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

Template NRR-058

March 31, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADMINISTRATIVE CHANGES (TAC NOS. MA4922 AND MA4923)

Dear Mr. Powers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 243 to Facility Operating License No. DPR-58 and Amendment No. 224 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendment consists of changes to Appendix A, Technical Specifications (TSs), in response to your application dated December 3, 1998.

The amendment would make administrative changes to several TS to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical. The proposed changes have been reviewed by the NRC staff and are in accordance with the regulations.

Amendment No. 216 for Unit 1 contains a TS page that is affected by the enclosed Amendment No. 243. Since Amendment No. 216 may not be implemented until December 31, 2000, the NRC is issuing two sets of TS pages with the enclosed amendment. The first set should be inserted when Amendment No. 243 is implemented. The second set replaces the Amendment No. 216 page that is affected by Amendment No. 243 and should be inserted into Amendment No. 216.

DF01

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

Sincerely,

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 243 to DPR-58
2. Amendment No. 224 to DPR-74
3. Safety Evaluation Report

cc w/encls: See next page

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

Sincerely,
/RA/
John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 243 to DPR-58
2. Amendment No. 224 to DPR-74
3. Safety Evaluation Report

cc w/encls: See next page

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 243
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (NRC) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated December 3, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amendment (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 the amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 243, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 31, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 243

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 0-3	3/4 0-3
3/4 3-21a	3/4 3-21a
3/4 4-38	3/4 4-38
3/4 4-40	3/4 4-40
3/4 7-15	3/4 7-15
3/4 9-1	3/4 9-1
3/4 9-13	3/4 9-13
5-6	5-6
5-7b	5-7b
6-4	6-4

SURVEILLANCE REQUIREMENTS

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Deleted

4.0.7 Deleted

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3	14*
Pump Start		2/bus (T11A – Train B; T11D – Train A)			
Valve Actuation (Both trains)		2/bus on (T11A & T11B or 2/busses T11C & T11D)			
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2	18*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kV Bus Degraded Voltage	3/Bus (T11A – Train B; T11D – Train A)	2/Bus (T11A – Train B; T11D – Train A)	2/Bus (T11A – Train B; T11D – Train A)	1, 2, 3, 4	14*

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a - MODES 1, 2, 3 and 4.
Specification 3.7.3.1.b - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling 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 3.7.3.1.b is applicable:

With no flowpath 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.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling 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.

c. By verifying pump performance pursuant to Specification 4.0.5.

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

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6

ACTION:

a. With the requirements of the above specification not satisfied, 1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, and 2) initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool† until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Deleted

2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

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* The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D.C. COOK - UNIT 2.

† This does not include the main load block. For purposes of this specification, a de-energized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-\ 26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

Figure 5.6-3 intentionally deleted.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must be qualified as specified in Section 6.2.2.g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 DELETED

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 243

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of Amendment No. 216 with the attached revised page. This page replaces Amendment No. 216 that is affected by the issuance of the enclosed Amendment No. 243. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3/4 9-1	3/4 9-1

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6

ACTION:

a. With the requirements of the above specification not satisfied, 1) immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except addition of water from the RWST, provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2, and 2) initiate and continue boration at greater than or equal to 34 gpm of 6,550 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 316

DONALD C. COOK NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (NRC) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated December 3, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amendment (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 the amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 224 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 31, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 224

TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 0-3	3/4 0-3
3/4 0-4	3/4 0-4
3/4 3-11	3/4 3-11
3/4 3-20	3/4 3-20
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-34	3/4 3-34
3/4 3-44d	3/4 3-44d
3/4 3-47	3/4 3-47
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-33	3/4 4-33
3/4 4-35	3/4 4-35
3/4 4-37	3/4 4-37
3/4 5-4	3/4 5-4
3/4 5-8	3/4 5-8
3/4 6-12	3/4 6-12
3/4 6-14	3/4 6-14
3/4 6-47	3/4 6-47
3/4 7-12	3/4 7-12
3/4 7-13	3/4 7-13
3/4 7-16a	3/4 7-16a
3/4 7-20	3/4 7-20
3/4 8-13	3/4 8-13
3/4 8-15	3/4 8-15
3/4 9-12	3/4 9-12
5-6	5-6
5-8	5-8
6-4	6-4

SURVEILLANCE REQUIREMENTS

b. Surveillance Intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Deleted

4.0.7 Deleted

SURVEILLANCE REQUIREMENTS

4.0.8 Deleted

4.0.9 Deleted

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

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kV Bus Degraded Voltage	3/Bus (T11A-Train B) (T11D-Train A)	2/Bus (T11A-Train B) (T11D-Train A)	2/Bus (T11A-Train B) (T11D-Train A)	1, 2, 3, 4	14*
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	2/train	1/train	2/train	1, 2, 3, 4	18
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
d. Steam Line Isolation	2/steam line (1 per train)	2/steam line (1 per train)	2/operating steam line (1 per train)	1, 2, 3	20

TABLE 4.3-2
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	---------- See Functional Unit 9 ----------				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure -- High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure -- Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines -- High	S	R	M	N.A.	1, 2, 3
f. Steam Line Pressure -- Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	---------- See Functional Unit 9 ----------				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure -- High-High	S	R	M(3)	N.A.	1, 2, 3
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	---------- See Functional Unit 9 ----------				
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
b. Phase "B" Isolation					
1) Manual	---------- See Functional Unit 9 ----------				
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
3) Containment Pressure-- High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Purge and Exhaust Isolation					
1) Manual	---------- See Functional Unit 9 ----------				
2) Containment Radioactivity -- High	S	R	Q	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 Pressure -- 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

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

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.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

		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)	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.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 6.9.1	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3.	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1.	N/A	N/A

$S = 3(N+n)\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

g. With PORVs and block valves not in the same line inoperable due to causes other than excessive seat leakage, within 1 hour restore the valves to OPERABLE status or close and de-energize the associated block valve and place the associated PORV in manual control in each respective line. Apply the portions of ACTION c or d above, relating to the OPERATIONAL MODE, as appropriate for two or three lines unavailable.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each PORV 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 operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

c. At least once per 18 months by operating solenoid air control valves and check valves in PORV control systems through one complete cycle of full travel, and

d. At least once per 18 months by performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, c, or d in Specification 3.4.11.

4.4.11.3 Deleted.

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. IMO-390	a. RWST to RHR	a. Open
b. IMO-315	b. Low head SI to Hot Leg	b. Closed
c. IMO-325	c. Low head SI to Hot Leg	c. Closed
d. IMO-262*	d. Mini flow line	d. Open
e. IMO-263*	e. Mini flow line	e. Open
f. IMO-261*	f. SI Suction	f. Open
g. ICM-305*	g. Sump Line	g. Closed
h. ICM-306*	h. Sump Line	h. Closed

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

* These valves must change position during the switchover from injection to recirculation flow following LOCA.

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.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.

d. At least once per 5 years by verifying the flow rate from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve specified 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 containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flow path in support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a. - MODES 1, 2, 3, 4.
Specification 3.7.3.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling 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 3.7.3.1.b is applicable:

With no flowpath to Unit 1 available, return at least one flowpath 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.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling 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.

c. By verifying pump performance pursuant to Specification 4.0.5.

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

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.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.13 volts under float charge.

2. The specific gravity, corrected to 77°F, and full electrolyte level (fluid at the bottom of the maximum level indication mark), of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the top of the minimum level indication mark and the bottom of the maximum level indication mark.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The battery charger will supply at least 140 amperes at greater than or equal to 250 volts for at least 4 hours.

d. At least once per 18 months, perform a battery service test during shutdown (MODES 5 or 6), by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status the actual or simulated emergency loads for the design duty cycle which is based on the composite load profile. The composite load profile envelopes both the LOCA/LOOP and Station Blackout profiles and provides the basis for the times listed in Table 4.8-2. The battery charger will be disconnected throughout the test. The battery terminal voltage shall be maintained greater than or equal to 210 volts throughout this test.

e. At least once per 60 months, conduct a performance test of battery capacity during shutdown (MODES 5 or 6), by verifying that the battery capacity is at least 80% of the manufacturer's rating. When this test is performed in place of a battery service test, a modified performance test shall be conducted.

Annual performance tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity, and has shown no signs of degradation, performance testing at two year intervals is acceptable until the battery shows signs of degradation.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 250-volt D.C. bus, and

1 - 250-volt battery bank and charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 250-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 250-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool+ until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Deleted.

2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

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* The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D. C. COOK - UNIT 1.

+ This does not include the main load block. For purposes of this specification, a de-energized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

CRITICALITY - SPENT FUEL (Continued)

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$- 22{,}670 + 22{,}220\ E - 2{,}260\ E^2 + 149\ E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$- 26{,}745 + 18{,}746\ E - 1{,}631\ E^2 + 98.4\ E^3$$

Where E = Initial Peak Enrichment

5.6.1.2 Fuel stored in the spent fuel storage racks shall have a nominal fuel assembly enrichment as follows:

	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
1)	Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2)	Exxon/ANF	15 x 15	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

Figure 5.6-3 intentionally deleted.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must be qualified as specified in Section 6.2.2.g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO243 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 224TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated December 3, 1998, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2 (D.C. Cook). The proposed amendments would make administrative changes to several TSs to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard TSs, provide clarification, and correct typographical errors.

2.0 EVALUATION

The evaluation of the proposed changes are described in the following paragraphs:

A. Proposed Revision to Boron Sampling Requirements in Mode 6

The current Unit 1 TS Surveillance Requirement 4.9.1.2 states that "The boron concentration of the reactor coolant system and refueling canal shall be determined by chemical analysis at least three times per seven days with a maximum time interval between samples of 72 hours." The licensee proposes to change the TS to read as "The boron concentration of the reactor coolant system and refueling canal shall be determined by chemical analysis at least once per 72 hours."

The purpose of TS 4.9.1.2 is to assure that a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The licensee has proposed to remove the Unit 1 restriction to determine the concentration at least three times per seven days in order to maintain consistency with the Unit 2 surveillance requirement, NUREG-1431, "Standard Technical Specifications," and NUREG-0452, "Standard Technical Specifications," Revision 4, fall 1981. The 72-hour maximum interval between samples is not changed. NUREG-1431 states that a minimum frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The

change does not constitute a reduction in safety. Therefore, the staff finds the proposed change acceptable.

B. Proposed Revision to Footnote for TS 3.9.12, Action a

TS 3.9.12, Action a, is modified by a footnote describing operation of the drumming room roll-up door. The current footnote states that "The crane bay roll-up door and the drumming room roll-up door may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool." The licensee has proposed to revise the footnote of TS 3.9.1.2, Action a, to read "The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool."

In the NRC's Safety Evaluation, the staff found that the operation of the crane bay roll-up door and drumming room roll-up door met the intent of Standard Review Plan, Sections 9.4.2, "Spent Fuel Pool Area Ventilation System," and 15.7.4, "Radiological Consequences of Fuel Handling Accidents," and was acceptable as such. The licensee proposes to remove "roll-up" from the description of the drumming room door. The drumming room door has been replaced by a door having a different design, although the function of the door remains the same. The replacement door provides a ventilation barrier as required by the analysis used in support of the previous amendment request. In addition, the name of the door is proposed to change to "south door of the auxiliary building crane bay" because it more accurately describes the door's current use. The staff finds the proposed changes to be acceptable, as the changes do not constitute a reduction in safety and clarify the TS.

C. Proposed Change to Figure 5.6-3

TS 5.6.1.1.c.3 includes equations for equivalent reactivity criteria for Region 2 and Region 3 in the spent fuel storage racks. The equations are also graphically depicted in Figure 5.6-3. Either the equations or the graph can be used to verify that fuel is stored in the appropriate region. The licensee proposes to delete Figure 5.6-3, as the information is redundant to the equations provided in TS 5.6.1.1.c.3.

The deletion does not alter the fuel storage requirements TSs and does not constitute a reduction in safety. Therefore, the staff finds the proposed change to be acceptable. The change is merely proposed to reduce unnecessary information in the TSs.

D. Proposed Change to Reference in TS 6.3.1

Current TS 6.3.1, "Facility Staff Qualifications," includes a requirement that the operations superintendent must hold or have held a senior operator license as specified in TS 6.2.2.h. The reference to 6.2.2.h is an administrative error made in previous amendments. Section 6.2.2.h does not exist. The proposed change provides a clear reference to the correct TS Section 6.2.2.g for the operations superintendent qualifications and does not change the current TS requirements. Therefore, the staff finds the change to be acceptable.

E. Proposed Deletion of Obsolete Notes

The following paragraphs state the proposed TS deletion and justification:

Current Surveillance Extensions Unit 1, TS 4.0.6 and 4.0.7

Unit 1, TS 4.0.6 states that "Amendments 100, 107 and 108 grant extensions for certain surveillances required to be performed on or before July 31, 1987, and until the end of the Cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1987 refueling outage."

TS 4.0.7 states that "Amendment 121 granted extensions for certain surveillances required to be performed on or before April 1, 1989, until the end of the Cycle 10-11 refueling outage. For these specific surveillances under this extension, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1989 refueling outage."

Current Surveillance Extensions Unit 2, TS 4.0.6, 4.0.7, 4.0.8, 4.0.9

For Unit 2, TS 4.0.6 states that "Amendment 78 granted extensions for certain surveillances required to be performed on or before March 31, 1986, until the end of the Cycle 5-6 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1986 refueling outage."

TS 4.0.7 states that "Amendments 97 and 99 granted extensions for certain surveillances required to be performed on or before July 1, 1988, until the end of the Cycle 6-7 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillances date during the Unit 2 1988 refueling outage."

TS 4.0.8 states that "By specific reference to this section, those surveillances which must be performed on or before August 13, 1994, and are designed as 18-month or 36-month surveillances (or required as outage-related surveillances under the provisions of Specification 4.0.5) may be delayed until the end of the Cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1994 refueling outage."

TS 4.0.9 states that "By specific reference to this section, those surveillances which must be performed on or before September 7, 1994, are designated as 18-month surveillances, so it may be delayed until just prior to core reload in the Unit 2 Cycle 9-10 refueling outage."

Justification for Deletion of Unit 1 TS 4.0.6, 4.0.7 and Unit 2 TS 4.0.6, 4.0.7, 4.0.8, and 4.0.9

Unit 1 TS 4.0.6 and TS 4.0.7, and Unit 2 TS 4.0.6, 4.0.7, 4.0.8, and 4.0.9 extensions were granted to accommodate scheduled work at the time and have been proposed to be deleted because they are no longer applicable. The references to Unit 1 TS 4.0.6 and 4.0.7 that indicated when the provision was applicable were deleted in previous amendments. The existing TS notes are no longer applicable and serve no function. For Unit 2, TSs 4.0.8 and 4.0.9 are proposed to be deleted because they no longer apply. References to TS 4.0.8 are also deleted from the following: Table 4.3-1, functional Units 7 through 11; Table 4.3-2, functional Units 1.d, 4.d, and 6.d; Table 4.3-6A, instruments 5 through 8; Table 4.3-10, items 2,

3, 11, 15, and 16; surveillance requirement (SR) 4.4.6.1.b; SR 4.4.11.1.d; SR 4.5.3.1; SR 4.6.2.2c; SR 4.6.3.1.2; SR 4.6.5.9; SR 4.7.3.1.b; SR 4.7.4.1.b; SR 4.7.5.1.e.2.a; SR 4.7.5.1.e.2.b; SR 4.7.7.1.a; and SR 4.8.1.2. Additionally, references to TS 4.0.9 are proposed to be deleted from SR 4.8.2.3.2.d and SR 4.8.2.4.2. The proposed changes are acceptable because all extensions mentioned above pertained to past refueling outages and are no longer applicable since the corresponding refueling outages have been completed. Deletion of the TSs do not eliminate any requirements. Therefore, the staff finds the proposed changes acceptable.

Surveillance Extensions Notes Unit 1 and 2, TS 4.4.12.1 and 4.4.12.2

For Unit 1 and 2, notes for SRs in 4.4.12.1 state: "Surveillance requirements to demonstrate the operability of each Reactor Vessel head vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification, and after the appropriate Plant Procedures have been written."

TS note 4.4.12.2 states: "Surveillance requirements to demonstrate the operability of each Pressurizer steam space vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification and after the appropriate Plant Procedures have been written."

Justification for Deletion of Notes for Unit 1 and Unit 2, TS 4.4.12.1 and 4.4.12.2

For Unit 1 and Unit 2, the notes for SRs 4.4.12.1 and 4.4.12.2 are proposed to be deleted. Plant Procedures 01-OHP-4030.STP.56 and 02-OHP-4030.STP.56 were developed to perform the surveillance. The surveillances are performed routinely as required and the exception allowed in the footnote is no longer applicable; therefore, the deletion of the notes for SRs in TS 4.4.12.1 and 4.4.12.2 are acceptable. The surveillance notes were written for a one-time relief extension and the relief extension time periods have expired. Deletion of the surveillance notes do not eliminate any requirements; therefore, the staff finds the proposed changes acceptable.

Note to Table 4.4-2, Unit 2

Table 4.4-2 and TS 3.3.3.1 include a note: "This Technical Specification will not be effective until after the 1982 refueling outage."

Justification for Deletion of Note to Table 4.4-2, Unit 2

For Unit 2, the notes for TS 3.3.3.1 and Table 4.4-2 are deleted. The provision has expired and it is no longer required to be included in the TSs. The staff finds the proposed change acceptable.

The exceptions that were granted in the above paragraphs have all expired and are no longer applicable. The changes do not represent a reduction in safety and deletion of the notes or TSs do not eliminate any requirements of the TSs. Therefore, the staff finds the proposed changes acceptable.

F. Proposed Changes to TS SR 4.7.3.1

Unit 1, TS 4.7.3.1

Unit 1 TS SR 4.7.3.1.d states: "At least once per 18 months during shutdown, by verifying that the cross-tie valves can cycle full travel." The licensee proposes to change the TS to "At least once per 18 months during shutdown, by verifying that the unit cross-tie valves can cycle full travel."

The clarification is proposed to indicate that the cross-tie valves are the unit cross-tie valves. The change is consistent with the equivalent Unit 2 requirement and is not intended to affect which valves are included in the surveillance. The staff finds the proposed change to be acceptable, as it does not eliminate or alter the TS requirement and allows for a clear reference in the TS.

Unit 2, TS 4.7.3.1

Unit 2 TS SR 4.7.3.1 reads: "At least two component cooling water loops shall be demonstrated OPERABLE," and has two requirements, 4.7.3.1.a and 4.7.3.1.b, to demonstrate operability of the component cooling water loops. A third requirement is included in SR 4.7.3.2. This requirement supports demonstrating that the system is operable. For Unit 2, the licensee proposes to add a new SR, 4.7.1.3.c ("By verifying pump performance pursuant to Specification 4.0.5"), to demonstrate operability by verifying pump performance pursuant to TS 4.0.5, and to renumber TS 4.7.3.2 as 4.7.3.1.d for consistency with Unit 1 TSs.

This change is consistent with the corresponding surveillance for Unit 1. In Amendment No. 164 to DPR-58 and Amendment No. 149 to DPR-74, the Nuclear Regulatory Commission approved changes requiring that all safety-related pumps in the TSs be tested at a frequency specified in TS 4.0.5. TS 4.0.5 states that safety-related pumps shall be tested in accordance with ASME Code, Section XI, unless written relief has been granted. The proposed changes are consistent with the changes approved in those amendments. There is no reduction in safety and deletion of the SRs do not eliminate or alter the TS requirement. Therefore, the staff finds the proposed change acceptable.

G. Proposed Change to Degraded Bus Voltage Instrumentation

Current Unit 1 and Unit 2 Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation," provides various instrumentation requirements. Functional Unit 8.b of Table 3.3-3 provides instrumentation requirements for the 4 kV bus degraded voltage (loss of power). On the 4 kV bus, there are three channels per bus total, with two channels per bus being required for operation in Modes 1, 2, 3, and 4. Two channels per bus are required to trip. These requirements provide assurance that the actuation will occur when required. The three channels are installed on buses T11A and T11D for Unit 1 and on buses T21A and T21D for Unit 2. The licensee proposes to add clarifying information to Table 3.3-3, to reflect the configuration of the instrumentation used to detect degraded voltage.

The configuration of the instrumentation used to detect degraded voltage may be unclear in Table 3.3-3. By design, the instrumentation is not installed on buses T11B or T11C for Unit 1 or on buses T21B or T21C for Unit 2. The licensee proposes to add references to the buses

with the appropriate instrumentation. This is similar to functional Unit 6.b, which has instrumentation requirements for 4 kV bus loss of voltage. The design and function of Unit 6.b has been reviewed and approved in the Safety Evaluation for Amendments No. 39 for Unit 1 and Amendment No. 22 for Unit 2. The proposed change is intended to clarify functional unit 8.b by indicating that the instrumentation is installed only on buses T11A and T11D. The change does not represent a reduction in safety, and allows for clarity in the TS. Therefore, the staff finds the proposed change acceptable.

H. Proposed Corrections to Typographical Errors

The licensee proposes to correct two typographical errors that were introduced in Amendment No. 131 to DPR-74 for Unit 2. These errors were to portions of the text that were not affected by the amendment. The proposed changes restore the text as it was issued in the previous amendment (Amendment 78). The valve number for line d of SR 4.5.2.a is changed from IMP-262 to IMO-262. Furthermore, the word "otherwise" is corrected in SR 4.5.2.b from the incorrect spelling of "otherswise."

Additionally, corrections are proposed to change "once" to "one" and "with" to "when" in TS 3.7.3.1 for Unit 1 and "in" to "to" in TS 3.7.3.1 for Unit 2.

These changes are acceptable since the changes are editorial and do not impact the requirements. The changes are intended to provide clarification and better direction to the operators. There is no reduction in safety by this change, therefore, the staff finds the proposed changes acceptable.

3.0 SUMMARY

The licensee has proposed changes to make several administrative changes to TSs for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed changes include: (1) revising boron sampling requirements in mode 6; (2) deleting a reference to obsolete equipment in a footnote, (3) deleting a redundant figure; (4) correcting a reference to another requirement; (5) deleting obsolete notes; (6) adding to SRs; (7) clarifying instrumentation configuration; and (8) correcting typographical errors. These changes are proposed to remove obsolete information, provide consistency between Unit 1 and Unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors. The proposed amendment does not cause changes to accident initiators or precursors, or to the accident analyses, and does not involve a significant reduction of safety.

Based on the above evaluation, the staff finds that the proposed changes to the TSs are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 50.35, an environmental assessment and finding of no significant impact have been prepared and published in the *Federal Register* on March 28, 2000 (65 FR 16421). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 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.

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