January 26, 1994

Docket No. 50-316

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

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT
RE: EXTENSION OF CERTAIN 18-MONTH SURVEILLANCES (TAC NO. M86340)

The Commission has issued the enclosed Amendment No. 159 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated April 16, 1993, and supplemented September 28 and December 3, 1993.

The amendment revises TS to allow certain tests normally designated as 18-month surveillances to be delayed until the next refueling outage scheduled to begin August 6, 1994. The current cycle for Unit 2 will be lengthened by approximately 5 months due to a planned power reduction in order to separate the refueling outages between Unit 1 and Unit 2. This addresses the remaining 12 of the 16 groups of surveillances for which you requested extensions in your application dated April 16, 1993. Surveillance extensions for four of the groups were previously approved for Unit 2 in Amendment 158 dated December 22, 1993.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 159 to DPR-74
2. Safety Evaluation

cc w/enclosures:
See next page

with [illegible] changes on pg 3 of SE re: Group (4) heading
JHB

*See previous concurrence

OFFICE	LA:PD31	PE:PD31	BC:HICB	BC:EELB	BC:EMEB	BC:SRXB	OGC	(A)D:PD31
NAME	CJamerson	BWetzel:	JWermiel	CBerlinger	JNorberg*	RCJones*	CPW	for RBlough
DATE	12/29/93	12/29/93	12/30/93	1/4/94	1/3/94	1/4/94	1/5/94	1/26/93

OFFICIAL RECORD COPY

FILENAME: G:\WPDOCS\DCCOOK\C086340A.AMD

9402010374 940126
PDR ADOCK 05000316
P PDR

310054

NRC FILE CENTER COPY

DF01

Docket No. 50-316

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

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: EXTENSION OF CERTAIN 18-MONTH SURVEILLANCES (TAC NO. M86340)

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated April 16, 1993, and supplemented September 28 and December 3, 1993.

The amendment revises TS to allow certain tests normally designated as 18-month surveillances to be delayed until the next refueling outage scheduled to begin August 6, 1994. The current cycle for Unit 2 will be lengthened by approximately 5 months due to a planned power reduction in order to separate the refueling outages between Unit 1 and Unit 2. This addresses the remaining 12 of the 16 groups of surveillances for which you requested extensions in your application dated April 16, 1993. Surveillance extensions for four of the groups were previously approved for Unit 2 in Amendment 158 dated December 22, 1993.

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,

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

Enclosures:
1. Amendment No. to DPR-74
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PE:PD31	BC:HICB	BC:EELB	BC:EMEB	BC:SRXB	OGC	(A)D:PD31
NAME	CJamerson	BWetzel:	JWermiel	CBerlinger	JNorberg	RCJones		RBlough
DATE	12/29/93	12/29/93	/ /	/ /	1/3/94	/ /	/ /93	/ /93

OFFICIAL RECORD COPY FILENAME: G:\WPDOCS\DCCOOK\C086340A.AMD

Docket No. 50-316

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

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: EXTENSION OF CERTAIN 18-MONTH SURVEILLANCES (TAC NO. M86340)

The Commission has issued the enclosed Amendment No.   to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated April 16, 1993, and supplemented September 28 and December 3, 1993.

The amendment revises TS to allow certain tests normally designated as 18-month surveillances to be delayed until the next refueling outage scheduled to begin August 6, 1994. The current cycle for Unit 2 will be lengthened by approximately 5 months due to a planned power reduction in order to separate the refueling outages between Unit 1 and Unit 2. This addresses the remaining 12 of the 16 groups of surveillances for which you requested extensions in your application dated April 16, 1993. Surveillance extensions for four of the groups were previously approved for Unit 2 in Amendment 158 dated December 22, 1993.

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,

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

Enclosures:
1. Amendment No.   to DPR-74
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PE:PD31	BC:HICB	BC:EELB	BC:EMEB	BC:SRXB	OGC	(A)D:PD31
NAME	CJamerson	BWetzel:	JWermiel	CBerlinger	JNorberg	RCJones		RBlough
DATE	12/29/93	12/29/93	/ /	/ /	/ /	1/4/94	/ /93	/ /93

OFFICIAL RECORD COPY FILENAME: G:\WPDOCS\DCCOOK\CO86340A.AMD cc:

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

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

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgman, Michigan 49106

Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
Bridgman, Michigan 49106

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

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

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

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

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

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

DATED: January 26, 1994

AMENDMENT NO. 159 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK, UNIT 2

Docket File
NRC & Local PDRs
PDIII-1 Reading
J. Roe
J. Zwolinski
R. Blough
C. Jamerson
J. Hickman
B. Wetzel
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
J. Wermiel 8/H/1
J. Norberg 7/E/21
C. Berlinger 7/E/1
R.C. Jones 8/E/21
ACRS (10)
OPA
OC/LFDCB
SEDB
cc: Plant Service list

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 159
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 April 16, 1993, as supplemented September 28 and December 3, 1993, 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.

9402010377 940126
PDR ADOCK 05000316
P PDR

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. 159 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for

A. Randolph Blough, Acting 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: January 26, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 159

TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

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

REMOVE	INSERT
3/4 1-22	3/4 1-22
3/4 3-11	3/4 3-11
3/4 3-14	3/4 3-14
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-44d	3/4 3-44d
3/4 3-47	3/4 3-47
3/4 4-14	3/4 4-14
3/4 4-33	3/4 4-33
3/4 5-5	3/4 5-5
3/4 5-8	3/4 5-8
3/4 6-47	3/4 6-47
3/4 7-6	3/4 7-6
3/4 7-13	3/4 7-13
3/4 7-20	3/4 7-20
3/4 7-31	3/4 7-31
3/4 8-4	3/4 8-4
3/4 8-9	3/4 8-9

POSITION INDICATOR CHANNELS-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one rod position indicator channel (excluding demand position indication) shall be OPERABLE for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*# and 5*#

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.†

*With the reactor trip system breakers in the closed position.

#See Special Test Exception 3.10.5.

† The provisions of Technical Specification 4.0.8 are applicable.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2,8),M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)†	M	1, 2
8. Overpower ΔT	S	R(9)†	M	1, 2
9. Pressurizer Pressure--Low	S	R†	M	1, 2
10. Pressurizer Pressure--High	S	R†	M	1, 2
11. Pressurizer Water Level--High	S	R†	M	1, 2
12. Loss of Flow - Single Loop	S	R(8)	M	1

† The provisions of Technical Specification 4.0.8 are applicable.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

† The provisions of Technical Specification 4.0.8 are applicable.

ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Purge and Exhaust Isolation					
1) Manual	-------------------	See Functional Unit 9	-------------------		
2) Containment Radio-activity-High	S	R	M	N.A.	1, 2, 3, 4
4. STEAM LINE ISOLATION					
a. Manual	-------------------	See Functional Unit 9	-------------------		
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Press-ure--High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	S	R†	M	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kV Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R†	N.A.	1, 2

† The provisions of Technical Specification 4.0.8 are applicable.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMP					
a. Steam Generator Water Level--Low-low	S	R	M	N.A.	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	N.A.	1, 2, 3
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	S	R†	M	N.A.	1, 2, 3, 4
b. 4 kv Bus Degraded Voltage	S	R†	M	N.A.	1, 2, 3, 4
9. MANUAL					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation-Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	N.A.	N.A.	N.A.	R†	1, 2, 3, 4
b. Containment Spray Containment Isolation-Phase "B" Containment Purge and Exhaust Isolation Containment Air Recirculation Fan	N.A.	N.A.	N.A.	R†	1, 2, 3, 4
c. Containment Isolation-Phase "A" Containment Purge and Exhaust Isolation	N.A.	N.A.	N.A.	R†	1, 2, 3, 4
d. Steam Line Isolation	N.A.	N.A.	M(1†)	R†	1, 2, 3

† The provisions of Technical Specification 4.0.8 are applicable.

TABLE 4.3-6A

APPENDIX R REMOTE SHUTDOWN MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1. Steam Generators 1 and 4 Level	LSI Cabinet 1 and LSI Cabinet 4	M	R
2. Steam Generators 2 and 3 Level	LSI Cabinet 2 and LSI Cabinet 4	M	R
3. Steam Generators 1 and 4 Pressure	LSI Cabinet 4 and LSI Cabinet 5	M	R
4. Steam Generators 2 and 3 Pressure	LSI Cabinet 4 and LSI Cabinet 6	M	R
5. Reactor Coolant Loop 4 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 5	M	R†
6. Reactor Coolant Loop 4 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 5	M	R†
7. Reactor Coolant Loop 2 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 6	M	R†
8. Reactor Coolant Loop 2 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 6	M	R†
9. Pressurizer Level	LSI Cabinet 3	M	R
10. Reactor Coolant System Pressure	LSI Cabinet 3	M	R
11. Charging Cross-Flow Between Units	Corridor Elev. 587'	n/a	R*
12. Source Range Neutron Detector (N-23)	LSI Cabinet 4	n/a	R

* Charging Cross-Flow between Units is an instrument common to both Unit 1 and 2. This surveillance will only be conducted on an interval consistent with Unit 1 refueling.

† The provisions of Technical Specification 4.0.8 are applicable.

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
15. Incore Thermocouples (Core Exit Thermocouples)(4)	M	R(1)†
16. Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	M(2)	R(3)†
17. Containment Sump Level*	M	R
18. Containment Water Level*	M	R

(1) Partial range channel calibration for sensor to be performed below P-12 in MODE 3.
(2) With one train of Reactor Vessel Level Indication inoperable, Subcooling Margin Indication and Core Exit Thermocouples may be used to perform a CHANNEL CHECK to verify the remaining Reactor Vessel Indication train OPERABLE.
(3) Completion of channel calibration for sensors to be performed below P-12 in MODE 3.
(4) The core exit thermocouples will not be installed until the 1988 refueling outage; therefore, surveillances will not be required until that time. See license amendment dated April 10, 1987.
* 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.
† The provisions of Technical Specification 4.0.8 are applicable.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One of the containment atmosphere particulate radioactivity monitoring channels (ERS-2301 or ERS-2401),

b. The containment sump level and flow monitoring system, and

c. Either the containment humidity monitor or one of the containment atmosphere gaseous radioactivity monitoring channels (ERS-2305 or ERS-2405).

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring channels are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,†

c. Containment humidity monitor (if being used) - performance of CHANNEL CALIBRATION at least once per 18 months.

† The provisions of Technical Specification 4.0.8 are applicable.

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.

† The provisions of Technical Specification 4.0.8 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.†

e. At least once per 18 months, during shutdown, by:†

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1.	Centrifugal charging pump	Greater than or equal to 2405 psig
2.	Safety Injection pump	Greater than or equal to 1409 psig
3.	Residual heat removal pump	Greater than or equal to 190 psig

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

† The provisions of Technical Specification 4.0.8 are applicable.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.†

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits, at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F as determined at least once per hour when any RCS cold leg temperature is between 152°F and 200°F.

† The provisions of Technical Specification 4.0.8 are applicable.

CONTAINMENT SYSTEMS

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:†

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-2.

b. Visually inspecting at least 95 percent of the seal's entire length and:

   1. Verifying that the seal and seal mounting bolts are properly installed, and

   2. Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

† The provisions of Technical Specification 4.0.8 are applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5 by:

a. Verifying that each motor driven pump develops an equivalent discharge pressure of greater than or equal to 1240 psig at 60°F in recirculation flow.

b. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of greater than or equal to 1180 psig at 60°F and at a flow of greater than or equal to 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.

c. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.

d. 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. This requirement is not applicable for those portions of the auxiliary feedwater system being used intermittently to maintain steam generator level.

e. Verifying at least once per 18 months during shutdown 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.†

f. Verifying at least once per 18 months during shutdown 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.†

g. Verifying at least once per 18 months during shutdown that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

† The provisions of Technical Specification 4.0.8 are applicable.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1

a. At least two independent essential service water loops shall be OPERABLE.

b. At least one essential service water flowpath associated with support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.4.1.a. - MODES 1, 2, 3, and 4.
Specification 3.7.4.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.4.1.a is applicable:

With only one essential service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.4.1.b is applicable:

With no essential service water flow path available in support of Unit 1 shutdown functions, return at least one flow path to available status within 7 days or provide equivalent shutdown capability in Unit 1 and return the equipment to service 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.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two essential service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.†

† The provisions of Technical Specification 4.0.8 are applicable.

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7.1 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.1.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspection†

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 3.7-9. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 3.7-9 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 156.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified as acceptable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that

† The provisions of Technical Specification 4.0.8 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.9.2 Each of the above required water spray and/or sprinkler systems shall be demonstrated to be OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel as provided by Technical Specification 4.7.9.1.1.e.

b. At least once per 18 months:†

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and*

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By visual inspection of deluge and preaction system piping (this is not required for systems supervised by air) to verify their integrity.

3. By visual inspection of each open head deluge nozzle to verify that there is no blockage.

c. At least once per 3 years by performing an air flow test through the piping of each open head deluge system and verifying each open head deluge nozzle is unobstructed.

† The provisions of Technical Specification 4.0.8 are applicable.

*The fire protection water flow surveillance testing may be suspended until the completion of the fire protection water storage tank and fire pump installations (May 31, 1993). The surveillance testing suspended as a result of this amendment will be initiated at its normal frequency within four months of the new fire protection water storage tanks and fire pumps being declared OPERABLE, with the exception of unit outage required testing which would be completed before the end of the next scheduled unit outage.

SURVEILLANCE REQUIREMENTS (Continued)

a) A kinematic viscosity of greater than or equal to 1.9 centistokes but less than or equal to 4.1 centistokes at 40°C (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6 but less than or equal to 40.1), if gravity was not determined by comparison with supplier's certification.

b) A flash point equal to or greater than 125°F.

2) By verifying, in accordance with the test specified in ASTM D1298-80 and prior to adding the new fuel to the storage tanks, that the sample has either an API gravity of greater than or equal to 30 degrees but less than or equal to 40 degrees at 60°F or an absolute specific gravity at 60/60°F of greater than or equal to 0.82 but less than or equal to 0.88, or an API gravity of within 0.3 degrees at 60°F when compared to the supplier's certificate or a specific gravity of within 0.0016 at 60/60°F when compared to the supplier's certificate.

3) By verifying, in accordance with the test specified in ASTM D4176-82 and prior to adding new fuel to the storage tanks, that the sample has a clear and bright appearance with proper color.

4) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are within the appropriate limits when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D2622-82.

d. At least once per 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-83, and verifying that total particulate contamination is less than 10 mg/liter when tested in accordance with ASTM D2276-83, Method A*.

e. At least once per 18 months, during shutdown, by:†

1. Subjecting the diesel engine to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

---

*The actions to be taken should any of the properties be found outside of the specified limits are defined in the Bases.

† The provisions of Technical Specification 4.0.8 are applicable.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. A day fuel tank containing a minimum of 70 gallons of fuel,

   2. A fuel storage system containing a minimum indicated volume of 46,000 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.5.†

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

† The provisions of Technical Specification 4.0.8 are applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 159 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated April 16, 1993, as supplemented September 28 and December 3, 1993, the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The proposed amendment would extend specific TS surveillances which have various due dates, the first of which is February 5, 1994. The surveillances would be extended to the Unit 2 refueling outage, which is currently scheduled to begin August 6, 1994. All of the requested surveillance extensions are associated with surveillances normally performed during refueling outages. The current cycle will be lengthened approximately 5 months due to a planned power reduction in order to separate the current dual unit outages. The licensee categorized the surveillances into groups of related surveillances. Extensions for four of these groups (Groups 1, 2, 6 and 11) were previously granted for Unit 2 in Amendment 158 dated December 22, 1993. This amendment pertains only to the remaining 12 groups, which are listed below:

Group	TSs Affected	Description of Change
(3)	Table 4.3-2, Item 6.d 4.7.1.2.e 4.7.1.2.f	Delay auxiliary feedwater system testing including channel functional testing of loss of main feedwater pump signal
(4)	4.8.1.1.2.e 4.8.1.2 4.4.11.3 4.7.4.1.b 4.3.2.1.1, Table 4.3-2, Items 8.a & 8.b	Delay diesel generator testing including relief valve testing and essential service water valve testing and delay calibrations of time delay relays for 4 Kv bus loss of degraded voltage.
(5)	Table 4.3-1 Items 7 & 8 4.3.2.1.2 (P-12) Table 4.3-2 Item 4.d Table 4.3-6A Items 5,6,7 & 8 Table 4.3-10 Items 2,3 & 11 4.3.3.6, Table 4.3-10, Item 15	Delay RTD calibrations and calibrations of incore thermocouples in mode 3

9402010378 940126
PDR ADOCK 05000316
P PDR

(7)	Table 4.3-10, Item 16	Delay reactor vessel level indication system calibration
(8)	4.1.3.3	Delay analog rod position indication functional testing
(9)	4.5.2.d.1 4.5.3.1	Delay RHR auto-closure interlock testing
(10)	4.7.7.1.a	Delay visual inspection of inaccessible snubbers
(12)	4.6.5.9	Delay divider barrier seal inspection
(13)	4.7.9.2.b.1	Delay RCP fire protection testing
(14)	Table 4.3-10, Item 18 4.5.2.d.2 4.5.3.1 4.4.6.1.b	Delay containment water level calibrations, sump visual inspection and calibration of containment flow monitoring system
(15)	4.2.5.2 Table 4.3-1 Items 12 & 13	Delay reactor coolant flow calibrations
(16)	Table 4.3-2 Items 9.a, 9.b, 9.c & 9.d	Delay ESF Manual Trip Actuating Device Operational Test

The due dates specified in the licensee's submittal for each TS affected are the most limiting due date, in that for multiple TSs the date when the first surveillance is due is listed. Also, the due dates given in the submittal include the 25% maximum allowable extension beyond the surveillance interval allowed by TS 4.0.2.

2.0 EVALUATION

Generic Letter (GL) 91-04, "Changes in Technical Specifications Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," was published April 2, 1991. The purpose of the GL was to provide guidance to licensees wishing to take advantage of improvements in reactor fuels to increase the duration of the fuel cycle for their facilities. Although the licensee is not requesting a permanent change to a 24-month fuel cycle, it is requesting a one-time surveillance extension, in which some of the guidance of GL 91-04 will apply.

The staff included in its guidance in GL 91-04 the following statement:

> The NRC staff has reviewed a number of requests to extend 18-month surveillances to the end of a fuel cycle and a few requests for changes in surveillance intervals to accommodate a 24-month fuel cycle. The staff has found that the effect on safety is small because safety systems use redundant electrical and mechanical components and because licensees perform other surveillances during plant operation that confirm that these systems and components can perform their safety functions. Nevertheless, licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to

accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small. Licensees should confirm that historical plant maintenance and surveillance data support this conclusion.

The licensee's request for surveillance extension is very similar to extensions granted previously, one for Unit 1 approved by the NRC on April 17, 1987, and two for Unit 2 approved by the NRC on December 28, 1987, and February 29, 1988. The reasons for the extension and the equipment included in this request are similar. The specific TS changes are addressed below.

Group (3) Auxiliary Feedwater Pump Testing

The proposed amendment requests a 3-month extension for auxiliary feedwater system testing including channel functional testing of loss of main feedwater pump signal required by TS 4.7.1.2.e, 4.7.1.2.f and TS Table 4.3-2 Item 6.d. Testing of these items would require either plant shutdown or a trip of a main feedpump with the resultant reduction in plant power and associated thermal transient.

Portions of the system have been inadvertently tested by two spurious reactor trips which occurred on August 2 and 27, 1993. The system performed as required during the reactor trips. The licensee has noted in its submittal that these items have had an excellent test history. The licensee received a similar extension for these surveillances in 1987. Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

Group (4) Diesel Generator and Valve Testing and Delay of Calibrations of Time Delay Relays for 4 Kv Bus

The surveillance requirements of TS 4.8.1.1.2.e for diesel generators (DG) 2AB and 2CD are required by TS to be performed during shutdown. Surveillance requirements of 4.8.1.1.2.e require subjecting each DG to an inspection which is in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service. This surveillance interval requirement expires on March 25, 1994, for DG 2AB and February 18, 1994, for DG 2CD. An extension is also necessary for part of the requirements of TS 4.8.1.2, since 4.8.1.1.2 is referenced there. The extension for these requirements is needed from February 18, 1994 (limiting due date), through the Unit 2 refueling outage.

The licensee has verified with the DG vendor that the DG inspection required by TS 4.8.1.1.2.e can be extended on a one-time basis for the requested period. During the 5½-month extension period, each DG should accumulate five additional starts and 5-7 additional running hours. The licensee stated in its submittal that the effect of these additional starts and running hours on the DGs is insignificant based on the wear history of each machine. The licensee currently has a trending program for the parameters measured during TS required monthly testing. If an adverse trend began to develop, corrective measures would be taken to prevent a significant problem from occurring. The licensee also performed a review of previous test results which indicated that the DG associated circuitry would pass the surveillance tests during the extended period. Therefore, the staff finds the licensee's request for a one-time surveillance extension for the DGs acceptable.

The licensee also proposes to extend the surveillance period for testing the emergency power supply for the power-operated relief valves (PORVs) and block valves required by TS 4.4.11.3 and testing of the automatic valves in the essential service water (ESW) system required by TS 4.7.4.1.b. The testing of the PORVs and block valves involves cycling of the valves and is generally performed during shutdown in conjunction with the DG testing, as suggested by TS 4.4.11.3. The testing of the automatic valves in the ESW system must be performed during shutdown per TS 4.7.4.1.b and is generally conducted in conjunction with the DG testing because some of the ESW valves involve cooling water flow to the DG and its associated equipment.

The extension for both the ESW valves and the PORV emergency power supply are needed for the period of April 15, 1994, through the Unit 2 refueling outage. Previous test results do not indicate any reason to suspect that the valves and their associated circuitry would not pass the required surveillance during the extended interval. Therefore, the staff finds the licensee's request for a one-time surveillance extension for the ESW valves and the PORV and block valves emergency power supply acceptable.

In its letter dated December 3, 1993, the licensee also requested an extension of the 4 Kv loss of voltage and degraded voltage time delay relays required by TS 4.3.2.1.1, Table 4.3-2, Items 8.a and 8.b. The extension is needed from February 5, 1994, until the Unit 2 refueling outage. The licensee does not consider it prudent to perform this surveillance during operation because the components involved cannot be isolated from their normal power supply. Performance of the surveillance at power could result in a challenge to safety-related components if a power transfer were to occur and personnel safety might be at risk because the surveillance would be performed on live equipment.

The time delay relays involved in this surveillance are electronic and were installed in 1986. The licensee considers electronic relays to be highly reliable, accurate and repeatable and therefore should not drift outside of their acceptable setpoints. This was demonstrated during the previous three-channel calibration surveillances where no adjustments were required on the as-found conditions. Thus, there is no reason to believe that the relays would not perform their intended functions during the extension period. Therefore, the staff finds the licensee's request for a one-time surveillance extension for the time delay relays acceptable.

Group (5) Resistance Temperature Detector (RTD) and Incore Thermocouple Calibrations (Core Exit Thermocouples)

The proposed amendment requests a 3½ month extension for the calibration of the RTDs, as specified by TS Tables 4.3-1 Items 7 and 8, 4.3-2 Item 4.d, 4.3-6A Items 5, 6, 7 and 8, 4.3-10 Items 2,3 and 11 and TS 4.3.2.1.2 (P-12). Testing of these sensors is required to be performed during shutdown because of the low temperatures necessary for the calibration and because isothermal conditions throughout the reactor coolant system (RCS) are required.

Channel checks and functional tests will continue to be performed and would be expected to provide indication of RTD drift. Another indication of possible RTD drift would be the comparison of delta-T to the calorimetric calculated at power. The licensee has noted in its submittal that the history of the RTDs has been very stable with no significant drifting problems. In addition, the licensee performed a drift analysis using as-found instrument errors from previous instrument calibrations to project the amount of drift the instruments will have during the extension period. Details of the licensee's analysis were submitted in a letter dated December 3, 1993. In its submittal the licensee provided the results of its analysis and concluded that, based on these results, the drift data clearly indicated that instrument drift will be within acceptable limits while operating in the requested surveillance extension period. Therefore, the staff finds the licensee's request for a one-time surveillance extension of the RTDs acceptable.

In its letter dated December 3, 1993, the licensee also requested an extension of calibration of the incore thermocouples required by TS 4.3.3.6, Table 4.3-10, Item 15. This surveillance cannot be performed during reactor operation because TS require the unit to be in mode 3. This extension is needed from April 28, 1994, until the Unit 2 refueling outage.

TS 3.3.3.6 requires a minimum of two core exit thermocouples per core quadrant. There are a total of 58 thermocouples monitored (13 in quadrant I, 16 in quadrant II, 17 in quadrant III and 12 in quadrant IV). A monthly channel check is required by TS 4.3.3.6, Table 4.3-10, Item 15, but is administratively performed weekly. This channel check confirms that the core exit thermocouples have not changed significantly from the average reading and verifies TS compliance and will continue to be performed during the extension period. The licensee performed a review of this cycle's and the previous two cycle's core exit thermocouple data to determine if instrument drift could lead to several core exit thermocouples becoming inoperable during the extension period. No thermocouple drift was observed; thus compliance with TS 3.3.3.6 (two operable thermocouples per core quadrant) should be assured during the extension period. Therefore, the staff finds the licensee's request for a one-time surveillance extension of the thermocouples acceptable.

Group (7) Reactor Vessel Level Indication System

The proposed amendment requests a 3½-month extension for the channel calibration of the Reactor Vessel Level Indication System (RVLIS) required by TS Table 4.3-10, Item 16. The calibration cannot be performed at power because entry into the lower containment volume and reactor head areas is required, and these areas are accessible only during shutdown.

RVLIS has two trains of indication which are subject to TS required monthly channel checks. These channel checks would be expected to provide indication of significant degradation of the system. Additional indication of inadequate core cooling is available to the operators in the form of the core exit thermocouple readings and subcooling margin monitors. The licensee performed a review of the maintenance history of the system and, based on this review, believes that the equipment will remain operable during the extension period. No drift data was available for the current transmitters installed in RVLIS because the transmitters were replaced during the most recent refueling outage as part of its 10-year recommended replacement frequency. However, the licensee believes that it would be aware of any excessive drift through the TS required monthly channel checks. Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

Group (8) Rod Position Indication System

The proposed amendment requests a 3-month delay of the functional testing of the rod position indicator (RPI) channels required every 18 months by TS 4.1.3.3. The surveillance cannot be performed at power because it requires full insertion of the control rods, which would violate TS rod insertion limits.

The operability of the RPI channels is functionally verified per TS once per 12 hours by comparison to the demand position indication system. Also, during the 31-day surveillance per TS 4.1.3.1.2, the rods are moved at least eight steps and the RPI meters are verified to track with the demand position. These comparisons would be expected to indicate significant degradation in the RPI channels. Other surveillances, such as flux maps, quadrant power tilt ratio and axial flux difference, are performed to indicate if the core is performing as designed. These surveillances would be expected to indicate significant discrepancies between indicated and actual rod position. Because other TS required surveillances used to verify the operability of the RPIs will continue to be performed, the licensee believes that the RPIs will continue to remain operable per TS 3/4.1.3.2 during the extension. Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

Group (9) RHR Auto-Closure Interlock

In a letter dated September 28, 1993, the licensee withdrew its request for extensions of two surveillances dealing with RHR auto-closure interlock testing because they were performed during a recent forced outage. These extensions are no longer needed.

Group (10) Visual Inspection of Inaccessible Snubbers

The proposed amendment would delay visual inspection of inaccessible snubbers required by TS 4.7.7.1.a. The extension is needed from March 19, 1994, through the Unit 2 refueling outage. The extension is required because the snubbers are inaccessible during reactor operation, thus requiring the inspections to be performed during shutdown. Functional testing of snubbers per TS 4.7.7.1.c is not required until after the scheduled refueling outage start date.

It should be noted that the licensee currently has a submittal with the NRC dated May 1, 1992, requesting a permanent change to the surveillance intervals for snubber visual inspections. The submittal is based on the guidance from Generic Letter 90-09, "Alternate Requirements for Snubber Visual Inspections Intervals and Corrective Actions." The submittal would implement up to a maximum interval of 48 months for visual inspections of inaccessible snubbers, which would be well beyond the scheduled refueling outage date.

On the basis of the history of D.C. Cook Unit 2 snubber testing and inspection results, there is high confidence in the operability of the D.C. Cook Unit 2 snubbers, and operation for approximately 5 additional months past the due date for snubber visual inspections will not result in a significant decrease in plant safety. Therefore, plant shutdown to perform snubber visual inspections at the due dates indicated above would be unwarranted and the licensee's requested extension is acceptable.

Group (12) Divider Barrier Seal Inspection

The proposed amendment would delay visual inspection and coupon removal and testing of the divider barrier seal required by TS 4.6.5.9, which is required to be performed while shut down. The extension is needed from March 8, 1994, until the Unit 2 refueling outage. The entire divider barrier seal was replaced during the cycle 7-8 refueling outage. Subsequent inspection of the seal and testing of the coupons following the last outage revealed no degradation of the seal and all acceptance criteria were satisfied. Based on the facts that the divider barrier seal is passive and not subjected to any outside forces other than the environment, and is new and has shown no degradation, the licensee believes there is no reason to suspect that it would not be operable during the extension period. Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

Group (13) Reactor Coolant Pump (RCP) Fire Protection Testing

The proposed amendment requests a 4-month delay of the functional testing of the RCP fire protection system required every 18 months by TS 4.7.9.2.b. In order to perform the test, instrumentation required per TS Table 3.3-11 and the fire suppression system must be made inoperable. The licensee does not consider this prudent during operation of the RCPs; therefore, the test should be performed during shutdown.

Based on the RCP sprinkler system surveillance history, the licensee has a high confidence that the system will be able to perform its intended safety function during the extension period. Also, there are seismically qualified oil collection systems on the RCPs, installed in accordance with 10 CFR Part 50, Appendix R. These systems are designed to mitigate the effects of an RCP lube oil leak. Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

Group (14) Containment Water Level Calibrations, Sump Visual Inspections and Containment Flow Monitoring System

In a letter dated December 3, 1993, the licensee withdrew its request for the extension of calibrations in TS 4.3.3.6, Table 4.3-10, Item 18, which pertains to containment water level calibrations, because it was determined that the calibrations can be performed without entry into lower containment. The licensee will therefore perform the surveillance prior to the refueling outage. This extension is no longer needed.

The proposed amendment also requests a 4½-month extension of TS 4.5.2.d.2 which requires that the sump and its inlets be subjected to an 18-month visual inspection. An extension of TS 4.5.3.1 is also needed, since it references TS 4.5.2. The inspection cannot be performed during reactor operation because entry into the containment sump area is restricted.

The visual inspection is performed to ensure that the system is clean prior to startup and was performed prior to startup after the previous refueling outage. During reactor operation, entry into the containment sump area is restricted. The licensee has strict material control requirements for entry into containment and a containment closeout tour is performed at the end of an outage to ensure that no material is left within containment, prior to establishing containment integrity, per TS 4.5.2.c. The closeout tour was last performed in August 1993 following a short duration outage. In addition, performance of previous visual inspections following reactor operation has shown that very little debris ever accumulates in the sump. The licensee has no reason to believe that the sump or its inlets would become blocked during the extension period. Therefore, the staff finds the licensee's request for a one-time extension of the visual inspections acceptable.

In its letter dated December 3, 1993, the licensee also requested an extension of calibration of containment sump flow monitoring system as required by TS 4.4.6.1.b. This surveillance cannot be performed during reactor operation since it requires entry into the lower volume of containment. The extension is needed from April 3, 1994, until the Unit 2 refueling outage.

The containment flow monitoring system is used to monitor and detect RCS leakage. Leakage rate is monitored by knowing the sump pump capacities and monitoring how long a pump runs. The licensee performed a review of the past surveillance history of this system which showed that the pumps have capacities well above their acceptance criteria. Details of this review are included in the licensee's letter dated December 3, 1993. If the pumps were to degrade, the flow rate would decrease; thus increasing the run time for the pump to deliver a given amount of water. This would result in an over-estimation of RCS leakage, which would be conservative with respect to TSs. Also, the pump run times are very short in duration (on the order of a minute per day); therefore, the licensee has no reason to believe that continued operation during the extension period would cause the pumps to become inoperable. The staff finds the licensee's request for a one-time extension of the containment sump flow monitoring system acceptable.

Group (15) Reactor Coolant Flow Calibrations

In a letter dated December 3, 1993, the licensee withdrew its request for extensions of surveillances pertaining to reactor coolant flow calibrations, because they were calibrated during October and November, 1992. The first transmitter calibration is due August 17, 1994 (after the requested extension date). The remainder of the channel can be calibrated at power. These extensions are no longer needed.

Group (16) Engineered Safety Features (ESF) Manual Trip Actuating Device Operational Test

The proposed amendment requests a 3½-month extension for the ESF manual actuation switches specified in TS 4.3.2.1.1, Table 4.3-2, Items 9.a, 9.b, 9.c, and 9.d. These tests cannot be performed at power since they would actuate their respective ESF functions.

The circuitry associated with manual actuation of the ESF functions is subject to TS required channel functional testing. The only portion of the channels that are not tested are the manual actuation switches. The manual actuation switches are highly reliable. During the entire surveillance history for both units none of the switches has ever failed a surveillance. Additionally, the licensee notes that the manual switches are a backup to automatic actuation of the same ESF functions. The automatic channels are subjected to TS required channel checks and channel functional tests. Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

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 amendment changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 41505 and 58 FR 67850). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: B. Wetzel

Date: January 26, 1994