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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. 37 to Facility Operating License No. DPR-58 and Amendment No. 20 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 May 9, 1980.

These amendments (1) provide for a one time only extension of the time for testing the Unit No. 2 ice condenser lower inlet doors and (2) revises the withdrawal schedule for the reactor vessel surveillance capsules for Unit Nos. 1 and 2.

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

Sincerely,

Original signed by:
S. A. Varga

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

Enclosures:
1. Amendment No. 37 to DPR-58
2. Amendment No. 20 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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SURNAME	SMiner:jb	CSParrish	SAVarga	TNovak		
DATE	05/ /80	05/ /80	05/ /80	05/ /80	05/ /80	

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

May 23, 1980

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:

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Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
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4. Notice of Issuance

cc: w/enclosures
See next page

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City of Bridgman, Michigan 49106

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Federal Activities Branch
Region V Office
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Chicago, Illinois 60604

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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. 37
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 May 9, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, 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. 37 , 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: May 23, 1980

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

Remove Pages	Insert Pages
B 3/4 4-11	B 3/4 4-11
3/4 4-29	3/4 4-29

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

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

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

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

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

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
Capsule T	1.25 EFPY
Capsule X	3 EFPY
Capsule Y	5 EFPY
Capsule U	9 EFPY
Capsule S	32 EFPY
Capsules V, W, Z	Standby

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray 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. 20
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 May 9, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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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. 20, 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: May 23, 1980

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A as follows:

Remove Pages	Insert Pages
B 3/4 4-10	B 3/4 4-10
3/4 4-27	3/4 4-27
3/4 6-39	3/4 6-39

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

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

*Estimate Based on USAEC Regulatory Standard Review Plan, Section 5.3.2 and MTEB 5-2.
(A) 60% Shear
(B) 70% Shear
NA - Not available or not applicable as appropriate.

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

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

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

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

3/4.4.10 STRUCTURAL INTEGRITY

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

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1. Capsule T	1 EFPY
2. Capsule Y	3 EFPY
3. Capsule X	5 EFPY
4. Capsule U	9 EFPY
5. Capsule S	32 EFPY
6. Capsules V, W, Z	Standby

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more ice condenser doors open or otherwise inoperable, POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained $\leq$ 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the inlet door position monitoring system, and

b. Demonstrated OPERABLE during shutdown at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 6 months# thereafter by:

   1. Verifying that the torque required to initially open each door is $\leq$ 675 inch pounds.

   2. Verifying that opening of each door is not impaired by ice, frost or debris.

   3. Testing a sample of at least 25% of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component. The doors selected for determination of the "door opening torque" shall be selected to ensure that all doors are tested at least once during four test intervals.

#For the lower inlet door inspection interval scheduled to end June 8, 1980, including the extension permitted by Specification 4.0.2, a one time only delay is allowed to extend this inspection interval through July 20, 1980.

SURVEILLANCE REQUIREMENTS (Continued)

4. Testing a sample of at least 25% of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component. The doors selected for determination of the "door closing torque" shall be selected to ensure that all doors are tested at least once during four test intervals.

5. Calculation of the frictional torque of each door tested in accordance with 3 and 4, above. The calculated frictional torque shall be ≤ 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

	Door	Lifting Force
1.	Adjacent to Crane Wall	≤ 37.4 lbs.
2.	Paired with Door Adjacent to Crane Wall	≤ 33.8 lbs.
3.	Adjacent to Containment Wall	≤ 31.8 lbs.
4.	Paired with Door Adjacent to Containment Wall	≤ 31.0 lbs.

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 20 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 May 9, 1980, Indiana and Michigan Electric Company submitted an application to amend the Technical Specifications appended to Facility Operating Licenses DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The requested change would (1) on a one time basis only, delay testing of the ice condenser inlet doors, about five weeks and (2) modify the withdrawal schedule of the reactor vessel material surveillance capsules.

Discussion

(1) Unit 2 Ice Condenser Lower Doors

The D. C. Cook Plant utilizes an ice condenser to limit the pressure buildup in the containment from a loss of coolant or steam line break accident. During such an accident, the lower doors in the ice condenser are opened by the differential pressure generated by the steam released and allows the air steam mixture to pass over the ice where the steam is condensed. To provide assurance that the doors will open when required and the doors remain in a sufficiently open position so that uniform distribution of the steam flow is achieved the Technical Specifications for D. C. Cook Unit 2 require that the licensee demonstrate every six months that the doors will open at about 67.5 inch pounds torque. In addition 25% of the doors are tested to verify that the torque required to open each door is less than 195 inch pounds when the door is 40 degrees open. The next surveillance interval including the grace period for Unit 2 will end on June 8, 1980. The licensee intends to bring Unit 2 off line to complete modifications to the Auxiliary Feedwater System late in June or early in July. In order to avoid two shutdowns within a short period of time the licensee requested that the surveillance period be extended to July 20, 1980.

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(2) Reactor Surveillance Capsules

Ten CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," requires a material surveillance program for reactor vessels to monitor changes in the fracture toughness properties of ferritic materials in the vessel beltline region resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens periodically withdrawn from the reactor vessel. This Appendix gives withdrawal schedules based on the amount of radiation damage predicted at the end of the service lifetime of the vessel. At the end of the service life, the D. C. Cook vessel materials are expected to have an increase in RT_{NDT} of less than 200°F. Based on this amount of radiation damage, Appendix H requires a four capsule surveillance program. The surveillance program for the D. C. Cook facility has a total of eight capsules for each unit and thus exceeds the Appendix H requirements.

Neutron irradiation causes the vessel material reference nil ductility temperature, RT_{NDT}, to increase with time and the material fracture toughness properties to decrease with time. These irradiated properties are used to establish pressure-temperature operating limit in accordance with Appendix G, 10 CFR Part 50.

Evaluation

(1) Unit 2 Ice Condenser Lower Doors

The licensee has requested that the surveillance period for testing the lower doors of Unit 2 ice condenser for the next test be extended to July 20, 1980, a delay of about five weeks. The licensee has surveillance tested the doors on four previous occasions and, except for tests conducted in November 1978 when one of the doors required a greater opening force than allowed by the Technical Specifications, all doors were found acceptable. Subsequently, all doors have been tested twice and have been found acceptable. In addition, the licensee has demonstrated in the FSAR that the failure of a single door will not prevent the ice condenser from performing its safety function. Since the testing to date has demonstrated satisfactory performance of the lower doors and since the possibility of a LOCA or steam line break occurring during the five week extension is remote, we find that extending the surveillance period from June 8, 1980 to July 20, 1980 on a one time only basis is acceptable.

(2) Reactor Surveillance Capsules

The proposed withdrawal schedule of the material surveillance capsules was recommended by Westinghouse and Southwest Research Institute, I&MEC service inspection consultants. In the present Technical Specifications, four capsules are required to be removed during service life and three are in standby. In the proposed schedule five capsules are to be withdrawn during service. Appendix H requires a four capsule program for D. C. Cook 1, one of which is a standby capsule. Thus, the proposed program exceeds Appendix H requirements. The time periods for withdrawal of capsules is in accordance with Appendix H requirements. We find the proposed program is acceptable.

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 §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of 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 accidents previously considered and do not involve a significant decrease in a safety margin, 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 23, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-315 AND 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 37 to Facility Operating License No. DPR-58, and Amendment No. 20 to Facility Operating License No. DPR-74 issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specifications for operation of Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

The amendments (1) provide for a one time only extension of the time for testing the Unit No. 2 ice condenser lower inlet doors and (2) revises the withdrawal schedule for the reactor vessel surveillance capsules for Unit Nos. 1 and 2.

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

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

For further details with respect to this action, see (1) the application for amendments dated May 9, 1980, (2) Amendment Nos. 37 and 20 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 23rd day of May, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

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