May 17, 1990

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: AMENDMENT NOS. 136 AND 121 TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74: (TAC NOS. 75798 AND 75799)

The Commission has issued the enclosed Amendment No. 136 to Facility Operating License No. DPR-58 and Amendment No. 121 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated December 8, 1989, and supplemented by letter dated March 6, 1990.

These amendments modify Technical Specifications (TS) so that Westinghouse fuel assemblies with enrichments of up to 4.95 weight percent U-235 may be received. A new Technical Specification (TS 3/4.9.15) is added for both units to require a minimum boron concentration in the fuel storage pool whenever fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool. TS 5.6.1.2, 5.6.2 and 5.3.1 (for Unit 2 only) are modified to reflect the increased allowable fuel enrichment. In addition, the license for both units is modified to reflect a maximum enrichment of 4.95 weight percent U-235 for fuel stored in spent fuel pool.

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

Sincerely,

Original signed by

Joseph Giitter, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 136 to DPR-58
2. Amendment No. 121 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRSP	PM/PD31:DRSP	NRR/PRPB	(A)D/PD31:DRSP	OGC
PShuttleworth	JGiitter	JCunningham	DDiIanni	Sturk as marked
5/18/90	5/7/90	5/9/90	05/07/90	5/15/90

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

May 17, 1990

Docket 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: AMENDMENT NOS. 136AND 121TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74: (TAC NOS. 75798 AND 75799)

The Commission has issued the enclosed Amendment No. 136 to Facility Operating License No. DPR-58 and Amendment No. 121 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated December 8, 1989, and supplemented by letter dated March 6, 1990.

These amendments modify Technical Specifications (TS) so that Westinghouse fuel assemblies with enrichments of up to 4.95 weight percent U-235 may be received. A new Technical Specification (TS 3/4.9.15) is added for both units to require a minimum boron concentration in the fuel storage pool whenever fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool. TS 5.6.1.2, 5.6.2 and 5.3.1 (for Unit 2 only) are modified to reflect the increased allowable fuel enrichment. In addition, the license for both units is modified to reflect a maximum enrichment of 4.95 weight percent U-235 for fuel stored in spent fuel pool.

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

Sincerely,

for

Joseph Giitter, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 136to DPR-58
2. Amendment No. 121to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:
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Mayor, City of Bridgman
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Special Assistant to the Governor
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Lansing, Michigan 48909

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

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. 136
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 December 8, 1989 and supplemented on March 6, 1990, 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, Facility Operating License No. DPR 58 is hereby amended by changing paragraph 2.C(4) paragraph 2 to read as follows:*

   "Fuel stored in the spent fuel pool shall not have an enrichment greater than 4.95% Uranium-235."

3. 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 136 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 17, 1990

*Page 4 is attached, for convenience, for the composite license to reflect this change.

Amendment No. 31

2.C(4) The licensee may proceed with and is required to complete the modifications identified in Table 1 of the Fire Protection Safety Evaluation Report for the Donald C. Cook Nuclear Plant dated June 4, 1979. These modifications shall be completed in accordance with the dates contained in Table 1 of that SER or Supplements thereto. Administrative controls for fire protection as described in the licensee's submittals dated January 31, 1977 and October 27, 1977 shall be implemented and maintained.

(5) Spent Fuel Pool Storage

Amendment No. 118,136

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 4.95% Uranium-235.

(6) Deleted by Amendment 80.

*2.D Physical Protection

Amendment No. 122

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Donald C. Cook Nuclear Plant Security Plan," with revisions submitted through July 21, 1988; "Donald C. Cook Nuclear Plant Training and Qualification Plan," with revisions submitted through December 18, 1986; and "Donald C. Cook Nuclear Plant Safeguards Contingency Plan," with revisions submitted through June 10, 1988. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

ATTACHMENT TO LICENSE AMENDMENT NO. 136

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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

REMOVE	INSERT
*	3/4 9-19
B 3/4 9-4	B 3/4 9-4
5-5	5-5
*	5-5a
5-6	5-6
*	5-6a

STORAGE POOL BORON CONCENTRATION*

LIMITING CONDITION FOR OPERATION

3.9.15 A boron concentration of greater than or equal to to 2,400 ppm shall be maintained in the fuel storage pool.

APPLICABILITY: Whenever fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool.

ACTION:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies in the fuel storage pool and restore the boron concentration to within its limit prior to resuming fuel movement. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.15 The boron concentration in the fuel storage pool shall be determined to be at least at its minimum required at least once per 7 days when fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool.

*Shared system with Cook Nuclear Plant - Unit 2

3/4.9.13 SPENT FUEL CASK MOVEMENT

The limitations of this specification ensure that, during insertion or removal of spent fuel casks from the spent fuel pool, fuel cask movement will be contrained to the path and lift height assumed in the Cask Drop Protection System safety analysis. Restricting the spent fuel cask movement within these requirements provides protection for the spent fuel pool and stored fuel from the effects of a fuel cask drop accident.

3/4.9.14 SPENT FUEL CASK DROP PROTECTION SYSTEM

The limitations on the use of spent fuel casks weighing in excess of 110 tons (nominal) provide assurance that the spent fuel pool would not be damaged by a dropped fuel cask since this weight is consistent with the assumptions used in the safety analysis for the performance of the Cask Drop Protection System.

3/4.9.15 STORAGE POOL BORON CONCENTRATION

The limitation on the fuel storage pool boron concentration of 2,400 ppm provides assurance that the K_{eff} would be below 0.95 in the unlikely event of fuel misloading.

DESIGN FEATURES

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 $\pm$ 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 between fuel assemblies placed in the storage racks.

c. 1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel with nominal enrichment above 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU. In Region 1, fuel shall be stored in a three-out-of-four cell configuration with one symmetric cell location of each 2 x 2 cell array vacant.

2. The boundary between the Region 1 mentioned above and the rest of the spent fuel storage racks (defined as Region 2) shall be such that the three-out-of-four storage requirement shall be carried into Region 2 by, at least, one row as shown in Figure 5.6-1.

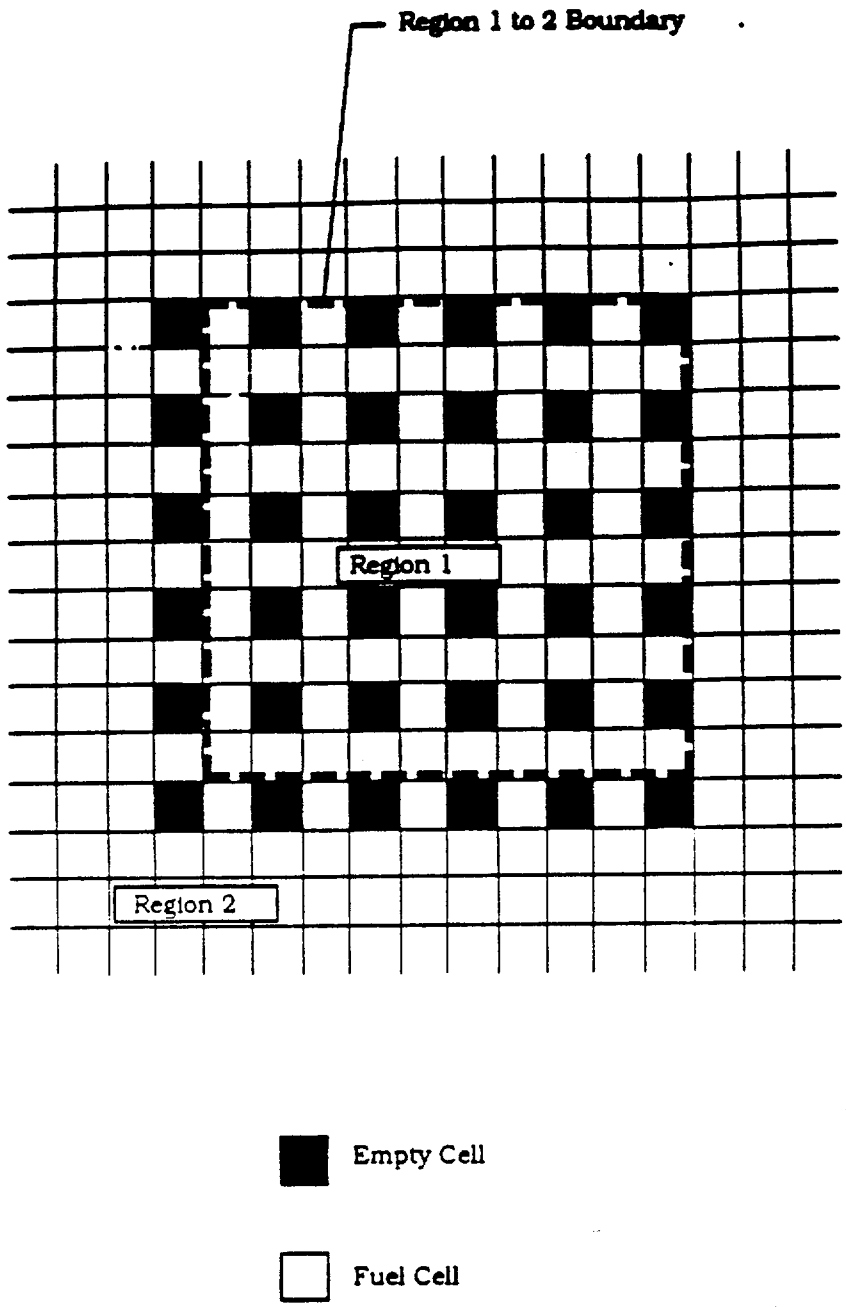

Figure 5.6-1: Donald C. Cook Nuclear Plant Schematic for Fuel Storage Racks Interface Boundary Between Regions 1 and 2

DESIGN FEATURES

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

Description		Maximum Nominal Fuel Assembly Enrichment Wt. % ^{235}U
1) Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2) Exxon/ANF	15 x 15	3.50
3) Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4) Exxon/ANF	17 x 17	4.23

CRITICALITY-NEW FUEL

5.6.2.1 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 assemblies are placed in the pit and aqueous foam moderation is assumed.

5.6.2.2 Fuel stored in the new fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows;

Description		Maximum Nominal Fuel Assembly Enrichment Wt. % ^{235}U
1) Westinghouse	15 x 15 STD 15 x 15 OFA	4.55
2) Exxon/ANF	15 x 15	3.50
3) Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.55
4) Exxon/ANF	17 x 17	4.23

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

DESIGN FEATURES

CAPACITY

5.6.4 **The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2050 fuel assemblies.**

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in the FSAR shall be designed and maintained to the original design provisions contained in the FSAR with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower shall be located as shown in Figure 5.1-1.

5.9 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

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. 121
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 December 8, 1989 and supplemented on March 6, 1990, 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, Facility Operating License No. DPR-74 is hereby amended by deleting 2.C.(3)(p) and change 2.C.(3)(s) paragraph 2 to read as follows:*

   "Fuel stored in the spent fuel pool shall not have an enrichment greater than 4.95% Uranium-235."

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 121 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 17, 1990

*Pages 5 and 6 are attached, for convenience, for the composite license to reflect this change.

(l) Deleted by Amendment 63.

(m) Deleted by Amendment 19.

(n) Deleted by Amendment 28.

(o) Fire Protection

Amendment No. 12

The licensee may proceed with and is required to complete the modifications identified in Table 1 of the Fire Protection Safety Evaluation Report for the Donald C. Cook Nuclear Plant dated June 4, 1979. These modifications shall be completed in accordance with the dates contained in Table 1 of that SER or Supplements thereto. Administrative controls for fire protection as described in the licensee's submittals dated January 31, 1977 and October 27, 1977 shall be implemented and maintained.

Amendment No. 64, 121

(p) Deleted by Amendment

(q) Deleted by Amendment 2.

(r) Deleted by Amendment 68.

(s) Spent Fuel Pool Storage

Amendment No. 104,121

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 4.95% Uranium-235.

*Amendment 3 deleted Paragraph (s), Amendment 13 added a new Paragraph (s).

(t) Deleted by Amendment 63.

2.C.(7) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall be described in the station chemistry manual and shall include:

Amendment No. 18

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to measure the values of the critical parameters;
3. Identification of process sampling points;
4. Procedure for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry conditions; and
6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

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

REMOVE	INSERT
*	3/4 9-18
B 3/4 9-4	B 3/4 9-4
5-4	5-4
5-5	5-5
*	5-5a
5-6	5-6

STORAGE POOL BORON CONCENTRATION*

LIMITING CONDITION FOR OPERATION

3.9.15 A boron concentration of greater than or equal to to 2,400 ppm shall be maintained in the fuel storage pool.

APPLICABILITY: Whenever fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool.

ACTION:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies in the fuel storage pool and resotre the boron concentration to within its limit prior to resuming fuel movement. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.15 The boron concentration in the fuel storage pool shall be determined to be at least at its minimum required at least once per 7 days when fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool.

*Shared system with Cook Nuclear Plant - Unit 1

BASES

Should the doors become blocked or stuck open while under administrative control, Technical Specification requirements will not be considered to be violated provided the Action Statement requirements of Specification 3.9.12 are expeditiously followed, i.e., movement of fuel within the storage pool or crane operation with loads over the pool is expeditiously suspended.

3/4.9.13 SPENT FUEL CASK MOVEMENT

The limitations of this specification ensure that, during insertion or removal of spent fuel casks from the spent fuel pool, fuel cask movement will be constrained to the path and lift height assumed in the Cask Drop Protection System safety analysis. Restricting the spent fuel cask movement within these requirements provides protection for the spent fuel pool and stored fuel from the effects of a fuel cask drop accident.

3/4.9.14 SPENT FUEL CASK DROP PROTECTION SYSTEM

The limitations on the use of spent fuel casks weighing in excess of 110 tons (nominal) provide assurance that the spent fuel pool would not be damaged by a dropped fuel cask since this weight is consistent with the assumptions used in the safety analysis for the performance of the Cask Drop Protection System.

3/4.9.15 STORAGE POOL BORON CONCENTRATION

The limitation on the fuel storage pool boron concentration of 2,400 ppm provides assurance that the K_{eff} would be below 0.95 in the unlikely event of fuel misloading.

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 may be enriched up to 4.95 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 $70^{o}F$.

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.

c. 1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel with nominal enrichment above 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU. In Region 1, fuel shall be stored in a three-out-of-four cell configuration with one symmetric cell location of each 2 x 2 cell array vacant.

2. The boundary between the Region 1 mentioned above and the rest of the spent fuel storage racks (defined as Region 2) shall be such that the three-out-of-four storage requirement shall be carried into Region 2 by, at least, one row as shown in Figure 5.6-1.

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

Description		Maximum Nominal Fuel Assembly Enrichment Wt. % 235_U
1) Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2) Exxon/ANF	15 x 15	3.50
3) Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4) Exxon/ANF	17 x 17	4.23

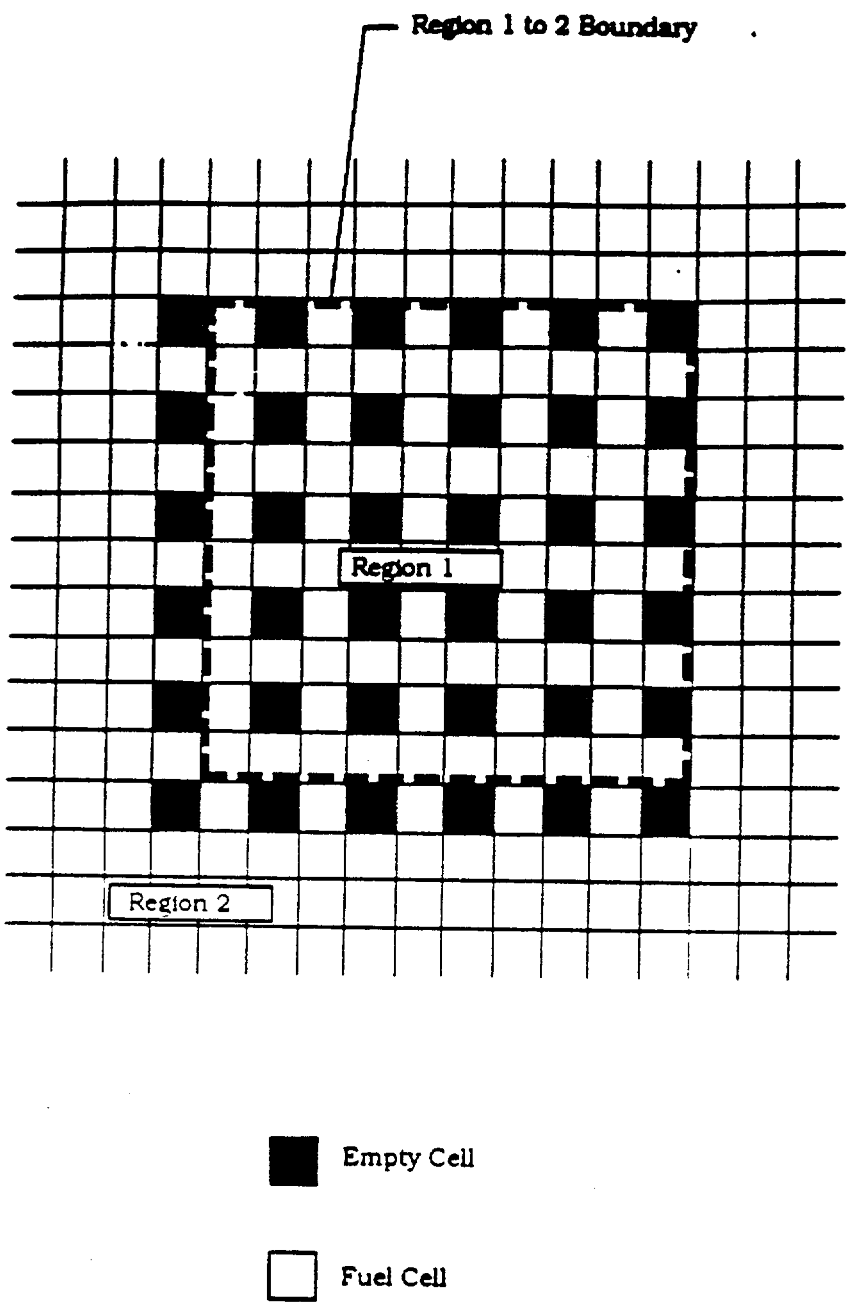

Figure 5.6-1: Donald C. Cook Nuclear Plant Schematic for Fuel Storage Racks Interface Boundary Between Regions 1 and 2

CRITICALITY-NEW FUEL

5.6.2.1 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 assemblies are placed in the pit and aqueous foam moderation is assumed.

5.6.2.2 Fuel stored in the new fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows;

Description		Maximum Nominal Fuel Assembly Enrichment Wt. % ^{235}U
1) Westinghouse	15 x 15 STD 15 x 15 OFA	4.55
2) Exxon/ANF	15 x 15	3.50
3) Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.55
4) Exxon/ANF	17 x 17	4.23

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2050 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

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

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letters dated December 8, 1989 (Ref. 1) and March 6, 1990 (Ref. 12), Indiana and Michigan Power Company (IMPC) (the licensee) requested a change to the Donald C. Cook Nuclear Plant (DCCNP), Facility Operating Licenses DPR-58 and DPR-74, which would (1) add a new Technical Specification (TS) 3/4.9.15 and corresponding basis for both units, (2) revise TS 5.3.1 for Unit 2 only, (3) revise TS 5.6.1.1 and 5.6.1.2 for both units, (4) revise TS 5.6.2 for both units, and (5) revise a license condition for both units. The proposed changes would permit the storage at DCCNP of fuel assemblies with enrichments up to 4.95 weight percent (w/o) uranium-235. The subsequent loading of this fuel in the DCCNP reactors will be based on plant and cycle specific safety analyses that would ensure that all applicable criteria and TS are met. The increase in fuel enrichment is needed to support extended burnup fuel cycles for the two DCCNP units.

Our review of the criticality aspects of these proposed changes is described below.

2.0 EVALUATION

2.1 Analytical Methods

The analytical methods used in the criticality analysis of the DCCNP fuel storage racks use the AMPX (Ref. 2) system codes for neutron cross section generation and the KENO-IV (Ref. 3) Monte Carlo computer code for reactivity determination. The 227 energy group neutron cross section library used in the AMPX system of codes is based on ENDF/B-V (Ref. 4). For processing the 227 group neutron cross section library to obtain multigroup cross sections for evaluating criticality experiments and the DCCNP fuel storage racks, the NITAWL (Ref. 2) code is used to provide the self-shielded resonance cross sections. NITAWL uses the Nordheim Integral Treatment for the resonances. The XSDRNPM (Ref. 2) one-dimensional S code is used to perform the energy and spatial weighting of cross sections. The multigroup neutron cross sections generated for a particular configuration are then input to the KENO-IV (Ref. 3) Monte Carlo code to evaluate the criticality of the critical experiments and DCCNP fuel storage racks.

These methods were benchmarked by Westinghouse, the vendor performing the analysis of the DCCNP fuel storage racks, by analyzing 33 critical experiments. These experiments covered water moderated, uranium-oxide fuel arrays separated by various materials that simulate light water reactor (LWR) fuel shipping and storage conditions (Ref. 5) to critical experiments using highly enriched uranium metal cylindrical arrays with various interspersed materials (Ref. 6). The results of the analysis of these critical experiments are: (1) the average calculated effective multiplication factor (k_{eff}) of the critical experiments is 0.992 (Ref. 7), (2) the standard deviation of the bias value is 0.0008 delta-k, and (3) the 95 percent probability with a 95 percent confidence level (95/95 probability/confidence level) uncertainty in reactivity of the analytical methods is 0.0018 delta-k.

2.2 Spent Fuel Storage Racks

The DCCNP spent fuel storage racks have a nominal 10.5 inch center-to-center spacing. Boral is sandwiched between stainless steel to form each wall of a storage cell. The Boral contains the neutron absorber boron-10 at a loading of 0.02 grams per square centimeter. The spent fuel storage racks can provide for the storage of 2050 fuel assemblies. The DCCNP spent fuel storage racks are currently licensed to store fuel assemblies which do not exceed an enrichment of 4.23 w/o uranium-235 for 17x17 EXXON/ANF 17x17 fuel assemblies and 3.50 w/o uranium-235 EXXON/ANF 15x15 fuel assemblies. The present submittal addresses the following spent fuel storage issues: (1) the maximum enrichment fuel assembly that can be stored in Region 1 with a loading pattern using three out of four storage locations (that is, one empty storage location in every 2x2 array of Region 1 storage locations), and (2) the maximum enrichment fuel assembly that can be stored in Region 2 using every storage location and with credit for fuel burnup.

2.2.1 Criticality Analysis of Region 1 Spent Fuel Storage Racks

Analyses were performed by Westinghouse to determine if a fuel assembly having the maximum proposed enrichment of 4.95 w/o uranium-235 could be stored in the Region 1 spent fuel storage racks using three out of four storage locations. The fuel assembly is assumed to be at its most reactive point in life, and no credit is taken for any burnable absorber in the fuel rods or any natural uranium axial blankets. The analyses were performed for 17x17 Westinghouse OFA fuel assemblies which give a larger effective neutron multiplication factor, k_{eff}, than does either the 17x17 Westinghouse STD fuel assemblies or the 15x15 Westinghouse STD or OFA fuel assemblies when all of the assemblies have the same fuel enrichment. Other assumptions that were made for the analyses are:

1. All fuel rods contain uranium dioxide fuel with the same uranium-235 enrichment over the entire length of the fuel rod. The fuel pellets are assumed to be at 96% of theoretical density, and no credit is taken for dishing or chamfering of the pellets. Westinghouse states that the 4.95 w/o uranium-235 enrichment can be taken as an enrichment limit because the conservatisms in the fuel rod and pellet parameters bound the standard 0.05 w/o enrichment tolerance.

2. No credit is taken for the uranium-234 or uranium-236 in the fuel or for burnup.

3. The spent fuel pool water is pure water at a temperature of 68°F with a conservative value of 1 gram per cubic centimeter for the water density.

4. No credit is taken for spacer grids or sleeves.

5. Fuel assemblies are loaded in every three out of four locations throughout Region 1 (that is, there is one empty storage location in every 2x2 array of storage locations in Region 1).

6. The Region 1 fuel storage racks are assumed to be infinite in extent in the lateral dimensions. The axial dimension is taken to be finite.

7. A minimum poison material loading of 0.02 grams per square centimeter is used throughout the array.

8. No credit is assumed for the soluble boron in the spent fuel pool water.

Westinghouse performed both a nominal analysis and a worst case analysis of the Region 1 spent fuel storage racks. For the nominal analysis, the KENO calculation gave a k_{eff} of 0.9005 with a 95 percent probability/95 percent confidence level uncertainty of 0.0065. For the worst case analysis, Westinghouse considered the storage racks with the minimum center-to-center spacings and with symmetrically placed fuel assemblies. This minimum center-to-center spacing is based on material tolerances and construction tolerances. This worst case model has been shown by Westinghouse to yield a conservative spent fuel storage rack k_{eff}. Based on this worst case analysis, Westinghouse determined that an enrichment of 4.95 w/o uranium-235 results in a k_{eff} of 0.9308 for the Region 1 spent fuel storage racks loaded in a three out of four configuration. Biases are added to this worst case k_{eff} to account for the biases in the calculational method and boron-10 poison particle self-shielding. These biases add 0.0097 delta-k to the worst case k_{eff}. In addition, uncertainties at the 95/95 probability/confidence level are added to k_{eff} to account for the uncertainty in the method bias and in the KENO-IV Monte Carlo uncertainty on the worst case k_{eff}. These uncertainties add 0.0049 delta-k to the worst case k_{eff}. The corrected k_{eff}, including biases and uncertainties, is 0.9454 for this worst case analysis.

We conclude that the storage of fresh Westinghouse fuel assemblies (15x15 STD and OFA, 17x17 STD, OFA and V5) enriched to 4.95 w/o in uranium-235 is acceptable for the Region 1 spent fuel storage racks loaded in a three out of four configuration because the k_{eff}, including biases and uncertainties, is less than the staff criterion of 0.95 and because suitably conservative analysis assumptions have been made.

Westinghouse also considered accidents that could increase the reactivity of the spent fuel storage racks. For these accident conditions, the staff position is that credit can be taken for the soluble boron in the spent fuel pool water. Westinghouse states that 2000 ppm of boron in the pool water will decrease reactivity by about 0.25 delta-k. This large decrease in reactivity will more than compensate any reactivity increase for conceivable accidents. Finally, the DCCNP will have a Technical Specification that will require the boron concentration in the spent fuel pool to be maintained at least at a concentration of 2400 ppm. We conclude that the accident evaluation is acceptable because k_{eff} will be less than the staff criterion of 0.95 when credit is taken for 2400 ppm of boron in the spent fuel pool water.

2.2.2 Criticality Analysis of Region 2 Spent Fuel Storage Racks

Analyses were performed by Westinghouse to determine the allowed fuel assembly initial enrichment as a function of the fuel assembly's burnup for storage in the Region 2 spent fuel storage racks. The PHOENIX code (Ref. 8) was used to determine the amount of fuel burnup required for a given fuel assembly initial enrichment using the method of reactivity equivalency. Using this method, a curve is generated that is based on a constant rack reactivity for Region 2. The starting point of the curve is a fuel assembly having an initial enrichment of 3.95 w/o uranium-235 and zero burnup. The end of point of the curve is a fuel assembly having an initial enrichment of 4.95 w/o uranium-235 and a burnup of 5,500 MWd/MTU. The reactivity results calculated with the PHOENIX code are subsequently normalized to a KENO-IV calculation for fuel with an enrichment of 3.95 w/o uranium-235 and with zero burnup.

The KENO-IV analysis of the Region 2 spent fuel storage racks used the same assumptions that were used in the analysis of the Region 1 spent fuel storage racks, with two exceptions. The first exception is that the fuel assembly enrichment is 3.95 w/o uranium-235 for the Region 2 rack analysis. The other exception is that every location in the Region 2 racks is allowed for fuel assembly storage. However, the boundary between the Region 1 and Region 2 racks is defined such that the three out of four storage requirement extends into Region 2 by one row.

Westinghouse performed both a nominal analysis and a worst case analysis of the Region 2 spent fuel storage racks. For the nominal analysis, the KENO calculation gave a k_{eff} of 0.9141 with a 95/95 probability/confidence level uncertainty of 0.0049. For the worst case analysis, Westinghouse made the same assumptions that were used for the Region 1 worst case analysis. Based on this worst case analysis, Westinghouse determined that an initial enrichment of 3.95 w/o uranium-235 for a fuel assembly having a zero burnup results in a k_{eff} of 0.9327 for the Region 2 spent fuel storage racks. Biases add 0.0097 delta-k to the worst case k_{eff}. Uncertainties at the 95/95 probability/confidence level add 0.0048 delta-k to the worst case k_{eff}. The corrected k_{eff}, including biases and uncertainties, is 0.9472 for this worst case analysis.

We conclude that the storage of Westinghouse fuel assemblies (15x15 STD and OFA, 17x17 STD, OFA, and V5) that have initial uranium-235 enrichments between 3.95 w/o and 4.95 w/o and discharge burnups in excess of 5,550 MWd/MTU is acceptable for the Region 2 spent fuel storage racks because the k_{eff}, including biases and uncertainties, is less than the staff criterion of 0.95 and because suitably conservative analysis assumptions have been made.

The same considerations apply to postulated accidents as was discussed in the previous section on the Region 1 rack analysis. The DCCNP will have a Technical Specification that will require the boron concentration in the spent fuel pool to be maintained at a concentration of at least 2400 ppm. This boron concentration will more than compensate for any reactivity increase caused by any accident.

2.3 New Fuel Storage Racks

The licensee has provided an analysis of the criticality of the new fuel storage racks in the DCCNP new fuel storage vault (NFSV) for both the fully flooded and low density hydrogenous moderation conditions. The analyses used the same calculational methods and neutron cross section libraries that were used in the analysis of the spent fuel storage racks. The NFSV analyses are based on the assumptions noted below.

1. The fuel assembly contains the highest enrichment authorized, is at its most reactive point in life, and no credit is taken for any burnable poison in the fuel rods or for any natural uranium axial blankets.

2. All fuel rods contain uranium dioxide fuel with the same uranium-235 enrichment over the entire length of the fuel rod. The fuel pellets are assumed to be at 96% of theoretical density, and no credit is taken for dishing or chamfering of the pellets. Westinghouse states that the 4.55 w/o uranium-235 enrichment can be taken as a nominal enrichment limit because the conservatisms in the pellet modelling bound the standard 0.05 w/o enrichment tolerance.

3. No credit is taken for any uranium-234 or uranium-236 in the fuel, nor is any credit taken for the buildup of fission products.

4. No credit is taken for any spacer grids or spacer sleeves.

For the fully flooded condition, the water is taken to be at 68°F and at a density of 1.0 grams per cubic centimeter. The analysis was performed for Westinghouse OFA fuel assemblies in an array that is infinite in both the lateral and axial dimensions. Westinghouse considered the most conservative, or worst case, KENO-IV model of the new fuel storage racks, that is, racks with no structural steel and with symmetrically placed fuel assemblies. Based on this worst case analysis, Westinghouse determined that an enrichment of 4.55 w/o uranium-235 results in a k_{eff} of 0.9324. A bias of 0.0083 delta-k is added to this worst case k_{eff} to account for the bias in the calculational methods. In addition, uncertainties at the 95/95 probability/confidence level are added to k_{eff} to account for the uncertainty in the method and in the Monte Carlo uncertainty on the worst case k_{eff}. These uncertainties add 0.0088 delta-k to the worst case k_{eff}. The corrected k_{eff}, including biases and uncertainties is 0.9495. This k_{eff} is below the staff criterion of 0.95 for the fully flooded NFSV.

A similar analysis was performed for the case of low density hydrogenous moderation. For this case, it was determined that a Westinghouse STD fuel assembly was more reactive than an OFA fuel assembly. Westinghouse determined that the maximum k_{eff} of 0.8817 occurred at a water density of 0.045 grams per cubic centimeter. A bias of 0.0083 delta-k and an uncertainty at a 95/95 probability/confidence level of 0.0074 is added to this k_{eff}. The calculated k_{eff} is 0.8974, which is well below the staff's criterion of 0.98.

We, therefore, conclude that the storage of new Westinghouse STD, OFA, and V5 fuel assemblies with a uranium-235 enrichment up to 4.55 w/o in the DCCNP NFSV is acceptable because such storage will meet the staff criterion on k_{eff} of less than or equal to 0.95 for the fully flooded case and on k_{eff} of less than or equal to 0.98 for the low density hydrogenous moderation case.

2.4 Design Basis Fuel Handling Accident

IMPC has requested authorization to store fresh fuel assemblies with enrichment of up to 4.95 weight percent U-235 in the spent fuel pool. The burnup of the fuel assemblies at discharge would not exceed 56,000 MWd/MTU. The staff has evaluated the potential impact of this change on the radiological assessment of design basis accidents (DBAs) which were previously analyzed in the licensing of D. C. Cook Units 1 and 2.

The 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 on Light Water Reactors, "NUREG/CR 5009, February 1988. The NRC contractor, the Pacific Northwest Laboratory (PNL) of Battelle Memorial Institute, examined potential changes in the NRC design basis accident assumptions, as described in appropriate sections of the staff's Standard Review Plan 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.

NRC Regulatory Guide 1.25, "Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors," stipulates that 10% of the fuel assembly inventory of noble gases and iodine (with the exception of krypton-85) would be available for release from the fuel-cladding gap. A release fraction of 0.3 was stipulated for krypton-85 due to its relatively long half-life of 10.7 years. The release fractions for extended burnup fuel (60,000 MWd/MTU) referenced in NUREG/CR 5009 are all lower than those assumed in Regulatory Guide 1.25, with the exception of I-131. PNL estimates that the I-131 gap release fraction for the peak fuel rod with 60,000 MWd/MTU burnup could be as high as 0.12.

The staff reevaluated the potential offsite doses associated with the design basis fuel handling accident as documented in the D. C. Cook Safety Evaluation Report (SER) dated September 10, 1973. The release fractions used in the SER analysis are consistent with Regulatory Guide 1.25. The fuel handling accident thyroid doses presented in the SER and the increased thyroid doses resulting from fuel initially enriched to 5.292 weight percent U-235, with burnup to 60,000 MWd/MTU are shown below. The SER doses were conservatively increased by 20% to obtain a revised dose estimate for the extended burnup case. The revised estimate is reasonable because I-131 is the dominant contributor to thyroid dose. As shown in the Table, the resulting doses are small fractions of the applicable regulatory requirements of 10 CFR Part 100.

CALCULATED DOSES DUE TO DESIGN BASIS FUEL HANDLING ACCIDENT

	2-Hour Exclusion Boundary Thyroid Dose (REM)*
September 10, 1973 SER estimate	11
Current NRC Staff estimate	13
Current FSAR analysis**	2
10 CFR Part 100	300

* Low Population Zone doses are less than Exclusion Area Boundary doses

** Based on I-131 release fraction of .05.

The staff concludes (1) that the bounding doses potentially increased are the thyroid doses at the Exclusion Area and Low Population Zone boundaries resulting from postulated fuel handling accidents, (2) that these doses remain well within the 300 rem thyroid exposure guideline values set forth in 10 CFR Part 100, (3) that the small incremental increase in doses is not significant and (4) that the Technical Specification change requested by the licensee is acceptable.

2.5 Technical Specification

The proposed changes to the DCCNP license conditions and Technical Specifications have been reviewed. We found the following license conditions and specifications to be acceptable.

1. License condition 2.C.(5) for Unit 1 and 2.C.(3)(s) for Unit 2

   These two license conditions are being revised to change the limitation on the enrichment of fuel that can be stored in the spent fuel pool to 4.95 w/o uranium-235. These changes are acceptable because they are supported by the safety analysis for the spent fuel pool.

2. License condition 2.C.3.(p) for Unit 2

   This license condition is being deleted. This is acceptable because it is consistent with the safety analysis.

3. Technical Specification 3/4.9.15 for Units 1 and 2

   This new specification requires that the boron concentration of the spent fuel pool shall be equal to or greater than 2400 ppm. This specification is applicable when fuel assemblies with an initial enrichment of greater than 3.95 w/o uranium-235 and a burnup of less than 5,550 MWd.MTU are in the spent fuel storage racks. The surveillance requirement states that the boron concentration shall be determined to be at least at its minimum value at least once per 7 days when the conditions of the applicability statement are met. This proposed Technical Specification is acceptable because it is consistent with the safety analysis for the spent fuel pool.

4. Technical Specification 5.3.1 for Unit 2

   This specification is being changed to permit the use of fuel enriched up to 4.95 w/o uranium-235. This change is acceptable because it is consistent with the safety analysis for the spent fuel pool.

5. Technical Specification 5.6.1.1 for Units 1 and 2

   This specification includes two new subparagraphs, c.1 and c.2. Subparagraph c.1 defines the criteria for the storage of fuel assemblies in the Region 1 spent fuel storage racks. Subparagraph c.2 defines the criteria for the storage of fuel assemblies on the boundary between the Region 1 and Region 2 spent fuel storage racks. These proposed additions to the specification are acceptable because they are consistent with the safety analysis.

6. Technical Specification 5.6.1.2 for Units 1 and 2

   The maximum nominal enrichment of all types of Westinghouse fuel assemblies is being changed to 4.95 w/o uranium-235. This change is acceptable because it is consistent with the safety analysis.

7. Technical Specification 5.6.2 for Units 1 and 2

   This specification was revised to reflect the nominal fuel assembly enrichments that can be stored in the new fuel storage racks. The revised specification is acceptable because it is consistent with applicable safety analyses.

Based on the review described above, we conclude that the proposed Technical Specification and License Condition modifications for the DCCNP are acceptable for the storage of Westinghouse fuel assemblies in the new and spent fuel storage racks. These modifications will allow the storage of Westinghouse fuel assemblies with initial enrichments of up to 4.95 w/o uranium-235 in the spent fuel storage racks and Westinghouse fuel assemblies with initial enrichments of up to 4.55 w/o uranium-235 in the new fuel storage racks.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change in a surveillance requirement. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding.

Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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