delay. Therefore, this total delay time of 11 seconds used in the analysis supports the relaxation of the MSIV closure time from 5 to 8 seconds.

2.2 Steamline Break Mass/Energy Release

Section S-3.3.4.1 of Supplement 1 of WCAP-11902 provided a new analysis of the mass and energy release for a steam line break inside containment for both Units 1 and 2. The resulting mass and energy releases were used as input to perform containment integrity analysis documented in Section S-3.4.2.1. The analysis assumed a steam line isolation within 11 seconds after the set point is reached for either High-High Steam Flow with Low Steam Pressure or High-High Containment Pressure. THe limiting scenarios were a break size of 4.6 ft^2 occurring at 102% power with a main steam isolation failure for the double ended rupture and a break size of 0.86 ft^2 occurring at 102% power with an auxiliary feedwater run out protection failure for split rupture. The resulting mass and energy release and the corresponding peak containment temperature and pressure are bounded by those in the FSAR which was analyzed with conservatively higher reactor coolant temperature and secondary pressure. Therefore, the containment integrity remains acceptable with an increase of steam line isolation to 11 seconds.

With regard to the effects of the increased steam isolation response times on equipment qualification, the licensee indicated that Westinghouse had reanalyzed the mass and energy release for the limiting cases of MSLB outside containment with the increased 3 seconds delay in the steam line isolation, and that the

licensee had evaluated the effects of these mass and energy release rate changes on the main steam enclosure temperature. They found that the effects were minimal and remained bounded by existing thermal limits, and concluded that the instruments remained qualified.

2.3 Steam Generator Tube Rupture

In the Chapter 15 USAR steam generator tube rupture (SGTR) analysis for D. C. Cook Units 1 and 2, the primary to secondary break flow was assumed to be terminated at 30 minutes after accident initiation. The operator actions to terminate the break flow and isolate the ruptured steam generator were not explicitly modelled in the analysis. Therefore, there is no impact of an increase of the MSIV closure time to the analysis of the SGTR accident.

2.4 Loss-of-Coolant Accidents (LOCA)

In the licensing basis LOCA analysis, the main steam isolation was assumed to occur immediately after the reactor trip on low pressurizer pressure. This instantaneous steam isolation assumption results in a stored energy in the secondary side which would be conservatively greater than what would exist if the main steam isolation time delay was considered. For a small break LOCA, the higher energy in the secondary coolant will reduce the heat transfer from the primary to the secondary side, maximize the steam produced in the reactor coolant system (RCS), and minimize the transient water level in the core. This will result in a more conservative calculation with respect to the core uncovery following a small break LOCA. For a large break LOCA analysis, the higher stored energy in the secondary coolant will increase the amount of thermal energy to be transferred into the RCS and increase the potential for steam binding to occur in the steam generator tubes during blowdown phase of the transient. This will prolong the time of blowdown and result in higher peak cladding temperature. Therefore, an increase in the MSIV closure time and steam isolation time by 3 seconds does not have an adverse effect on LOCA analyses.

2.5 Main Feedline Break

The licensee has performed a main feedline break (MFLB) analysis for Unit 2 to support the proposed transition to the Westinghouse 17x17 Vantage-5 fuel assembly assuming an steam isolation time which included an additional 3 seconds delay of main steam isolation. The results of the analysis demonstrate that the acceptance criteria for the event are met. Though no reevaluation of the MFLB was performed for Unit 1, this is acceptable because an increase in the steam isolation time delay will not have an adverse impact on the MFLB result. Following a MFLB, the reactor coolant system will initially undergo a cooldown due to expulsion of secondary water through the broken feedline. The RCS temperature transient will quickly turn around following the isolation of the main steam lines. An increase in the MSIV closure time will result in additional heat removal from RCS. Hence, the RCS will stabilize at a slightly lower temperature than in the licensing basis MFLB analysis. Thus, the results of a MFLB analysis with increased MSIV closure time are less severe than the results of the licensing basis MFLB analysis.

The staff has reviewed the limiting accident scenarios which rely upon the MSIV closure for mitigation of consequence. An increase of MSIV closure time has no adverse effect on the consequences of accident scenarios such as LOCA and MFLB. An increase of MSIV closure time has no effect on a SGTR accident because the isolation of ruptured steam generator tubes was not modelled in the analysis. The MSLB core response, and mass and energy release were analyzed with assumption of a steam line isolation response time of 11 seconds upon reaching the set point of high steam flow coincident with low steam pressure. The results of MSLB analysis remain acceptable and justify the proposed TS change to increase the response time.

In the total response time of 11 seconds, 8 seconds is being allocated for MSIV stroke time for closing, and 3 seconds is for signal processing and electronic delay. Therefore, the analyses with an 11 second steam isolation delay support the proposed changes to increase the MSIV closure time from 5 to 8 seconds, and changes of TS Table 3.3-5 to increase the response times from 8 and 7 seconds to 11 and 10 seconds, respectively, for the High Steam Flow Coincident With Low Steam Pressure and High Containment Pressure set points. The licensee also proposed to increase the steam line isolation response time upon High Steam Flow Coincident With Low-Low T_{avg} from 10 to 13 seconds. This is acceptable since it was considered as a backup signal and was not modelled in the licensing basis safety analysis.

We have reviewed the licensee's analysis, methodology and underlying calculations and find them acceptable. Therefore, the staff concludes that the licensee's proposed changes to the Units 1 and 2 TS Surveillance Requirement 4.7.1.5.1 and Table 3.3-5 regarding the MSIV closure time and steam isolation delay times are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change in a surveillance requirement. We have 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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.

4.0 CONCLUSION

We have 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.

5.0 REFERENCES

1. Letter from M. P. Alexich (Indiana Michigan Power) to USNRC, "Donald C. Cook Nuclear Plant Unit 1, Docket No. 50-315, License No. DPR-58, Expedited Technical Specification Change Request, Steam Generator Stop Valves," AEP:NRC:1120, January 31, 1990.

2. Letter from M. P. Alexich (Indiana Michigan Power) to USNRC, "Donald C. Cook Nuclear Plant Unit 2, Docket No. 50-316, License No. DPR-74, Technical Specification Change Request, Steam Generator Stop Valves," AEP:NRC:1123, May 14, 1990.

3. Letter from M. P. Alexich (Indiana Michigan Power) to USNRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316, License Nos. DPR-58 and DPR-74, Unit No. 2 Cycle 8 Reload Licensing, Proposed Technical Specifications for Unit 2 Cycle 8, and Related Unit 1 Proposals," AEP:NRC:1071E, February 6, 1990.

4. WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Unit 1 Licensing Report," October 1988.

5. WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2 Licensing Report," September 1989.

Date: August 22, 1990

Principal Contributor: Y. Hsii, NRR/SRXB