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April 13, 1983

Docket Nos. 50-315
and 50-316

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

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 71 to Facility Operating License No. DPR-58 and Amendment No. 53 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 11, 1981, as supplemented February 25, 1983.

These amendments revise the Technical Specifications to reflect revised surveillance requirements for safety-related snubbers. These amendments also add a section 4.0.5 to the Unit 1 Technical Specifications on surveillance requirements for inservice inspection and testing of components and deletes a portion of section 4.0.4 in Unit 1 and 4.0.5.a.1 in Unit 2, which are no longer applicable. These latter changes are made to clarify the Technical Specifications and make them consistent with Standard Technical Specification requirements.

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

Sincerely,

/S/

David L. Wigginton, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 71 to DPR-58
2. Amendment No. 53 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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Mr. John Dolan
Indiana and Michigan Electric Company

cc: Mr. M. P. Alexich
Assistant Vice President
for Nuclear Engineering
American Electric Power
Service Corporation
2 Broadway
New York, New York 10004

Mr. William R. Rustem (2)
Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Mr. Wade Schuler, Supervisor
Lake Township
Baroda, Michigan 49101

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
P. O. Box 458
Bridgman, Michigan 49106

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

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Honorable James Bemenek, Mayor
City of Bridgman, Michigan 49106

U.S. Environmental Protection Agency
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

Maurice S. Reizen, M.D.
Director
Department of Public Health
P.O. Box 30035
Lansing, Michigan 48109

William J. Scanlon, Esquire
2034 Pauline Boulevard
Ann Arbor, Michigan 48103

The Honorable Tom Corcoran
United States House of Representatives
Washington, D. C. 20515

James G. Keppler
Regional Administrator - Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. DPR-58

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated November 11, 1981, as supplemented February 25, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 71, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 0-2	3/4 0-2
3/4 0-3	3/4 0-3
3/4 7-27*	3/4 7-27*
3/4 7-28	3/4 7-28
3/4 7-29	3/4 7-29
3/4 7-30	3/4 7-30
3/4 7-31	3/4 7-31
3/4 7-32	3/4 7-32
3/4 7-33	3/4 7-33
3/4 7-34	3/4 7-34
3/4 7-35	3/4 7-35
3/4 7-36	3/4 7-36
3/4 7-37	3/4 7-37
3/4 7-38	3/4 7-38
3/4 7-39	3/4 7-39
3/4 7-40	3/4 7-40
-	3/4 7-40a
6-20	6-20
B 3/4 7-5*	B 3/4 7-5*
B 3/4 7-6	B 3/4 7-6

*Included for convenience

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

*4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

* For Unit 1 and until the Technical Specification can be revised to reference Section 4.0.5 or to specifically exempt certain actions or Surveillance Requirements from Section 4.0.5 requirements, this Section will apply only to Technical Specifications 3/4 7.8, 3/4 4.1 and 3/4 9.8.

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 5 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.

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources - Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.7.7.1.3 Reports - A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers listed in Table 3.7-4 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 4.7.8.C on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 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 Inspections

The first inservice visual inspection of snubbers shall be performed after four months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Table 3.7-4. If less than two (2) snubbers are found inoperable during the first inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

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 may be determined OPERABLE 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 particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.8.d However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8.d an additional 10% of that type of Snubber shall be functionally tested).

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)

3. Snubbers within 10 feet of the discharge from a safety relief valve

Snubbers identified in Table 3.7-4 as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design efficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleratioh in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Table 3.7-4 and shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
1	1-GRC-S519	Reactor Coolant Elev. 683'-5 1/2" in pressurizer enclosure	I	No	No
2	1-GRC-S537	Reactor Coolant Az 25° Elev. 610'-5" Between stem. Gen. No. 1 and RC pump No. 1	I	Yes	No
3	1-GRC-S538	Reactor Coolant Az 41° Elev. 614'-10" Below stm. Gen. No. 1	I	No	No
4	1-GRC-S555	Reactor Coolant Az 141° Elev. 614'-2" Below stm. Gen. No. 2	I	No	No
5	1-GRC-S562	Reactor Coolant Az 154° Elev. 610'-5" between stm. Gen. No. 2 and RC pump No. 2	I	Yes	No
6	1-GRC-S564	Reactor Coolant Az 313° Elev. 614'-10 1/8" below stm. Gen. No. 4	I	No	No
7	1-GRC-S566	Reactor Coolant Az 332° Elev. 610'-5" between stm. Gen. No. 4 and RC pump No. 4	I	Yes	No
8	1-GRC-S573	Reactor Coolant Az 223° Elev. 614'-10 1/8" below stm. Gen. No. 3	I	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
9	1-GRC-S575	Reactor Coolant Az 208° Elev. 610'-5" between stm. Gen. No. 3 and RC pump No. 3	I	Yes	No
10	1-GRC-S582	Reactor Coolant Az 212° Elev. 617'-4" near Reactor Cavity Wall, across from Stm. Gen. No. 3	I	Yes	No
11	1-GRC-S587	Reactor Coolant Az 260° Elev. 622'-4 1/4" in Containment	I	No	No
12	1-GRC-S592	Reactor Coolant Az 292° Elev. 683'-6 3/4 in Pressurizer Enclosure.	I	No	No
13	1-GRC-S594	Reactor Coolant Az 292° Elev. 691'-9" in Pressurizer Enclosure.	I	No	No
14	1-GRC-S596	Reactor Coolant Az 285° Elev. 691'-9" in Pressurizer Enclosure.	I	No	No
15	1-GRC-S598	Reactor Coolant Az 292° Elev. 670'-3 3/4" in Pressurizer Enclosure.	I	No	Yes
16	1-GRC-S599	Reactor Coolant Az 287° Elev. 672'-4" in Pressurizer Enclosure.	I	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
17	1-GRC-S604	Reactor Coolant Az 286° Elev. 688'-10" in Pressurizer Enclosure.	I	No	No
18	1-GRC-S608	Reactor Coolant Az 286° Elev. 693'-0" in Pressurizer Enclosure.	I	No	No
19	1-GRC-S614	Reactor Coolant Az 282° Elev. 681'-0" in Pressurizer Enclosure.	I	No	No
20	1-FW-S1	Feedwater Az 31° Elev. 634'-9" Behind Stm Gen. No. 1	I	No	No
21	1-FW-S2(L)	Feedwater Az 26° Elev. 633'-6" Behind Stm. Gen. No. 1	I	No	No
22	1-FW-S2(U)	Feedwater Az 26° Elev. 636'-0" Behind Stm Gen. No. 1	I	No	No
23	1-FW-S3	Feedwater Az 20° Elev. 629'-9" Behind Stm. Gen. No. 1	I	No	No
24	1-FW-S4(L)	Feedwater Az 154° Elev. 636'-8 3/8" Behind Stm. Gen. No. 2	I	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
25	1-FW-S4(U)	Feedwater Az 154° Elev. 640'-8 3/8" Behind Stm. Gen. No. 2	I	No	No
26	1-FW-S5	Feedwater Az 163° Elev. 634'-9" Behind Stm. Gen. No. 2	I	No	No
27	1-FW-S6	Feedwater Az 157° Elev. 629'-9" Behind Stm. Gen. No. 2	I	No	No
28	1-FW-S7	Feedwater Az 204° Elev. 634'-9" Behind Stm. Gen. No. 3	I	No	No
29	1-FW-S8(L)	Feedwater Az 200° Elev. 633'-6" Behind Stm. Gen. No. 3	I	No	No
30	1-FW-S8(U)	Feedwater Az 200° Elev. 636'-0" Behind Stm. Gen. No. 3	I	No	No
31	1-FW-S9	Feedwater Az 194° Elev. 629'-9" Behind Stm. Gen. No. 3	I	No	No
32	1-FW-S10(L)	Feedwater Az 334° Elev. 633'-6" Behind Stm. Gen. No. 4	I	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
33	1-FW-S10(U)	Feedwater Az 334° Elev. 636'-0" Behind Stm. Gen. No. 4	I	No	No
34	1-FW-S11	Feedwater Az 330° Elev. 634'-9" Behind Stm. Gen. No. 4	I	No	No
35	1-FW-S12	Feedwater Az 343° Elev. 629'-9" Behind Stm. Gen. No. 4	I	No	No
36	1-GCS-S634	Chem&Vol. Control Az 292° Elev. 613' in Containment	I	Yes	No
37	1-GCS-S637	Chem&Vol. Control Az 71° Elev. 608'-10" in Annulus.	A	No	No
38	1-GCS-S757	RC pump Seal Water Supply, Between RC Pump No. 2 and Crane Wall, immediately under grating Az 128° Elev.612'-7 1/8"	I	No	No
39	1-MSS-1	Main steam Az 8° Elev. 639'-1 1/4" between Stm Gen. No. 1 and No. 4	I	No	No
40	1-MSS-2	Main steam Az 17° Elev. 635' between Stm. Gen. No. 1 and No. 4	I	No	No
41	1-MSS-3	Main steam Az 172° Elev. 639'-1 1/4" Behind Stm. Gen. No. 1	I	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
42	1-MSS-4	Main steam Az 165° Elev. 635' Between Stm Gen. No. 2 and No.3	I	No	No
43	1-MSS-5	Main steam Az 191° Elev. 635' Between Stm.Gen.No.2 and No.3	I	No	No
44	1-MSS-6	Main steam Az 184° Elev. 639'-1 1/4" between Stm. Gen. No. 2 and No. 3	I	No	No
45	1-MSS-7	Main steam Az 349° Elev. 635' Behind Stm.Gen.No.1 and No.4	I	No	No
46	1-MSS-8	Main steam Az 356° Elev. 639'-1 1/4" between Stm Gen. No. 1 and No. 4	A	No	No
47	1-GCCW-S278	Component Cooling Water Elev. 609' in CCW pump area	A	No	No
48	1-GCCW-S309	Component Cooling Water Elev. 597'-1 5/8" in passageway near sampling room Aux. Bldg.	A	No	Yes
49	1-GCCW-S837	Component Cooling Water Elev. 621'-0" in CCW pump area	A	No	No
50	1-GCCW-S838	Component Cooling Water Elev. 621'-0" in CCW pump area	A	No	No
51	1-GCCW-S839	Component Cooling Water Elev. 621'-0" in CCW pump Area	A	No	No

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

ER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
-S840	Component Cooling Water Elev. 621'-0" in CCW pump area	A	No	No
-S841	Component Cooling Water Elev. 621'-0" in CCW pump area	A	No	No
-S842	Component Cooling Water Elev. 621'-0" in CCW pump area	A	No	No
-S844	Component Cooling Water Elev. 609'-0" in CCW pump area	A	No	No
S563	Stm. Gen. Blowdown Az 277° Elev. 608'-6 1/2" in Annulus	A	No	No
S569	Stm. Gen. Blowdown Az 278° Elev. 608'-6 1/2" in Annulus	A	No	No
S573	Stm. Gen. Blowdown Az 181° Elev. 607'-10 1/2" in Annulus	A	No	No
S574	Stm. Gen. Blowdown Az 181° Elev. 607'-10 1/2" in Annulus	A	No	No
S7A	Residual Heat Removal Elev. 581'-8 1/2" in 1-E RHR pump room	A	No	No
S7B	Residual Heat Removal Elev. 581'-4" in 1-E RHR pump room	A	No	No

D.C. COOK PLANT - UNIT NO. 1

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
79	N/A	Steam Generator No. 1 Elev. 665'.	I	No	Yes
80	N/A	Steam Generator No. 1 Elev. 665'.	I	No	Yes
81	N/A	Steam Generator No. 1 Elev. 665'.	I	No	Yes
82	N/A	Steam Generator No. 1 Elev. 665'	I	No	Yes
83	N/A	Steam Generator No. 2 Elev. 665'	I	No	Yes
84	N/A	Steam Generator No. 2 Elev. 665'.	I	No	Yes
85	N/A	Steam Generator No. 2 Elev. 665'.	I	No	Yes
86	N/A	Steam Generator No. 2 Elev. 665'.	I	No	Yes
87	N/A	Steam Generator No. 3 Elev. 665'.	I	No	Yes
88	N/A	Steam Generator No. 3 Elev. 665'.	I	No	Yes
89	N/A	Steam Generator No. 3 Elev. 665'.	I	No	Yes

D.C. COOK PLANT - UNIT NO. 1

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
90	N/A	Steam Generator No. 3 Elev. 665'.	I	No	Yes
91	N/A	Steam Generator No. 4 Elev. 665'.	I	No	Yes
92	N/A	Steam Generator No. 4 Elev. 665'.	I	No	Yes
93	N/A	Steam Generator No. 4 Elev. 665'.	I	No	Yes
94	N/A	Steam Generator No. 4 Elev. 665'.	I	No	Yes

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4 provided that a revision to Table 3.7-4 is included with the next License Amendment request.

Modifications to the "High Radiation Zone" column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4 is included with the next License Amendment request.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components identified in Table 5.9-1.

f. Records of training and qualification for current members of the plant staff.

g. Records of in-service inspections performed pursuant to these Technical Specifications.

h. Records of Quality Assurance activities required by the QA Manual.

i. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

j. Records of meetings of the PNSRC and the NSDRC.

k. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

l. **Records of the service lives of hydraulic snubbers listed on Table 3.7-4 including the date at which service life commences and associated installation and maintenance records.**

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consisten with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Quantities of interest to this specification which are exempt from the leakage testing are consistent with the criteria of 10 CFR Parts 30.11-20 and 70.19. Leakage from sources excluded from the requirements of this specification is not likely to represent more than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3/4.7.8 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure of failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results required a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufactured input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated November 11, 1981, as supplemented February 25, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7.1 All snubbers listed in Table 3.7-9 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 4.7.7.1c 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 Inspections

The first inservice visual inspection of snubbers shall be performed after four months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Table 3.7-9. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

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 may be determined OPERABLE 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 particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.7.1.d as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.7.1.d an additional 10% of that type of snubber shall be functionally tested).

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)

3. Snubbers within 10 feet of the discharge from a safety relief valve

Snubbers identified in Table 3.7-9 as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Table 3.7-9 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
36	2-FW-S9	Feedwater Az 194° Elev. 629'-9" near Stm. Gen. No. 3	I	NO	NO
37	2-FW-S10(L)	Feedwater Az 334° Elev. 634'-0" near Stm. Gen. No. 4	I	NO	NO
38	2-FW-S10(U)	Feedwater Az 334° Elev. 636'-7" near Stm. Gen. No. 4	I	NO	NO
39	2-FW-S11	Feedwater Az 330° Elev. 634'-9" near Stm. Gen. No. 4	I	NO	NO
40	2-FW-S12	Feedwater Az 343° Elev. 629'-9" near Stm. Gen. No. 4	I	NO	NO
41	2-GBD-S563(L)	Stm. Gen. Blowdown Az 275° Elev. 607'-11" in Annulus	I	NO	NO
42	2-GBD-S563(U)	Stm. Gen. Blowdown Az 275° Elev. 608'-6" in Annulus	I	NO	NO
43	2-GBD-S569(L)	Stm. Gen. Blowdown Az 275° Elev. 607'-11" in Annulus	I	NO	NO
44	2-GBD-S569(U)	Stm. Gen. Blowdown Az 275° Elev. 608'-6" in Annulus	I	NO	NO
45	2-GBD-S568	Stm. Gen. Blowdown Az 264° Elev. 608'-1" in Annulus	I	NO	NO

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
46	2-GRH-S6	Residual Heat Removal Elev. 581'-6" RHR Pump Room 2E	A	NO	NO
47	2-GRH-S7	Residual Heat Removal Elev. 581'-3" RHR Pump Room 2E	A	NO	NO
48	2-GRH-S24	Residual Heat Removal Elev. 581'-0" RHR Pump Room 2W	A	NO	NO
49	2-GRH-S25	Residual Heat Removal Elev. 580'-6" RHR Pump Room 2W	A	NO	NO
50	2-GCCW-S274	Component Cooling Water Elev. 621'-0" CCW Pump Area	A	NO	NO
51	2-GCCW-S308	Component Cooling Water Elev. 610'- 1/2" CCW Pump Area	A	NO	NO
52	2-GCCW-S317	Component Cooling Water Elev. 621'-0" CCW Pump Area	A	NO	NO
53	2-GCCW-S320	Component Cooling Water Elev. 610'-6" CCW Pump Area	A	NO	NO
54	2-GCCW-S519	Component Cooling Water Az 132° Elev. 623'-4" RC Pump Area	A	NO	NO
55	2-GCCW-S521	Component Cooling Water Az 132° Elev. 624'-9" RC Pump Area	A	NO	NO
56	2-GCCW-S550	Component Cooling Water Az 308° Elev. 619'-3" RC Pump Area	A	NO	NO

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
57	2-GCCW-S838	Component Cooling Water Elev. 621'-0" CCW Pump Area	A	NO	NO
58	2-GCCW-S839	Component Cooling Water Elev. 621'-0" CCW Pump Area	A	NO	NO
59	2-GCCW-S840	Component Cooling Water Elev. 620'-0" CCW Pump Area	A	NO	NO
60	2-GCCW-S843	Component Cooling Water Elev. 620'-5" CCW Pump Area	A	NO	NO
61	2-GCCW-S306	Component Cooling Water Elev. 596'-2 3/8" CCW Pump Area	A	NO	NO
62	2-GCS-S634	Chem&Vol. Control Az 299° Elev. 613'-1" in Annulus	A	NO	NO
63	2-GCS-S637	Chem&Vol. Control Az 72° Elev. 608'-10" in Annulus	I	NO	NO
64	2-GCS-S729	Chem&Vol. Control Az 234° Elev. 617'-0" RC Pump Area	I	NO	NO
65	2-MSS-1	Main Steam Az 8° Elev. 639'-1 1/4" between Stm. Gen. No. 1 and 4	I	NO	NO
66	2-MSS-2	Main steam Az 17° Elev. 635'-0" between Stm. Gen. No. 1 and 4	I	NO	NO

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICU TO REMOVE
67	2-MSS-3	Main steam Az 172° Elev. 639'-1 1/4" between Stm. Gen. No. 2 and 3	I	NO	NO
68	2-MSS-4	Main steam Az 165° Elev. 635'-0" between Stm. Gen. No. 2 and 3	I	NO	NO
69	2-MSS-5	Main steam Az 191° Elev. 635'-0" between Stm. Gen. No. 2 and 3	I	NO	NO
70	2-MSS-6	Main steam Az 184° Elev. 639'-1 1/4" between Stm. Gen. No. 2 and 3	I	NO	NO
71	2-MSS-7	Main steam Az 349° Elev. 635'-0" between Stm. Gen. No. 1 and 4	I	NO	NO
72	2-MSS-8	Main steam Az 356° Elev. 639'-1 1/4" between Stm. Gen. No. 1 and 4	I	NO	NO
73	2-GSI-S47	Safety Injection System Elev. 573'-0"	A	NO	YES
74	2-GSI-S51	Safety Injection System Elev. 573'-0"	A	NO	YES
75	2-GSI-S575	Safety Injection System Az 65° Elev. 598'-9 3/8" in Annulus	A	NO	NO

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
76	2-GSI-S657	Safety Injection System Az 185° Elev. 610'-0" in Annulus	A	NO	NO
77	2-GSI-S707	Safety Injection System Az 221° Elev. 608'-7" near RC Pump No.3	I	NO	NO
78	2-GCTS-S61	Containment Spray Elev. 579'-3" CTS Pump Area	A	NO	NO
79	2-GCTS-S113(E)	Containment Spray Elev. 582'-0" CTS Pump Area	A	NO	NO
80	2-GCTS-S113)W)	Containment Spray Elev. 582'-0" CTS Pump Area	A	NO	NO
81	2-GCTS-S114(N)	Containment Spray Elev. 582'-0" inside leak detector box pipe chase	A	NO	NO
82	2-GCTS-S114(S)	Containment Spray Elev. 582'-0" inside leak detector box pipe chase	A	NO	NO
83	2-GCTS-S115(N)	Containment Spray Elev. 579'-6" inside leak detector box pipe chase	A	NO	NO
84	2-GCTS-S115(S)	Containment Spray Elev. 579'-6" inside leak detector box pipe chase	A	NO	NO

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICU TO REMOVE
85	2-GCTS-S116(E)	Containment Spray Elev. 579'-6" inside leak detector box pipe chase	A	NO	NO
86	2-GCT-S116(W)	Containment Spray Elev. 579'-6" inside leak detector box pipe chase	A	NO	NO
87	N/A	Steam Generator No. 1 Elev. 665'	I	NO	YES
88	N/A	Steam Generator No. 1 Elev. 665'	I	NO	YES
89	N/A	Steam Generator No. 1 Elev. 665'	I	NO	YES
90	N/A	Steam Generator No. 1 Elev. 665'	I	NO	YES
91	N/A	Steam Generator No. 2 Elev. 665'	I	NO	YES
92	N/A	Steam Generator No. 2 Elev. 665'	I	NO	YES
93	N/A	Steam Generator No. 2 Elev. 665'	I	NO	YES
94	N/A	Steam Generator No. 2 Elev. 665'	I	NO	YES
95	N/A	Steam Generator No. 3 Elev. 665'	I	NO	YES

TABLE 3.7-9

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	HANGER MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
96	N/A	Steam Generator No. 3 Elev. 665'.	I	NO	YES
97	N/A	Steam Generator No. 3 Elev. 665'.	I	NO	YES
98	N/A	Steam Generator No. 3 Elev. 665'.	I	NO	YES
99	N/A	Steam Generator No. 4 Elev. 665'.	I	NO	YES
100	N/A	Steam Generator No. 4 Elev. 665'.	I	NO	YES
101	N/A	Steam Generator No. 4 Elev. 665'.	I	NO	YES
102	N/A	Steam Generator No. 4 Elev. 665'.	I	NO	YES

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-9 provided that a revision to Table 3.7-9 is included with the next License Amendment request.

Modifications to the "High Radiation Zone" column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-9 is included with the next License Amendment request.

3/4.7.8 SEALED SOURCE CONTAIMINATION

LIMITING CONDITION FOR OPERATION

3.7.8.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.8.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources and fission detectors previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient of operational cycles for those facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PNSRC and the NSDRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. **Records of the service lifes of hydraulic snubbers listed on Table 3.7-9 including the date at which service life commences and associated installation and maintenance records.**

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.7 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure of failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results required a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufactured input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.9 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinquish fires occuring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.10 PENETRATION FIRE BARRIERS

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, a continuous fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTRIC COMPANY

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

DOCKET NOS. 50-315 AND 50-316

INTRODUCTION

By letter dated November 11, 1981, the Indiana and Michigan Electric Company (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2. The proposed changes revise the surveillance requirements for safety-related hydraulic snubbers. The proposed changes are consistent with the sample TSs provided to the licensee by letter dated November 20, 1980, except for modifications made to the sample TSs to reflect the Cook Plant design (e.g. the lack of safety-related mechnical snubbers). By letter dated February 25, 1983, the licensee supplemented their application by revising the wording of section 4.0.5 to the Technical Specifications to include the Standard Technical Specification requirements for inservice inspection and testing. This change was necessary since the snubber Technical Specifications surveillance referenced the section 4.0.5.

BACKGROUND

To reflect accumulated experience obtained from operating plants in the past several years, NRC issued Revision 1 of the Standard Technical Specifications on the surveillance requirements for safety-related snubbers. On November 20, 1980, this document was transmitted to operating plants excluding those under the Systematic Evaluation Program (SEP) along with a request for submittal of appropriate license amendments to incorporate the requirements of this revision. The same request was extended to SEP plants on March 23, 1981.

DISCUSSION AND EVALUATION

The existing TS surveillance requirements for safety-related snubbers were generically incomplete and somewhat deficient in the following areas:

1. Mechanical snubbers were not included in the surveillance requirements.

2. The rated capacity of snubbers was not used as a limit to the inservice test requirement.

3. NRC approval was necessary for the acceptance of seal materials.

4. Inservice test requirements were not clearly defined.

5. In-place inservice testing was not permitted.

Since mechanical snubbers were not subject to any surveillance requirements, some licensees and permit holders believed that mechanical snubbers were preferred by NRC. Many plants used mechanical snubbers as original equipment and many others requested to replace their hydraulic snubbers with mechanical ones to simplify or avoid an inservice surveillance program. This is directly contradictory to NRC's intention, where for an unsurveyed mechanical snubber, the most likely failure is permanent lock-up. This failure mode can be harmful to the system during normal plant operations. This concern is not directly applicable to the Donald C. Cook Nuclear Plant as mechanical snubbers are not employed on safety-related systems.

When the first hydraulic snubber surveillance requirements in the Technical Specifications were drafted, the testing of snubbers was limited to those with rated capacity of not more than 50,000 lbs. This was because of the available capacity of the test equipment and the requirement to test some parameters at the snubber rated load. Since then, greater equipment capacity and better understanding of parametric correlation have been developed. Therefore, the limit of 50,000 lbs. has been removed.

Testing of snubbers was usually accomplished by removing snubbers from their installed positions, mounting them on a testing rig, conducting the test, removing them from rig, and reinstalling them to the working position. Many snubbers were damaged in the removing and reinstallation process. This defeated the purpose for conducting tests. Since methods and equipment have been developed to conduct in-place tests on snubbers, taking advantage of these developments could result in minimizing the damage to snubbers caused by removal and reinstallation plus time and cost savings to the plants.

From these shortcomings it was concluded that the snubber surveillance requirements for the Technical Specifications should be revised.

The revised surveillance requirements correct these deficiencies in the following manner:

1. Mechanical snubbers are now included in the surveillance program. (This item is not applicable to the Cook Plant.)

2. No arbitrary snubber capacity is used as a limit to the inservice test requirements.

3. Seal material no longer requires NRC approval. A monitoring program shall be implemented to assure that snubbers are functioning within their service life.

4. Clearly defined inservice test requirements for snubbers shall be implemented.

5. In-place inservice testing shall be permitted.

By letter dated February 25, 1983, the licensee supplemented the snubber Technical Specification request to include a section 4.0.5 in the Unit 1 TSs and to modify both Unit 1 and 2 TSs to remove requirements which are no longer applicable. The section 4.0.5 in Unit 1 on inservice inspection and testing is necessary to complete the requirements of the snubbers surveillance (both units reference requirements of section 4.0.5). The Unit 1 TSs section 4.0.4 was modified to delete a reference to the first inspection interval for certain fire protection surveillance requirements; this surveillance has begun for all the referenced sections and this exception to 4.0.4 is no longer valid. The Unit 2 TSs section 4.0.5.a.1 is removed since it dealt with inservice inspection and testing before plant operation; this provision is no longer applicable. These items are removed for clarification and their removal will have no effect on safe plant operation.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 50.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 13, 1983

Principal Reviewers:
H. Shaw
D. Wigginton

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-315 AND 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 71 to Facility Operating License No. DPR-58, and Amendment No. 53 to Facility Operating License No. DPR-74 issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

The amendment revise the Technical Specifications to reflect revised surveillance requirements for safety-related snubbers. These amendments also add a section 4.0.5 to the Unit 1 Technical Specifications on surveillance requirements for inservice inspection and testing of components and deletes a portion of section 4.0.4 in Unit 1 and 4.0.5.a.1 in Unit 2, which are no longer applicable. These latter changes are made to clarify the Technical Specifications and make them consistent with Standard Technical Specification requirements.

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

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

For further details with respect to this action, see (1) the application for amendments dated November 11, 1981, as supplemented February 25, 1983, (2) Amendment Nos. 71 and 53 to License Nos. DPR-58 and DPR-74, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Maude Reston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 13th day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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