November 13, 1992

Docket Nos. 50-315
and 50-316

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

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - AMENDMENT NOS. 168 AND 151 TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74 (TAC NOS. M79832 AND M79833)

The Commission has issued the enclosed Amendment No. 168 to Facility Operating License No. DPR-58 and Amendment No. 151 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated February 15, 1991 as supplemented September 13, 1991.

These amendments make administrative changes to the TS for both units. Four items in the proposed change were not purely administrative in nature. The changes dealt with operability of the automatic trip logic, engineered safety featured system instrumentation, containment air lock, and the physical stops on the Auxiliary Building Crane. These changes will be evaluated under a separate cover.

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

Sincerely,

/s/

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 168 to DPR-58
2. Amendment No. 151 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PM:PD31	OGC	D:PD31	
NAME	MShuttleworth	JStang:		LMarsh	
DATE	9/25/92	9/28/92	10/2/92	/ /92	/ /

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

November 13, 1992

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and 50-316

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Office of Nuclear Reactor Regulation

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3. Safety Evaluation

cc w/enclosures:
See next page

Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

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U.S. Nuclear Regulatory Commission
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Mr. S. Brewer
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DATED: November 13, 1992

AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK
AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DRP-74-D. C. COOK

~~Docket File~~
NRC & Local PDRs
PDIII-1 Reading
D.C. Cook Plant File
J. Roe
J. Zwolinski
L. Marsh
M. Shuttleworth
W. Dean
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (8), P-137
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cc: Plant Service list

190053

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated February 15, 1991 as supplemented September 13, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 168 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

L B Marsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 13, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 168

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
2-1	2-1
2-2	2-2
2-6	2-6
3/4 3-29	3/4 3-29
3/4 3-37	3/4 3-37
3/4 3-53	3/4 3-53
3/4 3-55	3/4 3-55
3/4 4-36	3/4 4-36
3/4 6-15	3/4 6-15
3/4 6-21	3/4 6-21
3/4 7-5	3/4 7-5
5-1	5-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

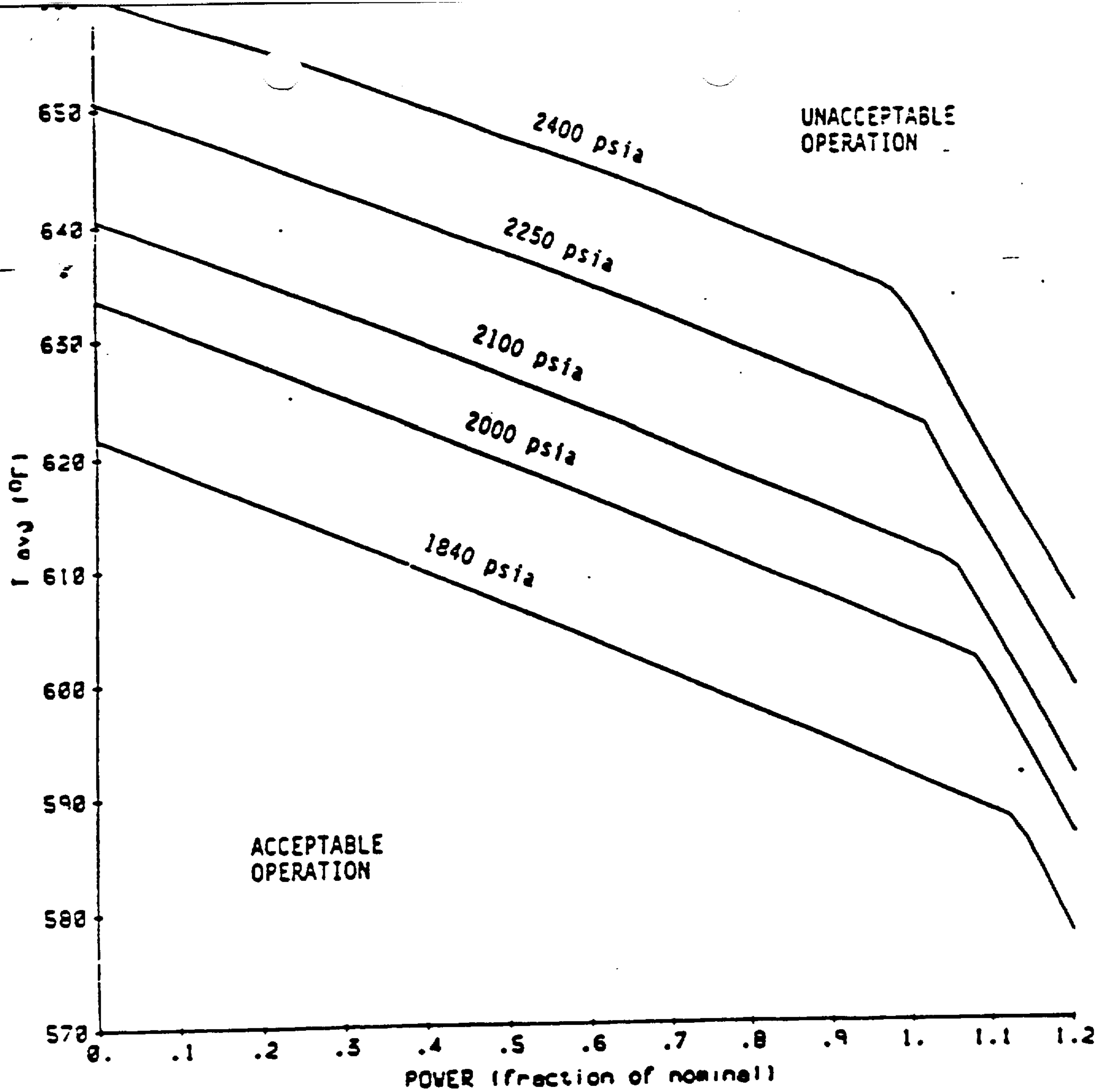
ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN DEGREES F)
1840	(0.0, 622.1), (1.13, 587.3), (1.20, 577.5)
2000	(0.0, 633.8), (1.08, 601.4), (1.20, 586.0)
2100	(0.0, 640.8), (1.06, 609.8), (1.20, 591.3)
2250	(0.0, 650.7), (1.02, 621.9), (1.20, 598.9)
2400	(0.0, 660.1), (0.98, 633.7), (1.20, 606.2)

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	Greater than or equal to 17% of narrow range instrument span - each steam generator	Greater than or equal to 16% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 0.71×10^6 lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 0.73×10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2750 volts - each bus	Greater than or equal to 2725 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz - each bus
17. Turbine Trip		
A. Low Fluid Oil Pressure	Greater than or equal to 800 psig	Greater than or equal to 750 psig
B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

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)	Less than or equal to 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0*/28.0**
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 14.0*/48.0**
h. Steam Line Isolation	Less than or equal to 11.0
7. Containment Pressure--High-High	
a. Containment Spray	Less than or equal to 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	Less than or equal to 10.0
d. Containment Air Recirculation Fan	Less than or equal to 600 .0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Less than or equal to 2.5
b. Feedwater Isolation	Less than or equal to 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

TABLE 3.3-6 (Continued)
TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirements, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test.

ACTION 22A- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

ACTION 22B- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.

4. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

TABLE 3.3-10
Unit 1 and Common Area Fire Detection Systems

Detector System Location	Total Number of Detectors Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
Auxiliary Building			
a) Elevation 573			23/0C
b) Elevation 587			55/0C
c) Elevation 609			41/0C
d) Elevation 633			41/0C
e) Elevation 650			34/0C
f) New Fuel STGE Area			4/0C
g) RP Access Control & Chem Labs			25/0
U1 East Main Steam Valve Enclosure			28/0**
U1 Main Steam Line Area El. 612 (Around Containment)			13/0**
U1 NESW Valve Area El. 612			2/0
U1 4KV Switchgear (AB)		0/3	0/2
U1 4KV Switchgear (CD)		0/3	0/2
U1 Engr. Safety System Switchgear & XFMR. Rm.		0/5	0/9
U1 CRD, XFMR. & Switchgear Rm. Inverter & Bttry. Rms.		0/5	0/8
U1 Pressurizer Heater XFMR. Rm.			12/0
U1 Diesel Fuel Oil Transfer Pump Rm.	0/1		
U1 Diesel Generator Rm. 1AB	0/2		
U1 Diesel Generator Rm. 1CD	0/2		
U1 Diesel Generator Ramp Corr.			4/0
U1&2 AFWP Vestibule			2/0C
U1 Control Room			45/0
U1 Switchgear Cable Vault		0/10***	0/13
U1 Control Room Cable Vault			0/65****
U1 Aux. Cable Vault			0/6
U1&2 ESW Basement Area			4/0C
U1 ESW Pump & MCC Rms.			9/0

C System protects area common to both Units 1 and 2

*(x/y) x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** circuit contains both smoke and flame detectors

*** two circuits of five detectors each

**** two circuits of 32 and 33 detectors each

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*****
15.	Incore Thermocouples (Core Exit Thermocouples)	2/Core Quadrant
16.	Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	One Train (3 Channels/Train)
17.	Containment Sump Level	1****
18.	Containment Water Level	2****

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

** PPC 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 Indicator - Limit Switches instruments.

**** The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases.

***** Pressurizer safety valve (SV-45A) position indicator acoustic monitor QR-107A is exempted from the above requirements until the end of Cycle 12.

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. 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 CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

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 when ACTION 3.4.11.a or 3.4.11.c 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 4.8.2.3.2.d and 4.8.1.1.2.e.

---

* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

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

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

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

4.6.3.1.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1 (Continued)

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
C. CONTAINMENT PURGE EXHAUST** (Continued)		
12. VCR-205	UPPER COMP. PURGE AIR INLET	5
13. VCR-206	UPPER COMP. PURGE AIR OUTLET	5
14. VCR-207*	CONT. PRESS RELIEF FAN ISOLATION	5
D. MANUAL ISOLATION VALVES(1)		
1. ICM-111	RHR TO RC COLD LEGS	NA
2. ICM-129	RHR INLET TO PUMPS	NA
3. ICM-250	BORON INJECTION OUTLET	NA
4. ICM-251	BORON INJECTION OUTLET	NA
5. ICM-260	SAFETY INJECTION OUTLET	NA
6. ICM-265	SAFETY INJECTION OUTLET	NA
7. ICM-305	RHR/CTS SUCTION FROM SUMP	NA
8. ICM-306	RHR/CTS SUCTION FROM SUMP	NA
9. ICM-311	RHR TO RC HOT LEGS	NA
10. ICM-321	RHR TO RC HOT LEGS	NA
11. NPX 151 VI	DEAD WEIGHT TESTER	NA
12. PA 343	CONTAINMENT SERVICE AIR	NA
13. SF-151	REFUELING WATER SUPPLY	NA
14. SF-153	REFUELING WATER SUPPLY	NA
15. SF-159	REFUELING CAVITY DRAIN TO PURIFICATION SYSTEM	NA
16. SF-160	REFUELING CAVITY DRAIN TO PURIFICATION SYSTEM	NA
17. SI-171	SAFETY INJECTION TEST LINE	NA
18. SI-172	ACCUMULATOR TEST LINE	NA

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2

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

1. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

2. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

b. At least one auxiliary feedwater flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.1.2.a - MODES 1, 2, 3.
Specification 3.7.1.2.b - At all times when Unit 2 is in MODES 1, 2, or 3.

ACTIONS:

When Specification 3.7.1.2.a is applicable:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

When Specification 3.7.1.2.b is applicable:

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

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

Site Boundary For Gaseous and Liquid Effluents

5.1.3 The SITE BOUNDARY for gaseous and liquid effluents shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 160 feet.*

c. Minimum thickness of concrete walls = 3'6".

d. Minimum thickness of concrete roof = 2'6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner, side and dome = 3/8 inches.

g. Nominal thickness of steel liner, bottom = 1/4 inch.

h. Net free volume = 1.24×10^6 cubic feet.

---

*From grade (Elev. 608') to inside of dome.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
License No. DPR-74

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

A. The application for amendment by Indiana Michigan Power Company (the licensee) dated February 15, 1991 as supplemented September 13, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 151 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

LBMarsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 151

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
1-7	1-7
2-1	2-1
2-6	2-6
3/4 2-1	3/4 2-1
3/4 3-36	3/4 3-36
3/4 3-46	3/4 3-46
3/4 3-52	3/4 3-52
3/4 4-33	3/4 4-33
3/4 6-27	3/4 6-27
3/4 7-5	3/4 7-5
3/4 8-20	3/4 8-20
3/4 9-9	3/4 9-9
5-1	5-1

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of radioactive liquid, resin and sludge wastes from liquid systems into a form that meets shipping and burial site requirements.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and the conduct of environmental radiological monitoring program.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	Greater than or equal to 21% of narrow range instrument span - each steam generator	Greater than or equal to 19.2% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 1.47 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 1.56 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2905 volts - each bus	Greater than or equal to 2870 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz - each bus
17. Turbine Trip		
A. Low Fluid Oil Pressure	Greater than or equal to 58 psig	Greater than or equal to 57 psig
B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

3.4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about a targe flux difference. The target band is specified in the COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

   1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

   2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test.

ACTION 22A- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or
2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.
3. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

ACTION 22B- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or
2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.
3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.
4. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

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
15.	Incore Thermocouples (Core Exit Thermocouples)	2/Core Quadrant
16.	Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	One Train (3 channels/Train)
17.	Containment Sump Level	1****
18.	Containment Water Level	2****

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

** PPC 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 Indicator - Limit Switches instruments.

**** The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases.

*****Pressurizer safety valve (SV-45C) position indicator acoustic monitor QR-107C is exempted from the above requirements until the end of Cycle 8.

TABLE 3.3-11
Unit 2 and Common Area Fire Detection Systems

Detection System Location	Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
		Total Number of Detectors	
Auxiliary Building			
a) Elevation 573			23/0C
b) Elevation 587			55/0C
c) Elevation 609			41/0C
d) Elevation 633			41/0C
e) Elevation 650			34/0C
f) New Fuel STGE Area			4/0C
U2 East Main Steam Valve Enclosure			28/0**
U2 Main Steam Line Area El. 612 (Around Containment)			13/0**
U2 NESW Valve Area El. 612			2/0
U2 4KV Switchgear (AB)		0/3	0/2
U2 4KV Switchgear (CD)		0/3	0/2
U2 Engr. Safety System Switchgear & XFMR. Rm.		0/5	0/14
U2 CRD, XFMR & Switchgear Rm. Inverter & AB Bttry. Rms.		0/5	0/17
U2 Pressurizer Heater XFMR. Rm.			12/0
U2 Diesel Fuel Oil XFMR. Rm.	0/1		
U2 Diesel Generator Rm. 2AB	0/2		
U2 Diesel Generator Rm. 2CD	0/2		
U2 Diesel Generator Ramp Corr.			4/0
U1&2 AFWP Vestibule			2/0C
U2 Control Room			42/0
U2 Switchgear Cable Vault		0/10***	0/13
U2 Control Rm. Cable Vault			0/76****
U2 Aux. Cable Vault			0/6
U1&2 ESW Basement Area			4/0C
U2 ESW Pump & MCC Rms.			9/0

C System protects area common to both Units 1 and 2

*(x/y) x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** circuit contains both smoke and flame detectors

*** two circuits of five detectors each

**** two circuits of 38 detectors each

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

   Within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

   within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. 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.

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 when ACTION 3.4.11.a or 3.4.11.c 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.e and 4.8.2.3.2.d.

*PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

TABLE 3.6-1 (Cont'd)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
D. MANUAL ISOLATION VALVES(1) (Cont'd)		
3. ICM-250	BORON INJECTION OUTLET	NA
4. ICM-251	BORON INJECTION OUTLET	NA
5. ICM-260	SAFETY INJECTION OUTLET	NA
6. ICM-265	SAFETY INJECTION OUTLET	NA
7. ICM-305	RHR/CTS SUCTION FROM SUMP	NA
8. ICM-306	RHR/CTS SUCTION FROM SUMP	NA
9. ICM-311#	RHR TO RC HOT LEGS	NA
10. ICM-321#	RHR TO RC HOT LEGS	NA
E. OTHER		
1. CS-442-1	SEAL WTR. TO RCP #1	NA
2. CS-442-2	SEAL WTR. TO RCP #2	NA
3. CS-442-3	SEAL WTR. TO RCP #3	NA
4. CS-442-4	SEAL WTR. TO RCP #4	NA

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2

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

1. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

2. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

b. At least one auxiliary feedwater flow path in support of Unit 1 shutdown function shall be available.

APPLICABILITY: Specification 3.7.1.2.a - MODES 1, 2, 3.
Specification 3.7.1.2.b - At all times when Unit 1 is in MODES 1, 2, or 3.

ACTIONS:

When Specification 3.7.1.2.a is applicable:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT Shutdown within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

When Specification 3.7.1.2.b is applicable:

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

ELECTRICAL POWER SYSTEMS

3/4.8.3 Alternative A.C. Power Sources

LIMITING CONDITION FOR OPERATION

3.8.3.1 The steady state bus voltage for the manual alternate reserve source* shall be greater than or equal to 90% of the nominal bus voltage.

APPLICABILITY: Whenever the manual alternate reserve source (69 kV) is connected to more than two buses.

ACTION: With bus voltage less than 90% nominal, adjust load on the remaining buses to maintain steady state bus voltage greater than or equal to 90% limit.

SURVEILLANCE REQUIREMENTS

4.8.3.1 No additional surveillance requirements other than those required by Specifications 4.8.1.1.1 and 4.8.1.2.

*Shared with Cook Nuclear Plant Unit 1.

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: During Core Alterations or movement of irradiated fuel within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

5.0 DESIGN FEATURES

5.1 SITE

Exclusion Area

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

Low Population Zone

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

Site Boundary For Gaseous and Liquid Effluents

5.1.3 The SITE BOUNDARY for gaseous and liquid effluents shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 160 feet.

c. Minimum thickness of concrete walls = 3'6".

d. Minimum thickness of concrete roof = 2'6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 1.24×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

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

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated February 15, 1991 as supplemented September 13, 1991, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The proposed amendments would make administrative changes to the TS for both units. Four items in the proposed change were not purely administrative in nature. The changes dealt with operability of the automatic trip logic, engineered safety featured system instrumentation, containment air lock, and the physical stops on the Auxiliary Building Crane. These changes will be evaluated under a separate cover.

2.0 EVALUATION

The February 15, 1991 application contains 25 proposed administrative changes to the TS. By letter dated September 13, 1991, the licensee provided additional information concerning the original application. The proposed changes fall into two categories. The first category of proposed changes include those changes that are strictly typographical in nature. Note: the number proceeding each of the following changes corresponds to the number on the proposal in the licensee's application.

a. (2) Unit 2 TS 1.29 spelling error, "resine" should be "resin."

b. (6) Unit 2 TS 3.2.1 spelling error, "target ban" should be "target band."

c. (7) Correct inconsistency in Unit 1 TS Table 3.3-5, "660 seconds" should be changed to "600 seconds" consistent with Unit 2 for the requirement to start the containment air recirculation system.

d. (10) Correct the cross referencing error in TS Table 3.3-6, Units 1 and 2. TS 6.9.1.13 has been deleted and should no longer be referenced.

e. (11) Correct typing error in Unit 1 TS Table 3.3-10 and Unit 2 Table 3.3-11 by listing the correct elevation of 573 for item a in the table.

f. (13) Correct cross reference error in TS 3/4.4.11.3, Units 1 and 2. For Unit 1, reference to 4.8.1.1.2b should be 4.8.1.1.2e. For Unit 2, reference to 4.8.1.1.2c should be 4.8.1.1.2e.

g. (20) Correct typographical error in TS 3/4.8.3, Unit 2 "Unit 2" should say "Unit 1."

h. (22) Remove the asterisk after the word "system."

i. (23) Change the term "site boundary" in TS 5.1.3 for both units to all capital letters.

The second category of proposed changes involve clarification or correction of terminology to correctly reflect as-built plant conditions and operations.

a. (3) Adds the word "average" to the limiting condition for operation in TS 2.1.1 to provide clarification.

b. (4) Change rated thermal power on TS page 2-2, Unit 1, "3414 MWT" to match licensed rated thermal power of "3250 MWT."

c. (5) Change functional unit 17.A in Table 2.2-1 from "Low Trip System Pressure" to "Low Fluid Oil Pressure" to reflect proper terminology.

d. (12) Change the name of the plant computer in Unit 1, TS Table 3.3-11 and Unit 2, TS Table 3.3-10 to reflect the installation of the new plant process computer.

e. (18) The terms "inlet" and "outlet" are being corrected in Table 3.6-1 in both units TS to reflect the proper reference terminology used by operators.

f. (19) Add descriptive terms to differentiate between auxiliary feedwater pumps in TS 4.9.7.1 for both units.

The changes are either typographical in nature or provide administrative clarifications. The proposed changes enhance safety by improving the usability of the TS while assuring all the necessary requirements are maintained in the current TS.

These corrections and enhancements will result in improved operator performance and reduce the probability of incorrect operator actions when using the TS. Based on the evaluation, the staff finds the above proposed changes to the TS are acceptable.

Proposed changes (15), (17), and (24) involving typographical errors have been acted on in previous amendments and, therefore, do not need to be evaluated in this amendment.

Items (8), (9), (14), (16), and (21) are not administrative and deal with operability. These proposed changes will be evaluated under a separate cover.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 31435 and 56 FR 57697). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

Principal Contributor: John Stang

Date: November 13, 1992