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U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-315
and 50-316

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
Post Office Box 18
Bowling Green Station
New York, New York 10004

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. DPR-58 and Amendment No. 34 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively.. The amendments consist of changes to the Facility Operating Licenses including changes to the Technical Specifications in response to your application transmitted by letter dated December 10, 1980.

These amendments revise the Technical Specifications and adds certain license conditions for the Category "A" Lessons Learned (NUREG-0578) requirements implemented at the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2.

During our review of the proposed changes we suggested certain modifications to your proposed Technical Specifications. These modifications were discussed with your staff and with their agreement were included in these amendments.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,
Original signed by:
S. A. Varga

CP

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 49 to DPR-58
2. Amendment No. 34 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

cp 8/17/81

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Mr. John Dolan
Indiana and Michigan Electric Company

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Washington, D. C. 20515

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. 49
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 December 10, 1980, 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, Facility Operating License No. DPR-58 is hereby amended as follows:

   (a) Changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(b) Add a new paragraph 2.J. to read as follows:

2.J. System Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(c) Add a new paragraph 2.K. to read as follows:

2.K. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 25, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. DPR- 58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 0-2	3/4 0-2
3/4 3-16	3/4 3-16
*3/4 3-21	*3/4 3-21
3/4 3-21a	3/4 3-21a
3/4 3-22	3/4 2=22
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-26a	3/4 3-26a
3/4 3-29	3/4 2-29
3/4 3-31	3/4 3-31
3/4 3-33	3/4 3-33
3/4 3-33a	3/4 3-33a
---------	3/4 3-54
---------	3/4 3-55
---------	3/4 3-56
*3/4 4-5	*3/4 4-5
3/4 4-6	3/4 4-6
-------	3/4 4-41
-------	3/4 4-42
*3/4 6-17	*3/4 6-17
3/4 6-18	3/4 6-18
3/4 7-6	3/4 7-6
*B3/4 3-3	*B3/4 3-3
B3/4 3-4	B3/4 3-4
*B3/4 4-1	*B3/4 4-1
B3/4 4-2	B3/4 4-2

*Included for convenience

(Continued)

Remove Pages	Insert Pages
*B3/4 4-11	*B3/4 4-11
B3/4 4-12	B3/4 4-12
6-4	6-4
6-5	6-5
*6-6	*6-6

*Included for convenience

3/4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

The provisions of Specification 4.0.4 are not applicable to the performance of surveillance activities associated with fire protection technical specifications, 4.3.3.7, 4.7.9 and 4.7.10, until the completion of the initial surveillance interval associated with each specification.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	14
e. Differential Pressure Between Steam Lines - High				1, 2, 3##	
Four Loops Operating	3/steam line	2/steam line any steam line	2/steam line		14
Three Loops Operating	3/operating steam line	1###/steam line, any operating steam line	2/operating steam line		15

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
COINCIDENT WITH EITHER T_{avg}--Low-Low				1, 2, 3##	
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops		14
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		15
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops		14
Three Loops Operating	1 pressure/operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14

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	2/Bus	2/Bus	2/Bus	1, 2, 3	14*
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2, 3	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	2/Bus	2/Bus	1, 2, 3, 4	14*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT STANDBY within the following 6 hours.

b. Above P-11 and P-12, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the bypass condition; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT SHUTDOWN within the following 12 hours.

TABLE 3.3-3 (Continued)

b. Above P-11 or P-12, demonstrate that the Minimum Channels OPERABLE requirement is met within 1 hour; operation may continue with the inoperable channel bypassed and one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel 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.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels $\geq$ 1915 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 3 of 4 T_{avg} channels $\geq$ 544°F.	P-12 prevents or defeats manual block of safety injection actuation high steam line flow and low steam line pressure.
	With 2 of 4 T_{avg} channels < 540°F.	Allows manual block of safety injection actuation on high steam line flow and low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	$\leq$ 1.1 psig	$\leq$ 1.2 psig
d. Pressurizer Pressure--Low	$\geq$ 1815 psig	$\geq$ 1805 psig
e. Differential Pressure Between Steam Lines--High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low or Steam Line Pressure--Low	$\leq 1.42 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load	$\leq 1.56 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load.
	$T_{avg} \geq$ 541°F	$T_{avg} \geq$ 539°F
	$\geq$ 600 psig steam line pressure	$\geq$ 580 psig steam line pressure

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	$\geq$ 17% of narrow range instrument span each steam generator	$\geq$ 16% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196$\pm$18 volts with a 2$\pm$.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	$\geq$ 17% of narrow range instrument span each steam generator	$\geq$ 16% or narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	$\geq$ 2750 Volts--each bus	$\geq$ 2725 Volts--each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196$\pm$18 volts with a 2$\pm$.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 min. time delay	3596$\pm$18 volts with a 2.0 minute $\pm$ 6 second time delay

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	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
d. Containment Air Recirculation Fan	≤ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor 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

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	1, 2. 3
d. Pressurizer Pressure--Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	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	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	M	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	MODES IN WHICH SURVEILLANCE REQUIRED
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	1, 2, 3
8. LOSS OF POWER				
a. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3, 4
b. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3, 4

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, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. 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.

TABLE 3.3-11

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2
4. Reactor Coolant Pressure - Wide Range	2
5. Pressurizer Water Level	2
6. Steam Line Pressure	2/Steam Generator
7. Steam Generator Water Level - Narrow Range	1/Steam Generator
8. Refueling Water Storage Tank Water Level	2
9. Boric Acid Tank Solution Level	1
10. Auxiliary Feedwater Flow Rate	1/Steam Generator*
11. Reactor Coolant System Subcooling Margin Monitor	1**
12. PORV Position Indicator - Limit Switches***	1/Valve
13. PORV Block Valve Position Indicator - Limit Switches	1/Valve
14. Safety Valve Position Indicator - Acoustic Monitor	1/Valve

*Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument.

**PRODAC 250 subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

***Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Position Indicator - Limit Switches instruments.

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	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

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2485 PSIG ± 1% in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume less than or equal to 62% of span and at least 150 kW of pressurizer heaters.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the required capacity of heaters.

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.11 Three power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either (1) restore the block valve(s) to OPERABLE status, or (2) close the block valve(s) and remove power from the block valve(s), or (3) close the associated PORV(s) and remove power from the associated solenoid valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.*

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs 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 performance of a CHANNEL CALIBRATION.

*When ACTION 3.4.11.b.(3) is applied, no report pursuant to Specification 6.9.1.9 is required for the PORV.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested nor is a report required pursuant to Specification 6.9.1.9 when ACTION 3.4.11.a is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.1.1.2.b and 4.8.2.3.2.c.

TABLE 3.6-1 (Continued)

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
A. PHASE "A" ISOLATION (Continued)			
27. ECR-10	Cont. H_2 Sample Return	Yes	10
28. ECR-11	Cont. H_2 Sample - Air to Rec. E	Yes	10
29. ECR-12	Cont. H_2 Sample - Air from Rec. E	Yes	10
30. ECR-13	Cont. H_2 Sample - Low. Cont. Vol.	Yes	10
31. ECR-14	Cont. H_2 Sample - Low Cont. Vol.	Yes	10
32. ECR-15	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
33. ECR-16	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
34. ECR-17	Cont. H_2 Sample - Air to Rec. W	Yes	10
35. ECR-18	Cont. H_2 Sample - Air from Rec. W	Yes	10
36. ECR-19	Cont. H_2 Sample - Cont. Dome Vol.	Yes	10
37. ECR-20	Cont. H_2 Sample-Return	Yes	10
38. ECR-21	Cont. H_2 Sample - Air to Rec. E.	Yes	10
39. ECR-22	Cont. H_2 Sample - Air fr. Rec. E	Yes	10
40. ECR-23	Cont. H_2 Sample - Low Cont. Vol.	Yes	10
41. ECR-24	Cont. H_2 Sample - Low Cont. Vol.	Yes	10
42. ECR-25	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
43. ECR-26	Cont. H_2 Sample - Up Cont. Vol.	Yes	10
44. ECR-27	Cont. H_2 Sample - Air to Rec. W.	Yes	10
45. ECR-28	Cont. H_2 Sample - Air Fr. Rec. W.	Yes	10
46. ECR-29	Cont. H_2 Sample - Cont. Dome Vol.	Yes	10
47. GCR-301	N_2 Supply to Pressurizer Relief Tank	Yes	10
48. GCR-314	N_2 Supply to Accumulators	Yes	10
49. ICR-5	Accumulators Sample	Yes	10
50. ICR-6	Accumulators Sample	Yes	10
51. MCR-251	Sample Line from Steam Gen. Outlet #1	Yes	10
52. MCR-252	Sample Line from Steam Gen. Outlet #2	Yes	10
53. MCR-253	Sample Line from Steam Gen. Outlet #3	Yes	10
54. MCR-254	Sample Line from Steam Gen. Outlet #4	Yes	10
55. NCR-105	Hot Leg Sample	Yes	10
56. NCR-106	Hot Leg Sample	Yes	10

TABLE 3.6-1 (Continued)

VALVE NUMBER		FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)			
57.	NCR-107	PRZ Liquid Sample	Yes	10
58.	NCR-108	PRZ Liquid Sample	Yes	10
59.	NCR-109	PRZ Steam Sample	Yes	10
60.	NCR-110	PRZ Steam Sample	Yes	10
61.	NCR-252	Primary Water to Pressurizer Relief Tank	Yes	10
62.	QCM-250	RCP Seal Water Discharge	No	15
63.	QCM-350	RCP Seal Water Discharge	No	15
64.	QCR-300	Letdown to Letdown Hx.	No	10
65.	*QCR-301	Letdown to Letdown Hx.	No	10
66.	RCR-100	PRZ Relief Tank to Gas Anal.	Yes	10
67.	RCR-101	PRZ Relief Tank to Gas Anal.	Yes	10
68.	VCR-10	Glycol Supply to Fan Cooler	Yes	10
69.	VCR-11	Glycol Supply to Fan Cooler	Yes	10
70.	VCR-20	Glycol Supply from Fan Cooler	Yes	10
71.	VCR-21	Glycol Supply from Fan Cooler	Yes	10
72.	XCR-100	Control Air to Containment	No	10
73.	XCR-101	Control Air to Containment Isolation	No	10
74.	XCR-102	Control Air to Containment Isolation	No	10
75.	XCR-103	Control Air to Containment	No	10
B.	PHASE "B" ISOLATION			
1.	CCM-451	CCW from RCP Oil Coolers	No	60
2.	CCM-452	CCW from RCP Oil Coolers	No	60
3.	CCM-453	CCW from RCP Thermal Barrier	No	30
4.	CCM-454	CCW from RCP Thermal Barrier	No	30
5.	CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	No	60
6.	CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	No	60
7.	ECR-31	Containment Air Particle Radio Gas Detector	No	10
8.	ECR-32	Containment Air Particle Radio Gas Detector	No	10

*To be installed during the 1982 refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

3. 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.

4. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

3/4.3.3.6 AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)

The OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to 1) monitor the core flux patterns that are representative of the power peaking factor in the limiting fuel rod. The limiting fuel rod is the fuel rod that has the least margin to the exposure dependent F_Q limit curve, and 2) limit the core average axial power profile such that the total power peaking factor F_Q in the limiting fuel rod is maintained within acceptable limits.

R_j factors are used to determine the APDMS setpoint limits $[F_j(Z)]_S$. On a full core basis the R_j and σ_j factors are calculated in accordance with the equations on Pages 3/4 2-15 and 3/4 2-16.

However, near BOC, thimbles not in the region of fuel which contains the limiting total peaking factor, $F_{Qi\ell}$, may not follow the axial power distribution of the hot rod. This situation will manifest itself in the form of large σ_j for thimbles not in the same region as the total peak $F_{Qi\ell}$. In this situation, if the rod with the limiting total peaking factor were to move from one fuel region to another, the neutron flux in the thimble with the smallest σ_j would not necessarily follow the axial power distribution of the power in the new limiting rod.

In order to cope with this difficulty, it is permissible to calculate as many σ_j's and $\overline{R}_j$'s for each thimble as there are fuel types or regions in the core. Each $\overline{R}_j$ and σ_j for a thimble j is to be calculated from the equations on Pages 3/4 2-15 and 3/4 2-16 with the following exception. For each $\overline{R}_j$ and σ_j for thimble j, a different $F_{Qi\ell}$ and T(E) shall be used. The different σ_j's and R_j's for thimble j shall be calculated substituting for $F_{Qi\ell}^{meas}$ and T(E) the values pertaining to the limiting peak relative power from each fuel region. Obviously for one of these calculations the limiting peak relative power from one region will be the core limiting total peaking factor.

If this option is chosen, the σ_j set to use for APDMS thimble selection and the R_j set to use for the calculation of $[F_J(Z)])_S$ shall be the set obtained using the limiting peak relative power from the same fuel type as the $F_{Qi\ell}$ from the most recent incore flux map.

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3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

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.

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.30. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (51 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation conditions.

3/4.4.5 STEAM GENERATORS

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 parameter 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 (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 12 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems", 1971 Edition and Addenda through Winter 1972.

All areas scheduled for volumetric examination have been pre-service mapped using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels and the requirement for remote underwater accessibility.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

The nondestructive testing for repairs on components greater than 2 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 2 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Collant System is unchanged. Therefore, satisfactory performance of a system leak test at 2235 psig following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing and Figure 3.4-1.

3/4.4.11 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should be relief valve become inoperable. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
NON-Licensed	2	1
Shift Technical Advisor	1**	None Required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

**Shared with D. C. Cook Unit 2.

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 position, except for (1) the Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (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.

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 Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

6.5.1.1 The PNSRC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of the:

Chairman:	Plant Manager or designated alternate
Member:	Asst. Plant Manager
Member:	Operations Supervisor
Member:	Technical Supervisor
Member:	Maintenance Supervisor
Member:	Instrument and Control Engineer
Member:	Nuclear Engineer
Member:	Chemical Supervisor
Member:	Performance Supervising Engineer
Member:	Radiation Protection Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PNSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSRC shall meet at least once per calendar month and as convened by the PNSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PNSRC shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PNSRC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

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. 34
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 December 10, 1980, 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 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 fo the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-74 is hereby amended as follows:

   (a) Changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 34 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(b) Renumber paragraph H to 3.

(c) Add a new paragraph 2.H. to read as follows:

2.H. System Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(d) Add a new paragraph 2.K. to read as follows:

2.K. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 25, 1981

ATTACHMENT TO LICENSE AMENDMENT

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

DOCKET NO. 50-316

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-20a	3/4 3-20a
B3/4 3-22	B3/4 3-22
3/4 3-25a	3/4 3-25a
3/4 3-28	3/4 3-28
3/4 3-30	3/4 3-30
*3/4 3-31	*3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-32a	3/4 3-32a
*3/4 3-45	*3/4 3-45
3/4 3-46	3/4 3-46
3/4 3-47	3/4 3-47
*3/4 3-48	*3/4 3-48
*3/4 4-5	*3/4 4-5
3/4 4-6	3/4 4-6
-------	3/4 4-30
-------	3/4 4-31
*3/4 6-19	*3/4 6-19
3/4 6-20	3/4 6-20
3/4 6-21	3/4 6-21
*3/4 6-22	*3/4 6-22
3/4 7-5	3/4 7-5
3/4 7-6	3/4 7-6
B3/4 3.2	B3/4 3.2
*B3/4 3-3	*B3/4 3-3
*B3/4 4-1	*B3/4 4-1
B3/4 4-2	B3/4 4-2

*Included for convenience

(Continued)

Remove Pages	Insert Pages
*B3/4 4-9a	*B3/4 4-9a
B3/4 4-10	B3/4 4-10
*6-3	*6-3
6-4	6-4
6-5	6-5
*6-6	*6-6

*Included for convenience

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	2/Bus	2/Bus	2/Bus	1, 2, 3	14*
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2, 3	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	2/Bus	2/Bus	1, 2, 3, 4	14*

TABLE 3.3-3 (Continued)

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel 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.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure pressure channels $\geq$ 2010 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 3 of 4 T_{avg} channels $\geq$ 544°F.	P-12 prevents or defeats manual block of safety injection actuation high steam line flow and low steam line pressure.
	With 2 of 4 T_{avg} channels < 540°F.	Allows manual block of safety injection actuation on high steam line flow and low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks.

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 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 volts with a 2 ± 0.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 minute time delay	3596 ± 18 volts with a 2.0 minute ± 6 second time delay

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
6.	Steam Line Pressure--Low	
a.	Safety Injection (ECCS)	≦ 12.0#/24.0##
b.	Reactor Trip (from SI)	≦ 2.0
c.	Feedwater Isolation	≦ 8.0
d.	Containment Isolation-Phase "A"	≦ 18.0#/28.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Motor Driven Auxiliary Feedwater Pumps	≦ 60.0
g.	Essential Service Water System	≦ 14.0#/48.0##
h.	Steam Line Isolation	≦ 8.0
7.	Containment Pressure--High-High	
a.	Containment Spray	≦ 45.0
b.	Containment Isolation-Phase "B"	Not Applicable
c.	Steam Line Isolation	≦ 7.0
d.	Containment Air Recirculation Fan	≦ 600.0
8.	Steam Generator Water Level--High-High	
a.	Turbine Trip-Reactor Trip	Not Applicable
b.	Feedwater Isolation	Not Applicable
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

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}-- Low-Low	S	R	M	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	M	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	MODES IN WHICH SURVEILLANCE REQUIRED
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	1, 2, 3
8. LOSS OF POWER				
a. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3, 4
b. 4 kv Bus Degraded Voltage	S	R	M	1, 2, 3, 4

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, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. 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.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2
4. Reactor Coolant Pressure - Wide Range	2
5. Pressurizer Water Level	2
6. Steam Line Pressure	2/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator
8. Refueling Water Storage Tank Water Level	2
9. Boric Acid Tank Solution Level	1
10. Auxiliary Feedwater Flow Rate	1/steam generator*
11. Reactor Coolant System Subcooling Margin Monitor	1**
12. PORV Position Indicator - Limit Switches***	1/valve
13. PORV Block Valve Position Indicator - Limit Switches	1/valve
14. Safety Valve Position Indicator - Acoustic Monitor	1/valve

*Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument.

**PRODAC 250 subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

***Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Position Indicator - Limit Switches instruments.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	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

INSTRUMENTATION

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.7 The axial power distribution monitoring system (APDMS) shall be OPERABLE with:

a. At least two detector thimbles available for which $\overline{R}$ has been determined from full incore flux maps. These two thimbles shall be those having the lowest uncertainty, σ, covering the full configuration of permissible rod patterns permitted at RATED THERMAL POWER.

b. At least two movable detectors, with associated devices and readout equipment, available for mapping $F_j(Z)$ in the above required thimbles.

APPLICABILITY: When the APDMS is used for monitoring the axial power distribution*#.

ACTION: With the APDMS inoperable, do not use the system for determining the Axial Power Distribution. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 The full incore flux maps used to determine $\overline{R}$ and for monitoring $F_j(Z)$ shall be updated at least once per 31 days. The continued accuracy and representativeness of the selected thimbles shall be verified by using their latest flux maps to update the $\overline{R}$ for each representative thimble. The original uncertainty, σ, shall not be updated, except as follows:

*Except as provided in Specification 4.2.6.1.b.
#The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume less than or equal to 62% of span and at least 150 kW of pressurizer heaters.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters, either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the required capacity of heaters.

REACTOR COOLANT SYSTEM

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.11 Three power operator relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either (1) restore the block valve(s) to OPERABLE status, or (2) close the block valve(s) and remove power from the block valve(s), or (3) close the associated PORV(s) and remove power from the associated solenoid valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.*

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs 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 performance of a CHANNEL CALIBRATION.

*When ACTION 3.4.11.b.(3) is applied, no report pursuant to Specification 6.9.1.9 is required for the PORV.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested nor is a report required pursuant to Specification 6.9.1.9 when ACTION 3.4.11.a is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.1.1.2.b and 4.8.2.3.2.c.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
A. PHASE "A" ISOLATION (Continued)		
49. 1CR-5	Accumulators Sample	≤ 10
50. 1CR-6	Accumulators Sample	≤ 10
51. MCR-251#	Sample Line from Steam Gen. Outlet #1	≤ 10
52. MCR-252#	Sample Line from Steam Gen. Outlet #1	≤ 10
53. MCR-253#	Sample Line from Steam Gen. Outlet #3	≤ 10
54. MCR-254#	Sample Line from Steam Gen. Outlet #4	≤ 10
55. NCR-105	Hot Leg Sample	≤ 10
56. NCR-106	Hot Leg Sample	≤ 10
57. NCR-107	PRZ Liquid Sample	≤ 10
58. NCR-108	PRZ Liquid Sample	≤ 10
59. NCR-109	PRZ Steam Sample	≤ 10
60. NCR-110	PRZ Steam Sample	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)		
61.	NCR-252	Primary Water to Pressurizer Relief Tank	≤ 10
62.	QCM-250	RCP Seal Water Discharge	≤ 15
63.	QCM-350	RCP Seal Water Discharge	≤ 15
64.	QCR-300	Letdown to Letdown Hx.	≤ 10
*65.	QCR-301	Letdown to Letdown Hx.	≤ 10
66.	RCR-100	PRZ Relief Tank to Gas Anal.	≤ 10
67.	RCR-101	PRZ Relief Tank to Gas Anal.	≤ 10
68.	VCR-10	Glycol Supply to Fan Cooler	≤ 10
69.	VCR-11	Glycol Supply to Fan Cooler	≤ 10
70.	VCR-20	Glycol Supply from Fan Cooler	≤ 10
71.	VCR-21	Glycol Supply from Fan Cooler	≤ 10
72.	XCR-100	Control Air to Containment	≤ 10
73.	XCR-101	Control Air to Containment Isolation	≤ 10

*To be installed during the 1982 refueling outage.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)		
74.	XCR-102	Control Air to Containment Isolation	≤ 10
75.	XCR-103	Control Air to Containment	≤ 10
B.	Phase "B" ISOLATION		
1.	CCM-451	CCW from RCP Oil Coolers	≤ 60
2.	CCM-452	CCW from RCP Oil Coolers	≤ 60
3.	CCM-453	CCW from RCP Thermal Barrier	≤ 30
4.	CCM-454	CCW from RCP Thermal Barrier	≤ 30
5.	CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	≤ 60
6.	CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	≤ 60
7.	ECR-31	Containment Air Particle Radio Gas Detector	≤ 10
8.	ECR-32	Containment Air Particle Radio Gas Detector	≤ 10

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
B.	PHASE "B" ISOLATION (Continued)		
9.	WCR-901	NESW to Low. Containment Vent #1	≤ 10
10.	WCR-903	NESW from Low. Containment Vent #1	≤ 10
11.	WCR-905	NESW to Low. Containment Vent #2	≤ 10
12.	WCR-907	NESW from Low. Containment Vent #2	≤ 10
13.	WCR-909	NESW to Low. Containment Vent #3	≤ 10
14.	WCR-911	NESW from Low. Containment Vent #3	≤ 10
15.	WCR-913	NESW to Low. Containment Vent #4	≤ 10
16.	WCR-915	NESW from Low. Containment Vent #4	≤ 10
17.	WCR-921	NESW to Up. Containment Vent #1	≤ 10
18.	WCR-923	NESW from Up. Containment Vent #1	≤ 10
19.	WCR-925	NESW to Up. to Containment Vent #2	≤ 10
20.	WCR-927	NESW from Up. Containment Vent #2	≤ 10

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate emergency busses and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a discharge pressure $\geq$ 1375 psig on recirculation flow.

2. Verifying that the steam turbine driven pump develops a discharge pressure of $\geq$ 1285 psig at a flow of $\geq$ 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3. 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.

4. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and normalizing its respective output.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

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.

3/4.3.3.6 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.

3/4.3.3.7 AXIAL POWER DISTRIBUTION MONITIORING SYSTEM (APDMS)

OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to 1) monitor the core flux patterns that are representative of the peak core power density and 2) limit the core average axial power profile such that the total power peaking factor F_Q is maintained within acceptable limits.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain calculated DNBR above the design DNBR value during Condition I and II events. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the calculated DNBR will be maintained above the design DNBR value. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (51 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

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 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 (primary-to-secondary leakage = 500 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. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

COMPONENT	HEAT NO.	MATERIAL TYPE	CU %	P %	NDTT °F	MINIMUM 50 FT-LB/35 MIL TEMP °F: Parallel to Major Working Direction	MINIMUM 50 FT-LB/35 MIL TEMP °F: Normal to Major Working Direction	RT_{NDT} °F	AVG. UPPER SHELF (FT-LB): Parallel to Major Working Direction	AVG. UPPER SHELF (FT-LB): Normal to Major Working Direction
BOT. HD.	B0018-1B	A533BCL1	NA	NA	-50	-11	9*	-50	177	115*
BOT. PEEL SEG.	C5823-2	A533BCL1	NA	NA	-10	25	45*	-10	129	84*
BOT. PEEL SEG.	A4957-3	A533BCL1	NA	NA	-10	0	20*	-10	149	97*
WELD	L5WC39	C5592-1 TO C5521-2	.05	.019	-40	NA	25	-35	NA	97
HAZ		C5592-1 TO C5521-2	NA	NA	-10	NA	80	20	NA	109

*Estimate Based on USAEC Regulatory Standard Review Plan, Section 5.3.2 and MTEB 5-2.

(A) 60% Shear

(B) 70% Shear

NA - Not available or not applicable as appropriate.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a smple can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should be relief valve come inoperable. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

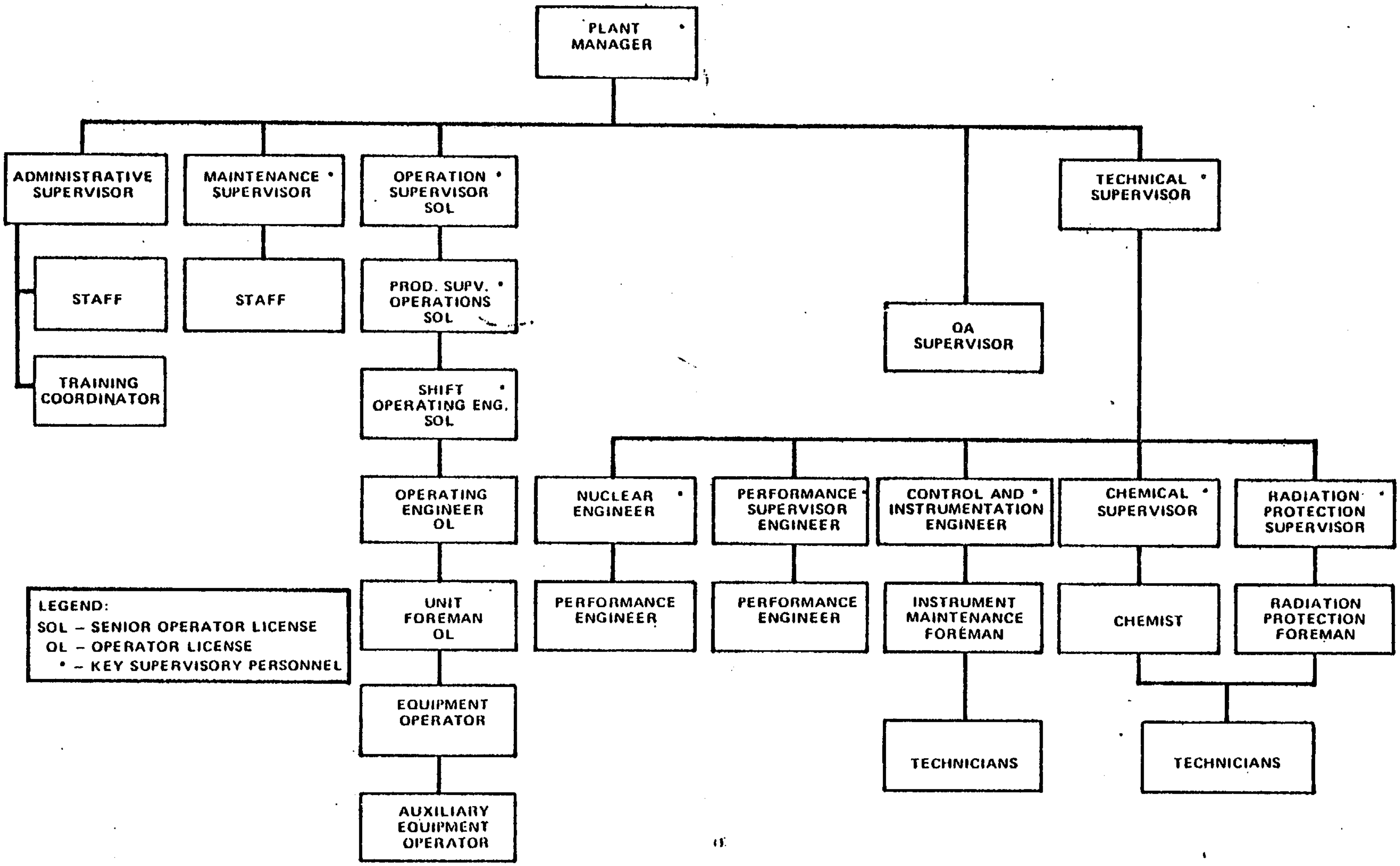

FIGURE 6.2-2 Facility Organization – Donald C. Cook – Unit No. 2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1
Shift Technical Advisor	1**	None required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

**Shared with D. C. Cook Unit 1.

6.0 ADMINISTRATIVE CONTROLS

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 Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (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.

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 Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

6.5.1.1 The PNSRC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of the:

Chairman:	Plant Manager or designated alternate
Member:	Asst. Plant Manager
Member:	Operations Supervisor
Member:	Technical Supervisor
Member:	Maintenance Supervisor
Member:	Instrument and Control Engineer
Member:	Nuclear Engineer
Member:	Chemical Supervisor
Member:	Performance Supervising Engineer
Member:	Radiation Protection Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PNSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSRC shall meet at least once per calendar month and as convened by the PNSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PNSRC shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PNSRC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTRIC COMPANY

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

DOCKET NOS. 50-315 AND 50-316

Introduction

By letter dated December 10, 1980, Indiana and Michigan Electric Company, (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-58 and DPR-74 for Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensees's request is in direct response to the NRC staff's letter dated July 2, 1980.

Background Information

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter dated March 20, 1980.

Subsequently, by letter dated July 2, 1980, we requested that the licensee amend their TS to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

Evaluation

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has provided the requisite emergency power supplies. The licensee has proposed revised technical specifications which provide appropriate surveillance and actions in the event of component inoperability and are thus acceptable. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

2.1.3.a Direct Indication of (of Flow) Valve Position

The licensee has provided a direct indication of power-operated relief valve (PORV) and safety valve position in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a 31-days channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. These instruments, subcooling meters, receive and process data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated March 20, 1980. The licensee submitted TSs with a 31-day channel check and an 18-month channel calibration requirements and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.4 Diverse Containment Isolation

The licensee has modified the containment isolation system so that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. We have reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated March 20, 1980. The modification is such that is does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Reopening of containment isolation would require deliberate operator action. The TSs submitted by the licensee list each affected containment isolation valve and provide for the appropriate surveillance and actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.a Auto Initiation of Auxiliary Feedwater Systems

The licensee has provided for the automatic initiation of auxiliary (emergency) feedwater flow. We have previously reviewed the design and installation of this system as part of our Lessons Learned Category "A" program. The circuits are designed to be testable and the design retains the capability of manual actuation from the control room even in the event of failure of the auto-initiating circuitry. The TSs submitted by the licensee list the appropriate components, describe the tests and provide for proper test frequency. The TSs contain appropriate actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.b Auxiliary (Emergency) Feedwater Flow Indication

The licensee has installed auxiliary (emergency) feedwater flow indication that meets our testability and vital power requirements. We reviewed this system in our Safety Evaluation dated March 10, 1980. The licensee has proposed a TS with 31-day channel check and 18-month channel calibration requirements. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

2.2.1.b Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift to perform the function of accident assessment. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Part of the STA duties are related to operating experience review function. Based on our review, we find the licensee's submittal to satisfy our requirements and is acceptable.

EVALUATION TO SUPPORT LICENSE CONDITIONS

2.1.4 Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. By letter dated December 10, 1980, the licensee agreed to adopt such a license condition; accordingly we have included this condition in the license.

2.1.8.c Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. By letter dated December 10, 1980 the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because that amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: August 25, 1981

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-315 AND 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 49 to Facility Operating License No. DPR-58, and Amendment No. 34 to Facility Operating License No. DPR-74 issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications and adds certain License Conditions for the Category "A" Lessons Learned (NUREG-0578) requirements implemented at the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated December 10, 1980, (2) Amendment Nos. 49 and 34 to License Nos. DPR-58 and DPR-74, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Maude Reston Palenski Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Wahington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 25th day of August, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing