November 14, 1988

Docket Nos. 50-315
and 50-316

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

Dear Mr. Alexich:

SUBJECT: AMENDMENTS NOS. 118 AND 104 TO FACILITY OPERATING LICENSES NOS. DPR-58 AND DPR-74: UNIT 2 CYCLE 7 INCREASED ENRICHMENT (TACS NOS. 69062 and 69063)

The Commission has issued the enclosed Amendment No. 118 to Facility Operating License No. DPR-58 and Amendment No. 104 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the licenses and to the Technical Specifications in response to your application dated August 19, 1988, as supplemented by your letter dated September 30, 1988.

These amendments change the Technical Specifications and License Conditions 2.C.(5) for Unit 1 and 2.C.(3)(s) for Unit 2 by increasing the maximum enrichment of the fuel assemblies from the present values to 4.23 weight percent U-235. The changes are necessary because the Unit 2 Cycle 7 reload includes fuel assemblies which have enrichment in excess of the 3.84 weight percent presently allowed by the Technical Specifications. In addition, the amendment for Unit 2 makes an editorial change to Technical Specification 5.6.1.2.

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

original signed by

Wayne Scott, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 118 to DPR-58
2. Amendment No. 104 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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DREP

LA/PD31:DRSP	PM/PD31:DRSP	PD31:DRSP	(A)D/PD31:DRSP	OGC	SPLB:DEST	PRPB:DEST
RIngram	WScott	JStang	TQuay		JCraig	LCunningham
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 14, 1988

Dockets Nos. 50-315
and 50-316

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

Dear Mr. Alexich:

SUBJECT: AMENDMENTS NOS. 118 AND 104 TO FACILITY OPERATING LICENSES NOS. DPR-58 AND DPR-74: UNIT 2 CYCLE 7 INCREASED ENRICHMENT (TACS NOS. 69062 and 69063)

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-58 and Amendment No. to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the licenses and to the Technical Specifications in response to your application dated August 19, 1988, as supplemented by your letter dated September 30, 1988.

These amendments change the Technical Specifications and License Conditions 2.C.(5) for Unit 1 and 2.C.(3)(s) for Unit 2 by increasing the maximum enrichment of the fuel assemblies from the present values to 4.23 weight percent U-235. The changes are necessary because the Unit 2 Cycle 7 reload includes fuel assemblies which have enrichment in excess of the 3.84 weight percent presently allowed by the Technical Specifications. In addition, the amendment for Unit 2 makes an editorial change to Technical Specification 5.6.1.2.

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Wayne Scott

Wayne Scott, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 118 to DPR-58
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3. Safety Evaluation
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cc w/enclosures:
See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

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

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
Post Office 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
2300 N Street, N.W.
Washington, DC 20037

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

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

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3500 N. Logan Street
Post Office Box 30035
Lansing, Michigan 48909

Mr. S. Brewer
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 118
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated August 10, 1988, 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 paragraphs 2.C.(2) and 2.C.(5) of Facility Operating License No. DPR-58 are hereby amended to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 118 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.C.(5) Spent Fuel Pool Storage

   The licensee is authorized to store D. C. Cook, Unit 1 and Unit 2 fuel assemblies, new or irradiated, in any combination up to a total of 2050 fuel assemblies in the shared spent fuel pool at the Donald C. Cook Nuclear Plant subject to the following conditions:

   Fuel stored in the spent fuel pool shall not have an enrichment greater than 4.23% Uranium-235 or a fissile fuel density greater than 50.5 grams of Uranium-235 per axial centimeter of fuel assembly or the reactivity equivalent thereof.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore Quay

Theodore Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 14, 1988

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated August 19, 1988, 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 paragraphs 2.C.(2) and 2.C.(3)(s) of Facility Operating License No. DPR-74 are hereby amended to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 104 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.C.(3)(s) Spent Fuel Pool Storage

   The licensee is authorized to store D. C. Cook, Unit 1 and Unit 2 fuel assemblies, new or irradiated, in any combination up to a total of 2050 fuel assemblies in the shared spent fuel pool at the Donald C. Cook Nuclear Plant subject to the following conditions:

   Fuel stored in the spent fuel pool shall not have an enrichment greater than 4.23% Uranium-235 or a fissile fuel density greater than 50.5 grams of Uranium-235 per axial centimeter of fuel assembly or the reactivity equivalent thereof.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore Quay

Theodore Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1988

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1: The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 10.5 inch center-to-center distance betwen fuel assemblies placed in the storage racks.

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

Fuel Type	Description	Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
I	Westinghouse 15 x 15	3.50
II	Exxon 15 x 15	3.50
III	Westinghouse 17 x 17	3.50
IV	Exxon 17 x 17	4.23
V	Westinghouse OFA 15 x 15	4.00

CRITICALITY - NEW FUEL

5.6.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when Fuel Types I, II, III, IV & V (as defined in Section 5.6.1.2) are placed in the pit and aqueous foam moderation is assumed.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.3 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.23 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 ± 100 cubic feet as a nominal T_{avg} of 70°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water.

b. A nominal 10.5 inch center-to-center distance between fuel assemblies, placed in the storage racks.

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

Fuel Type	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % ^{235}U
I	Westinghouse	15 x 15	3.50
II	Exxon	15 x 15	3.50
III	Westinghouse	17 x 17	3.50
IV	Exxon	17 x 17	4.23
V	Westinghouse OFA	15 x 15	4.00

CRITICALITY - NEW FUEL

5.6.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that K_{eff} will not exceed 0.98 when Fuel Types I, II, III, IV & V (as defined in Section 5.6.1.2) are placed in the pit and aqueous foam moderation is assumed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

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

DOCKETS NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated August 19, 1988, the Indiana Michigan Power Company (the licensee) requested amendments to the licenses and to the Technical Specifications (TSs) appended to Facility Operating Licenses Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Units Nos. 1 and 2. By letter dated September 30, 1988, the licensee provided additional information. The proposed amendments would change the TSs and License Conditions 2.C.(5) for Unit 1 and 2.C.(3)(s) for Unit 2 by increasing the maximum enrichment of the fuel assemblies from present values to 4.23 weight percent U-235. The changes are necessary because the Unit 2 Cycle 7 reload includes fuel assemblies which have enrichments in excess of the 3.84 weight percent U-235 presently allowed by the licenses and the TSs. In addition, the proposed amendments would make an editorial change to Unit 2 TS 5.6.1.2. The term "OFA" is added to Fuel Type V to clarify that Westinghouse Type V fuel is of the optimized fuel assembly design.

2.0 EVALUATION

The analyses presented by the licensee as justification for the proposed changes were performed by the Advanced Nuclear Fuels (ANF) Corporation, the supplier of the Unit 2 Cycle 7 fuel. The criticality analysis was performed to 5.00 weight percent U-235 enrichment. However, the range of the criticals used in the benchmarking of the codes extends only to 4.31 weight percent U-235. Therefore, the NRC staff considers the validity of the analyses limited to a maximum of 4.31 weight percent U-235 enrichment, which is higher than 4.23, the value to which this request has been limited in the additional information provided in the licensee's letter dated September 30, 1988, and within the range of the data base. New fuel assembly enrichment is considered for the spent fuel pool because the new fuel assemblies are stored temporarily in the spent fuel pool racks prior to loading in the reactor.

2.1 Methodology and Criticality Codes

The computer codes and cross sections used in these evaluations are part of the SCALE system of codes which has been extensively benchmarked. For example, the KENO-Va is a Monte Carlo-based code generally used by the industry for spent fuel criticality calculations. Cross sections were prepared using the BONAMI/NITAWL codes, and k_{inf} configurations were computed using the deterministic code XSDRNPM. Parametric studies were performed with the CASMO code, which has also been extensively benchmarked. The above methods and codes are acceptable for spent fuel pool and new fuel storage vault criticality calculations.

2.2 New Fuel Storage Vault Criticality

KENO-Va calculations demonstrated that (a) the most reactive configuration of a flooded new fuel storage vault is that with pure water and (b) the k_{eff} of such configuration for a 5.00 weight percent U-235 enrichment is 0.9437. Therefore, the k_{eff} for the 4.23 weight percent U-235 enrichment of this request would be much lower and thus below the Standard Review Plan (NUREG-0800) acceptance criteria maximums of 0.95 for the dry configuration and 0.98 for a flooded vault with optimum moderation.

The NRC staff has reviewed the licensee's methodology and results and finds them acceptable for the new fuel storage vault criticality analysis.

2.3 Spent Fuel Pool Criticality

KENO-Va and CASMO-3 calculations were performed for the most reactive fuel assemblies to be stored in the spent fuel pool; i.e., the ANF 17x17 configuration. For a maximum U-235 enrichment of 4.23 weight percent and to a 95% confidence level, the nonborated flooded pool will have a k_{eff} less than 0.95. This configuration is unrestricted. The k_{eff} value is based on the design of the racks and the enrichment and design of the fuel. Lower k_{eff} values could result if a zoned fuel loading pattern were calculated; however, at this time, no restrictions have been requested.

Reactivity increase can be postulated under accident conditions such as a dropped fuel assembly or other handling accident. However, the reactivity reduction due to the required pool boration of 2,400 ppm more than offsets the potential reactivity increases from postulated fuel mishandling accidents.

The NRC staff has reviewed the licensee's methodology and results and finds them acceptable for an unrestricted loading pattern in the spent fuel pool.

2.4 Spent Fuel Pool Thermal Hydraulics

The present spent fuel pool thermal-hydraulic analysis of record was performed by ANF in support of reracking the spent fuel pool. The analyses were submitted by the licensee in letter AEP:NRC:116 dated January 22, 1979, and approved by the NRC via Amendments 32 (Unit 1) and 13 (Unit 2) dated October 16, 1979. Those analyses considered assemblies with average exposure up to 40,000 MWD/MTU. The new assemblies, with their higher enrichments, are capable of an average exposure up to 50,000 MWD/MTU and, therefore, could have a higher decay heat generation rate. The new ANF analyses were performed in a fashion similar to the former analyses.

To determine the decay heat rate, ANF used the methodology described in Branch Technical Position ASB 9-2, entitled "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." Thermal-hydraulic calculations were performed using ANF's XCOBRA-IIIC code, which had been reviewed and previously approved by the NRC staff as indicated above. The analysis consisted of several portions, involving determination of the maximum heat rate per assembly, the maximum heat input to the pool, natural circulation cooling of the fuel in the pool under normal and accident conditions, and the pool water heatup rate following loss of pool cooling.

The pool water heatup rate following loss of pool cooling was determined to be 9.54°F/hr. The calculation assumed a complete failure of both trains of the spent fuel pool cooling system and completely filled racks, including a full core offload within a week after shutdown. Assuming an initial pool temperature of 130°F, it would take 8.6 hours until the pool would begin bulk boiling. (This represents only a minor change over the previous analysis, which predicted a heatup rate of 8.2°F/hr.) The analysis determined that fuel element temperatures would be within acceptable limits for all credible spent fuel pool conditions. The highest calculated peak clad temperature was 236°F, for the case of complete loss of spent fuel pool cooling. This temperature is much lower than normal operating clad temperature of approximately 650°F and is approximately an order of magnitude lower than the 10 CFR 50.46 peak clad temperature limit of 2200°F.

The NRC staff has reviewed the licensee's methodology and results and finds them acceptable for the spent fuel pool thermal hydraulic analysis.

2.5 Design Basis Accident Analysis Relative To Extend Fuel Burnup

The licensee has requested authorization to increase fuel enrichment to 4.23 weight percent of U-235 and to allow fuel burnup up to 50,000 MWD/MTU. The NRC staff and licensee evaluated the potential impact of this change on the radiological assessment of design basis accidents (DBA) which were previously analyzed in the licensing of the D. C. Cook Nuclear Plant, Unit 2.

The NRC staff reviewed the licensee's submittals and also reviewed a publication which was prepared for the NRC entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors, "NUREG/CR 5009, February 1988. The NRC contractor, the Pacific Northwest Laboratory (PNL) of Battelle Memorial Institute, examined the changes in the NRC DBA assumptions, described in the various appropriate Standard Review Plan (SRP) sections and/or Regulatory Guides, that could result from the use of extended burnup fuel (up to 60,000 MWD/MTU). The staff agrees that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. PNL estimates that I-131 fuel gap activity in the peak fuel rod with 60,000 MWD/MTU burnup could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors).

The NRC staff, therefore, reevaluated the fuel handling accidents for the D. C. Cook Unit 2 facility with an increase in iodine gap activity in the fuel damaged in a fuel handling accident. Table 1 presents the fuel handling accident thyroid doses presented in the licensee's Updated FSAR and the increased thyroid doses (by 20%) resulting from extended burnup fuel.

Table 1

Thyroid Doses as a Consequences of DBA Fuel Handling Accidents

Exclusion Area

Thyroid Dose (Rem)

Fuel Handling Accident	A*	B**
In Fuel Building	4.62	5.54
In Reactor Building	84.7	101.64

* A Updated FSAR dose
** B Extended fuel burnup dose

The NRC staff concludes that the only potential increased doses potentially resulting from DBA with extended fuel burnup to 50,000 MWD/MT is the thyroid dose resulting from fuel handling accidents and these doses remain well within the 300 Rem thyroid exposure guideline values set forth in 10 CFR Part 100 and that this small calculated increase is not significant.

2.6 Editorial Changes

In addition to the changes discussed above, the licensee also proposed an editorial change to Unit 2 TS 5.6.1.2. The term "OFA" is added to Fuel Type V to clarify that Westinghouse Type V fuel is of the optimized fuel assembly design, as opposed to Westinghouse Type I fuel, which is of the standard 3 Westinghouse design. This change makes the Units 1 and 2 TSs consistent on this point. The change is purely editorial and thus would not be expected to involve a significant increase in the probability or consequences of a previously analyzed accident, create the possibility of a new or different kind of accident from any previously analyzed or evaluated, or involve a significant reduction in a margin of safety. Based on the above evaluation, the staff finds the editorial change acceptable.

3.0 SUMMARY

The licensee's analysis provided acceptable justification that fuel enrichment of ANF 17x17 assemblies to 4.23 weight percent U-235 has k_{eff} values within the required limits and thus is acceptable. In the spent fuel pool, fuel assemblies with this enrichment can be loaded in an arbitrary pattern. The analysis provided is acceptable for a maximum enrichment of 4.31 weight percent U-235. Based on the above evaluation, the NRC staff finds the licensee's analysis justifying the increase in enrichment of the fuel assemblies to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on November 14, 1988. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner and (2) 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.

Date: November 14, 1988

Principal Contributors: L. Lois and J. Stang

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKETS NOS. 50-315 AND 50-316

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The United States Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 118 and 104 to Facility Operating Licenses Nos. DPR-58 and DPR-74, issued to the Indiana Michigan Power Company (the licensee), which revised the licenses and Technical Specifications (TSs) for operation of the Donald C. Cook Nuclear Plant, Units Nos. 1 and 2, located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

These amendments change the TSs and License Conditions 2.C.(5) for Unit 1 and 2.C.(3)(s) for Unit 2 by increasing the maximum enrichment of the fuel assemblies from the present values to 4.23 weight percent U-235. The changes are necessary because the Unit 2 Cycle 7 reload includes fuel assemblies which have enrichment in excess of the 3.84 weight percent presently allowed by the TSs. In addition, the amendment for Unit 2 makes an editorial change to TS 5.6.1.2.

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.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on October 11, 1988 (53 FR 39679). No request for hearing or petition to intervene was filed following this notice.

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on November 14, 1988, at 53 FR 45830 .

For further details with respect to this action, see (1) the application for amendments dated August 19, 1988, (2) Amendments Nos. and to Licenses 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, Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the Maude Preston Palenski 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, DC 20555, Attention: Director, Division of Reactor Projects - III, IV, V, and Special Projects.

Dated at Rockville, Maryland, this 14th day of November , 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Wayne Scott

Wayne Scott, Project Manager
Project Directorate III-I
Division of Reactor Projects - III,
IV, V & Special Projects