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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-58

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated January 22, 1982, as supplemented by letter dated July 3, 1983, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change in Technical Specifications is to become effective within 30 days of issuance of this amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensee shall adhere to the Technical Specifications for the systems, components, or operation existing at the time. The period of time between changeover of systems, components, or operation shall be minimized or compensated for by suitable temporary alternatives.

4. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to Technical
Specifications

Date of Issuance: November 22, 1983

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 57
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated January 22, 1982, as supplemented by letter dated July 3, 1983, 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. 57 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change in Technical Specifications is to become effective within 30 days of issuance of this amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensee shall adhere to the Technical Specifications for the systems, components, or operation existing at the time. The period of time between changeover of systems, components, or operation shall be minimized or compensated for by suitable temporary alternatives.

4. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to Technical
Specifications

Date of Issuance: November 22, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. DPR-58

AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NOS. 50-315 AND 50-316

Revise Appendix A as follows:

Remove Pages - Unit 1	Insert Pages - Unit 1
3/4 3-11	3/4 3-11*
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13*
3/4 3-14	3/4 3-14
3/4 3-26a	3/4 3-26a

Remove Pages - Unit 2	Insert Pages - Unit 2
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12*
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14*
3/4 3-25a	3/4 3-25a

*Included for convenience only

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 0.6 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 0.6 seconds
14. Steam Generator Water Level--Low-Low	$\leq$ 1.5 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip.	NOT APPLICABLE

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	N.A.	N.A.	S/U(1)	N.A.
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M	1, 2
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)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6)	M and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R	M	1, 2
8. Overpower ΔT	S	R	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	M	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow - Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	M(5) and S/U(1)	1, 2*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2*

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial imbalance above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 17% of narrow range instrument span each steam generator	≥ 16% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay.	3196, +18, -36 volts with a 2±.2 second delay
c. Safety Injection	Not Applicable	Not Applicable
d. Loss of Main Feedwater Pumps	Not Applicable	Not Applicable
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 17% of narrow range instrument span each steam generator	≥ 16% or narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	≥ 2750 Volts--each bus	≥ 2725 Volts--each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196, +18, -36 volts with a 2±.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 min. time delay	3596, +36, -18 volts with a 2.0 minute ± 6 second time delay

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	N.A.	N.A.	S/U(1)	N.A.
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M	1, 2
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)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6)	M and S/U(1)	2(7), 3(7), 4 and
7. Overtemperature ΔT	S	R	M	1, 2
8. Overpower ΔT	S	R	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	M	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow - Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R.	M	1, 2
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	M(5) and S/U(1)	1, 2*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2*

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference > 2 percent.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.

INSTRUMENTATION

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 and with RESPONSE TIMES as shown in Table 3.3-5.

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 and CHANNEL FUNCTIONAL 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 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 20% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196, +18, -36 volts with a 2 ± 0.2 second delay
c. Safety Injection	Not Applicable	Not Applicable
d. Loss of Main Feedwater Pumps	Not Applicable	Not Applicable
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 20% of narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	≥ 2750 Volts--each bus	≥ 2725 Volts--each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196, +18, -36 volts with a 2 ± 0.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 minute time delay	3596, +36, -18 volts with a 2.0 minute ± 6 second time delay