Docket Nos. 50-315
and 50-316

January 14, 1993

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

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - AMENDMENT NOS. 169 AND 152 TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74 (TAC NOS. M80615 AND M80616)

The Commission has issued the enclosed Amendment No. 169 to Facility Operating License No. DPR-58 and Amendment No. 152 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 (TS) in response to your application dated July 26, 1991. By letters dated June 7, 1991, February 4, April 1, and October 26, 1992, the licensee provided additional information concerning the application.

The proposed amendments would change the License and the TS to permit expansion of the spent fuel pool (SFP) storage capacity from 2050 assemblies to 3613 assemblies. The expansion will be accomplished by replacing the existing SFP storage racks with higher density storage racks and by placing racks in areas of the pool where currently no racks exist.

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

William M. Dean, Sr. Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 169 to DPR-58
2. Amendment No. 152 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICIAL RECORD COPY FILENAME: C080615.AMD

OFFICE	LA:PD31	PM:PD31	OGC	D:PD31	
NAME	MShuttleworth	WDean:vsb		LMarsh	
DATE	12/15/92	12/14/92	1/6/93	1/13/93	/ /

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

January 14, 1993

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William M. Dean

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Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 169 to DPR-58
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3. Safety Evaluation

cc w/enclosures:
See next page

Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

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Mr. S. Brewer
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Corporation
1 Riverside Plaza
Columbus, Ohio 43216

DATED: January 14, 1992

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK
AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK

Docket File
NRC & Local PDRs
PDIII-1 Reading
D.C. Cook Plant File
J. Roe
T. King
L. Marsh
M. Shuttleworth
W. Dean
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (8), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
W. Shafer, R-III

cc: Plant Service list

120009

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. 169
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 July 26, 1991 as supplemented June 7, 1991, February 4, April 1, and October 26, 1992, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 169, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, the license is amended by changes to paragraph 2.C.(5) and adding paragraph 2.C.(8) to read as follows:*

   (5) Spent Fuel Pool Storage

   The licensee is authorized to store D. C. Cook, Unit 1 and Unit 2 fuel assemblies, new or irradiated, up to a total of 3613 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 a nominal enrichment greater than 4.95% Uranium-235.

   (8) The provisions of Specification 3/4.9.7 are not applicable for loads being moved over the pool for the duration of the spent fuel pool reracking project. Control of loads moving over the spent fuel pool during the spent fuel pool reracking project shall comply with the criteria of NUREG-0612, "Controls of Heavy Loads at Nuclear Power Plants." Administrative controls shall be in place to prevent any load not rigged in compliance with the criteria of NUREG-0612 from passing over the spent fuel pool with the crane interlocks, required by T/S 3/4.9.7, disengaged.

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

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachments:
1. Changes to the Technical Specifications
2. Page 4 and 4a replaces 3a and 4 of License*

Date of Issuance: January 14, 1993

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

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, up to a total of 3613 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 a nominal enrichment greater than 4.95% Uranium-235.

Admt. No. 118, 11/14/88
Admt. No. 169, 1/14/93

(6) Amendment No. 32, 10-16-79

Deleted - Amendment No. 80, 04-27-84

(7) Secondary Water Chemistry Monitoring Program

The licensee shall inplement 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:

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."

Admt. No. 36, 2/29/80

(8) The provisions of Specification 3/4.9.7 are not applicable for loads being moved over the pool for the duration of the spent fuel pool reracking project. Control of loads moving over the spent fuel pool during the spent fuel pool reracking project shall comply with the criteria of NUREG-0612, "Controls of Heavy Loads at Nuclear Power Plants." Administrative controls shall be in place to prevent any load not rigged in compliance with the criteria of NUREG-0612 from passing over the spent fuel pool with the crane interlocks, required by T/S 3/4.9.7, disengaged.

Admt. No. 169, 1/14/93

2.D. Physical Protection

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 19, 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.

Admt. No. 122, 3/15/89

ATTACHMENT TO LICENSE AMENDMENT NO. 169

TO 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 vertical lines indicating the area of change.

REMOVE	INSERT
3/4 9-3	3/4 9-3
3/4 9-19	3/4 9-19
5-4	5-4
5-5	5-5
5-6	5-6
5-7	5-7
----	5-7a
---	5-7b
5-8	5-8
5-9	5-9

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 168 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 168 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 168 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

STORAGE POOL BORON CONCENTRATION*

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

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.

*Shared system with Cook Nuclear Plant - Unit 2

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 5.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 204 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 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 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:

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 ± 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 8.97 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 152 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 letter dated July 26, 1991 as supplemented June 7, 1991, February 4, April 1, and October 26, 1992, the Indiana Michigan Power Company (IMPC or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2 (D. C. Cook). The proposed amendments are intended to permit expansion of the spent fuel pool (SFP) storage capacity from 2050 assemblies to 3613 assemblies. The licensee intends to accomplish the proposed expansion by replacing the existing racks with higher storage density racks and by placing racks in areas of the pool where racks are not currently located.

The present D. C. Cook SFP storage racks have a total storage capacity of 2050 cells, 1403 of which are presently occupied. Since the full core for each unit has 193 fuel assemblies, maintaining full core off-load capability from one reactor implies that 1857 storage cells (2050 minus 193) are available for normal off-load storage. Consideration of previous and future fuel assembly discharges indicates that D. C. Cook will lose full core discharge capability (for one reactor) in 1995. Therefore, to preclude this situation and to ensure that sufficient spent fuel storage capacity continues to exist at D. C. Cook, IMPC plans to install high density spent fuel storage racks whose design incorporates Boral as a neutron absorber in the cell walls, thereby allowing for more dense storage of spent fuel. The new racks would provide an ultimate storage capacity of 3613 fuel assemblies and extend the date of loss of full core discharge capability through the year 2008.

2.0 EVALUATION

2.1 Criticality Aspects

Three separate storage regions are provided in the SFP with independent criteria defining the highest potential reactivity in each of the three regions. Region I is designed to accommodate new fuel with a maximum enrichment of 4.95 (nominal) + 0.05 weight percent (w/o) U-235 or spent fuel regardless of its discharge burnup. These cells are located alternately along the rack periphery (where neutron leakage reduces reactivity) or along the boundary between two storage modules (where the water gap provides a flux-trap which reduces reactivity). An alternative configuration consists of loading internal storage cells in a checkerboard pattern of fresh fuel (or fuel of any burnup) with empty cells. This configuration is intended primarily to facilitate a full core unload when needed, prior to the time the racks are beginning to fill up. Region 2 is designed to accommodate fuel of maximum nominal initial enrichment of 4.95 w/o which has been irradiated to at least 50,000 MWD/MTU (assembly average). Region 3 is designed to accommodate fuel of maximum nominal initial enrichment of 4.95 w/o which has been irradiated to at least 38,000 MWD/MTU (assembly average).

The analysis of the reactivity effects of fuel storage in Region 1, 2, and 3 was performed with the KENO-5a Monte Carlo computer code using the 27-group SCALE cross-section library. Since the KENO-5a code package does not have burnup capability, depletion analyses and the determination of equivalent enrichments were made with the two-dimensional transport theory code, CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Cook spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, these two independent methods of analysis (KENO-5a and CASMO-3) showed very good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a calculations, a minimum of 500,000 neutron histories in 1000 generations of 500 neutrons each was accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-5a reactivity calculations. The staff concludes that the analysis methods used are capable of predicting the reactivity of the Cook storage racks with a high degree of confidence and are, therefore, acceptable.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Unborated pool water at the temperature yielding the highest reactivity (68°F).

(2) Assumption of infinite radial array of storage cells except for the boundary storage cells where leakage is inherent.

(3) Neutron absorption effect of structural material is neglected.

The staff concludes that appropriately conservative assumptions were made.

For the nominal storage cell design, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, fuel enrichment and density tolerances, and eccentric fuel positioning and minimum water-gap tolerance were accounted for. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations and the effect of the axial distribution in burnup. The final maximum calculated reactivity resulted in a k_{eff} of 0.940 when combined with all known uncertainties. This meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

As previously mentioned, the KENO-5a package does not have burnup capability. Therefore, irradiated fuel was represented by fuel of equivalent enrichment as determined by CASMO-3 calculations in the storage cell (i.e., an enrichment which yields the same reactivity in the storage cell as the irradiated fuel does). Figure 4-6 in the licensee's July 26, 1991, application shows the equivalent enrichment for fuel of 4.95 w/o initial enrichment at various discharge burnups, evaluated in the storage cell. From this Figure, it can be seen that fuel attaining 55,000 MWD/KgU burnup which had an initial enrichment of 4.95 w/o U-235 is equivalent to fresh fuel (zero burnup) having an initial enrichment of approximately 1.20 w/o U-235. This reactivity equivalencing method is the standard one used for storage rack reactivity evaluations and is acceptable.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the misloading of an assembly with a burnup and enrichment combination outside of the acceptable area in Figure 4-3 of the licensee's July 26, 1991, application or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of approximately 2400 ppm of boron in the pool water required at all times by the plant TS since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents.

2.2 Control of Heavy Loads

The D. C. Cook SFP currently contains 20 medium density rack modules with a total of 2050 storage cells. During the period from June 1993 to July 1994, the time period proposed for the reracking operation, approximately 1678 out of 2050 cell locations will be occupied with spent fuel. The licensee determined that a sufficient number of unoccupied cells will be present in the pool to permit relocation of all fuel such that the existing rack modules can be emptied and removed from the pool, and new rack modules installed in a programmed manner. The rack modules will not be anchored to the pool floor.

In the Safety Evaluation issued pursuant to Amendment 100 to Facility Operating License DPR-74 for D. C. Cook, Unit 2, the staff found that the design of both the modified original and supplementary auxiliary building cranes at D. C. Cook comply with the criteria for single-failure-proof cranes presented in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979. In the licensing report prepared for the licensee by Holtec International, the licensee specifically committed to employ in the reracking process a lifting rig designed to meet the criteria of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980. The licensee also stated that operator training, crane inspection, safe load path development, and procedure development for the reracking operation will comply with the criteria of NUREG-0612.

The modified original and the newer supplementary auxiliary building cranes have maximum critical load capacities of 55 and 60 tons, respectively. The licensee stated that the cranes will not be used to lift more that 50 percent of their capacity. In addition, the licensee committed to conduct an inspection of the crane to be used in the reracking process within three months of the beginning of reracking operations.

The licensee has developed preliminary safe load paths to ensure rack modules will not be carried over any region of the pool containing fuel. The licensee also committed to administratively impose restrictions on the handling of all racks in the process of being transported. Racks undergoing transfer in or out of the pool will be empty. In addition, the licensee committed to establish a training program in order to train crew members involved in the reracking process.

The licensee also evaluated the consequences of postulated load drop and crane uplift scenarios. The limiting load drop scenario was determined to be a rack module loaded with two rows of fuel dropped from the maximum allowable height of six inches. Certain peripheral rack modules are planned to be partially loaded with fuel assemblies approximately 12 to 18 inches from their final location. These rack modules will then be lifted less than six inches above the pool floor and moved to their final locations. The licensee determined by analysis that the structural integrity of the SFP, rack module, and fuel assembly will remain unimpaired following this postulated accidental load drop. The consequences of a single fuel assembly dropped from a height of 36 inches above the top of the rack was also evaluated. The load transmitted to the SFP liner was found to be bounded by that load considered for seismic concerns. Analysis of a local crane uplift load of 3000 lb indicates that induced rack stresses are bounded by other evaluated scenarios. Since the rack modules are not anchored, no damage to the SFP liner is expected. The D. C. Cook Updated Final Safety Analysis Report (UFSAR) limits the maximum load applied to a stuck fuel assembly to 2950 lbs.

The staff has previously determined that the design of the cranes to be used in the reracking operation comply with the criteria for single-failure-proof cranes presented in NUREG-0554. The licensee has committed to employ in the reracking process a single-failure-proof handling system designed to meet the criteria of Section 5.1.6 of NUREG-0612. Single-failure-proof cranes and associated lifting devices which conform to the criteria of NUREG-0554 and NUREG-0612 satisfy the guidance of Regulatory Guide 1.13 and Section 9.1.5 of the Standard Review Plan (SRP), NUREG-0800, and the requirements of General Design Criteria 4 and 61 of Appendix A to 10 CFR Part 50 with regard to the design of heavy load handling systems. Therefore, the staff finds the licensee has committed to employ an acceptable heavy load handling system in the reracking process.

The requirements of TS 3.9.7 related to the movement of loads over the spent fuel storage pool cannot be met due to the weight of the fuel storage racks. The licensee has requested that the provisions of TS 3.9.7 be non-applicable for loads being moved over the spent storage pool during the entire reracking project. The licensee has committed to complying with the criteria of NUREG-0612 when moving loads over the spent fuel storage pool and putting into place administrative controls to prevent any loads not rigged in compliance with NUREG-0612 from passing over the spent fuel storage pool with the crane interlocks, required by TS 3.9.7, disengaged. The licensee has also committed to develop operator training programs, crane inspection plans, safe load paths, and procedures for the reracking operation which comply with the criteria of Section 5.1.1 of NUREG-0612. In addition, the licensee plans to impose other administrative restrictions on the handling of the rack modules. These plans and commitments are consistent with the defense-in-depth approach of NUREG-0612 and the guidance of Section 9.1.5 of the SRP and are acceptable.

Although the provision of a single-failure-proof handling system substantially reduces the probability of a heavy load drop event, the licensee evaluated several postulated load drop events. These load drops were found by analysis not to impair the structural integrity of the SFP. This analysis provides additional assurance that an uncontrolled decrease in pool cooling water inventory would not result from postulated load drops. The staff determined that the licensee has taken measures consistent with the guidance of Section 9.1.2 of the SRP to demonstrate postulated crane uplift loads would not result in a decrease in SFP inventory.

2.3 Spent Fuel Pool Thermal-Hydraulics

The licensing report states that the decay heat load calculation for the SFP was performed in accordance with the provisions of Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Rev. 2, July 1981. In order to evaluate the total decay heat load, an inventory of 3438 fuel assemblies accumulated through scheduled discharges from December 1976 to July 2008 was assumed to be present in the SFP. In addition to the heat load from the inventory described above, additional heat loads resulting from scenarios involving normal discharges of 80 fuel assemblies to the SFP and full-core off-loads of 193 fuel assemblies to the SFP were evaluated. The heat load from the inventory of 3438 fuel assemblies was assumed to be constant for all discharge scenarios. A period of approximately 3.5 years of full power operation was assumed for all stored fuel. The normal discharge and the full-core off-load to the SFP were assumed to occur at a rate of four fuel assemblies per hour following 168 hours of decay in the reactor vessel.

A transient calculation was performed to evaluate bulk pool temperature. Convective heat transfer and evaporative cooling from the pool surface, and heat removal through operating SFP heat exchangers were credited in the analysis. The heat removal rate through operating SFP heat exchangers was calculated based on a temperature effectiveness factor obtained by rating the heat exchanger on a proprietary thermal-hydraulic computer code. In obtaining the temperature effectiveness value, the heat exchanger was assumed to be fouled to the design maximum extent. The most limiting design basis scenario with regard to bulk pool temperature was found to be a normal discharge of 80 fuel assemblies in July 2009 coincident with a single failure of one of the two SFP cooling trains. The calculated maximum bulk pool temperature for this scenario was evaluated to be 159.5°F at a time 207 hours following reactor shutdown. The full-core off-load of 193 fuel assemblies, assumed to occur 30 days after the normal refueling off-load of 80 fuel assemblies from the opposite unit in July 2009, was found to result in a maximum bulk pool temperature of 143.8°F with both trains of SFP cooling in operation.

The UFSAR specifies a minimum design temperature of 200°F for SFP support system components. In addition, the licensee states that all elements of the SFP purification system, including the ion exchange resins, are qualified for a 200°F design temperature. Long-term exposure to temperatures above 150°F may damage the reinforced concrete structure of the SFP. However, the licensee expects the maximum concrete bulk temperature to remain below 150°F at all times due to the temperature lag caused by the transient nature of the heat up and the continuously decreasing decay heat load of the SFP inventory.

The licensing report also evaluated the transient response of the SFP following a loss of all forced cooling. The loss of cooling was assumed to occur coincident with the maximum bulk temperature reached for each scenario evaluated. The response was evaluated assuming no make up water addition. Of the design basis scenarios evaluated, the full-core off-load of 193 fuel assemblies, assumed to occur 30 days after the normal refueling off-load of 80 fuel assemblies from the opposite unit in July 2009, was found to be most limiting. Both SFP cooling trains were assumed to be operating initially. For this scenario, bulk boiling conditions were determined to exist in the SFP 5.74 hours following the loss of forced cooling.

In the safety evaluation issued pursuant to Amendment No. 32 to Facility Operating License No. DPR-58 and Amendment No. 13 to Facility Operating License No. DPR-78 for D. C. Cook, Units 1 and 2, respectively, the staff accepted the chemical and volume control system hold-up tanks as the Seismic Category I source of make up water to the SFP. The hold-up tank recirculation pump, which is rated at 500 gpm, can be used to pump water from the hold-up tanks to the SFP. The licensee has identified a number of alternate sources of make up water and established a procedure to provide make up water from these various sources to the SFP in the event of a loss of forced cooling. In Section 9.1.2, "Spent Fuel Storage," of the September 11, 1973, Safety Evaluation for the Donald C. Cook Nuclear Plant, the staff also found that the spent fuel storage facility meets the design criteria of Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

In order to verify cladding integrity is not threatened, a model was developed to calculate the maximum local cladding temperature. The model was used to determine the location of minimum flow in an idealized, axially symmetric arrangement of fuel assemblies. The calculation assumed that the fuel assembly located in the minimum flow region is the most thermally limited. As an additional conservatism, the fuel assembly cladding was assumed to have a crud deposit which covered the entire surface. For both unblocked and 50 percent blocked flow conditions, the calculation indicated no incidence of nucleate boiling and no potential for fuel cladding damage.

The staff has previously determined that the design of the D. C. Cook spent fuel storage facility meets the design criteria of Regulatory Guide 1.13 and that the make up water supply is acceptable. Section 9.1.3 of the SRP provides guidance in evaluating the heat load imposed on the SFP cooling system. The guidance specifies evaluation of the following two scenarios: (1) a single active failure assumed coincident with a SFP inventory consisting of one refueling off-load after 150 hours decay, one refueling off-load after one year decay, and one refueling off-load after 400 days decay; and (2) a full-core off-load after 150 hours decay, one refueling off-load after 36 days decay, and one refueling off-load after 400 days decay with no assumed equipment failures. The staff compared these scenarios to the licensee's design basis scenarios and found the licensee's design basis scenarios to be more conservative. Therefore, the licensee's design basis scenarios will be used in this evaluation.

The staff calculated heat loads for the fuel inventories representative of the limiting design basis scenarios presented by the licensee. The staff used the methodology of both Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling," and ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," in calculating the heat loads. The staff also assumed a period of approximately 3.5 years of full power operation for all stored fuel. The normal discharge and the full-core off-load to the SFP were assumed to occur following 168 hours of decay in the reactor vessel. Since the staff performed a steady-state calculation, the total heat load was assumed to be constant for each scenario.

Using these heat load values, the heat exchanger temperature effectiveness factor calculated by the licensee, and design data from the UFSAR, the staff then calculated SFP steady-state temperatures assuming heat removal only through the SFP heat exchanger. Calculated maximum steady-state temperatures for the limiting normal discharge and full-core off-load scenarios were determined to be 148°F and 146°F, respectively. These values differ from the values presented in the licensing report results primarily due to a different assumed heat load for the oldest 3438 fuel assemblies and a different assumed component cooling water inlet temperature to the SFP heat exchanger. These differences resulted in the licensee's analysis of bulk pool temperature being more conservative. Therefore, the bulk pool temperature calculated by the licensee will be evaluated here.

The most limiting normal operating condition was found to be the normal refueling discharge with one SFP cooling system train inoperable. This scenario was calculated to result in a maximum SFP temperature of 159.5°F. The licensee has indicated that all SFP support systems are qualified to 200°F, including the purification system resin. Although long-term exposure of concrete structures to temperatures in excess of 150°F may result in damage to these structures, the staff does not consider this to be a concern here. Based on the transient nature of the SFP temperature, the short time SFP bulk temperature is calculated to be above 150°F, the conservative approach of the calculation, and the heat transfer that exists through the concrete, the staff concludes that the bulk temperature of the SFP concrete structure will not exceed 150°F for a period sufficient to cause structural damage. Since the postulated maximum SFP bulk temperature has not been found to result in damage to structures or systems, the staff concludes that the intent of the guidance of Section 9.1.3 of the SRP is met with regard to providing adequate cooling for the postulated spent fuel inventory resulting from normal operations.

The maximum SFP bulk temperature for the abnormal full-core off-load condition assuming both trains of SFP cooling are in operation, calculated to be 143.8°F, is below the temperature associated with the onset of bulk boiling. Therefore, the guidance of Section 9.1.3 of the SRP is met with regard to providing adequate cooling for the postulated spent fuel inventory under abnormal operating conditions.

For the scenarios of concern, the licensee calculated a minimum time of 5.74 hours to reach bulk boiling conditions in the SFP following a loss of all forced cooling. Based on the existing procedure to supply make up water to the SFP following a loss of forced cooling and the availability of a number of alternate sources of make up water, the staff concludes that adequate time is available to provide make up water to the SFP prior to the onset of bulk boiling. The staff calculated the maximum rate of water loss for the postulated fuel inventory following the onset of SFP bulk boiling to be approximately 120 gpm. The accepted source of make up water is capable of supplying 500 gpm. Therefore, the staff finds that the guidance of Section 9.1.3 of the SRP is met with regard to provision of make up water.

For the potential fuel inventory following the proposed reracking of the SFP, the cooling and make up water supply to the SFP is adequate to meet the intent of applicable guidance contained in Section 9.1.3 of the SRP. Therefore, the staff finds the proposed reracking acceptable with regard to potential SFP thermal-hydraulic concerns. Licensee calculations of local fuel cladding temperature provides additional assurance that SFP cooling is adequate to protect cladding integrity following the proposed reracking.

2.4 Structural Integrity

The SFP is a reinforced concrete structure and is designed as a Seismic Category I structure. The dimension of the pool is approximately 39 feet wide, 58 feet long and 50 feet high with 5 feet thick slab. Wetted surfaces of the pools are lined with stainless steel to ensure water tight integrity.

The loading on the pool consists of static, dynamic, and thermal loads. Static loads include weight of pool structure, water in the pool, and weight of fuel assemblies and rack modules. For the dynamic loads, both design basis earthquake (DBE) and operating basis earthquake (OBE) are considered. In addition, stresses generated from a thermal gradient across the thick concrete walls and slab due to temperature differential between the pool water and the atmosphere external to the slab and walls are investigated.

The SFP was analyzed using the finite element method. Only the linear option of the ANSYS was utilized to perform the analysis. Strength design methods were used and the results of the load components were combined in accordance with Section 3.8.4 of the SRP.

Structural concrete supporting the pool slab strength capacities in terms of moment, shear, and buckling of columns were evaluated in accordance with the Building Code Requirement of the American Concrete Institute, ACI 318-89. Concrete strength capacities were then compared with anticipated loads on the pool structure discussed above. It was found that the strength capacities of the concrete were more than required in all critical regions. The critical regions examined were the pool slab and wall sections adjoining the pool slab as well as columns.

An increase of a significant number of fuel assemblies from that in the existing rack arrangement did not alter safety margin appreciably because they only contributed to little more than 10 percent to the total dead weight since the massive concrete structure and water in the pool are the major contributors.

The staff, therefore, concludes that the SFP would continue to support additional loads due to additional fuel for normal and severe environmental, as well as accident conditions, and maintain its integrity.

2.5 Refueling Accidents

The licensee has investigated the consequences of dropping a new or spent fuel assembly as it is being moved over stored fuel.

The following three accidents were evaluated by the licensee: (a) a fuel assembly is dropped from an elevation 36 inches above a storage location and impacts the base of the module, (b) a fuel assembly is dropped from an elevation of 36 inches above the rack and hitting the top of the rack, and (c) the same as (b) except that the fuel assembly is assumed to be dropped in an inclined manner on the top of the rack. The height of the fuel assembly drop is limited to 36 inches by the constraints of the fuel transfer equipment.

In the case of a fuel assembly drop on a module base (case (a) above), the licensee determined that there will be no change in the spacing between cells, local deformation of the baseplate in the neighborhood of the impact will occur, but the dropped assembly will be contained and not impact the pool liner. When a fuel assembly is dropped on the top of the rack (case (b) above), the licensee's analysis indicated that although local deformation occurs, it is confined to a region above the active fuel area thus not altering the subcriticality of the fuel assemblies. Investigation of the inclined fuel assembly drop (case (c)) has shown that this case is bounded by case (b).

The degree of damage to the rack is obtained by equating strain energy of the deforming rack module to the kinetic energy of the falling fuel assembly. A refueling accident affects only a few fuel assemblies and accessibility to the pool allows immediate assessment of the damage and subsequent repair if needed. For this reason, the safety impact of the accident to the general public would be minimal. The staff finds the licensee's evaluation reasonable and agrees with the conclusion that public safety would not be compromised in case of a refueling accident.

2.6 High Density Racks

The spent fuel storage racks are Seismic Category I equipment and, therefore, they are required to remain functional during and after a DBE. They are neither anchored to the pool floor, the pool wall, nor structurally interconnected. Each rack module is provided with leveling pads which support the rack. The fuel rack structure is a folded metal plate assemblage welded to a baseplate and supported on four legs except for an odd shaped rack where more than four legs are provided. The rack is modeled by a system of springs and lumped masses. There are many aspects of the modeling that make rack dynamic response analysis highly nonlinear. There are gap spring elements that simulate gaps between fuel assembly and rack cell. Sliding elements are provided to model sliding action of rack modules with respect to the pool floor. Water in the pool is modeled by equivalent hydrodynamic mass. These elements are modeled into a computer model suitable for analyses by a code named DYNARACK which executes time history integration of the resulting nonlinear differential equations of motion representing 24 degrees of freedom for the system of rack and fuel assemblies.

The licensee concluded that the rack modules would not lift from the pool floor when both horizontal and vertical DBEs are considered. The licensee also concluded that the racks would not impact each other nor the vertical pool walls. The stresses in the racks were found to be small.

After reviewing additional information provided to assist the evaluation, the staff concluded that the verification process employed for the analysis, including the DYNARACK code, was rather limited. Specifically, there have been no realistic physical tests to verify the analytical results obtained by the code. In addition, the licensee has not demonstrated that errors from the analytical method are within an acceptable range and stability of the analytical response is controlled. The staff, for these reasons, performed additional independent assessments to augment the licensee's analyses. The staff assessments are discussed below.

The staff investigated the actual safety margin against overturning of the rack. It was found that the estimated safety factor against overturning for the rack was greater than 2.0. This is larger than the 1.1 factor provided in Section 3.8.5 of the SRP and is acceptable. The investigation was based on the conservation of energy principle, whereby the kinetic energy of the rack was induced by an earthquake and is equated to the potential energy that is needed to raise the rack to a position where the center of gravity of the rack moves beyond the lines connecting the two supporting legs of the rack. The procedure provides an overall assessment of the stability of the free standing racks under vibratory ground motion induced by an earthquake.

Another aspect of the rack response that the staff considered is the structural integrity of the rack itself with regard to lateral impact. Maximum impact that can be realized is that extreme bounding condition where there is no friction between the rack support and pool floor. The licensee's simulated test indicated that there is no impact between the boxes and between the box and the container wall. The box simulates the rack and the container simulates the SFP. The container is filled with water and the boxes are placed in the container. For the test, the boxes are suspended from above, thus providing no solid contact between the bottom of the box and the floor of the container. Dynamic motion is applied only to the container. The test provides a good indication that water between the objects provides a significant cushion, thus not allowing the closely spaced objects to impact. Even if there are isolated impacts, forces are not expected to be significant and the basic function of the rack to keep the fuel assemblies upright would not be compromised.

Lastly, the staff investigated the stress level in the rack for a DBE when the base of the rack is fully fixed, thus not allowing any sliding at all. This is another bounding case of opposite extreme from the above bounding case where zero friction is considered. A simple hand calculation indicated that the stresses in the rack are small and below the allowable stresses.

Based on the above discussion and review of the licensee's submittal, the staff concludes that the rack modules will perform their function during and after a DBE in combination with other applicable loads.

2.7 Occupational Exposure Control

The licensee estimated in its July 26, 1991, application that total occupational exposure for planned reracking activities would be between 6 and 11 person rem.

In preparing this estimate, the licensee identified a series of expected activities or steps expected to be performed during the reracking operation. These steps included such activities as removing, washing, and decontaminating empty racks, removing underwater appurtenances, installing new racks, and preparing old racks for shipment.

The licensee has indicated that diving operations will be performed to remove track lighting and miscellaneous piping within the SFP; however, no deep diving operations are to be performed. During a phone conversation on March 18, 1992, and by letter dated April 1, 1992, the licensee committed to follow the guidance outlined in draft Regulatory Guide DG-8006 (Control of Access to High and Very High Radiation Areas in Nuclear Power Plants, Appendix A) to ensure procedures are in place and followed for radiation protection of divers during diving operations.

The licensee noted that detailed procedures prepared with consideration of ALARA principles will be utilized. In addition, IMPC noted that continuous air samples would be utilized where a potential for significant airborne activity exists and that personnel would wear protective clothing and, as necessary, respiratory protective equipment. Further, work activities would be governed by a Radiation Work Permit (RWP) which would specify appropriate individual personnel monitoring equipment. In addition to routine use of pocket dosimeter and thermoluminescent dosimeters, extremity badges and alarming dosimeters will be utilized as required. The licensee stated that work activities, personnel traffic, and equipment movement will be monitored and controlled such that contamination is minimized and that personnel exposures are maintained as low as reasonably achievable (ALARA).

2.8 Solid Radioactive Waste

The licensee also stated that after removal and decontamination of the existing storage racks, they would be packaged and shipped offsite to licensed processing and disposal facilities and that such shipments will conform to applicable State and Federal Department of Transportation requirements.

In addition, the licensee noted that while a certain amount of additional (spent) resins (estimated at 10-30 cubic feet) may be generated by the pool cleanup system on a one-time basis, a significant increase in the volume of solid waste produced was not expected to result from the increased storage capacity.

Based on our review, the staff finds that the licensee's plans for disposal of solid radioactive waste generated in connection with the planned reracking operation meet our criteria and are acceptable.

2.9 Design Basis Accidents

In the staff's evaluation of Amendment No. 136 to the D. C. Cook Unit 1 license and Amendment No. 121 to the D. C. Cook Unit 2 license which pertained to the receipt of fuel enriched to 4.95 weight percent U-235, fuel handling accident consequences associated with the use of high burnup fuels were considered. In this evaluation, the staff reconsidered the potential offsite dose consequences associated with the design basis fuel handling accident as documented in the D. C. Cook Safety Evaluation Report (SER) dated September 10, 1973. As noted in NUREG/CR-5009, dated February 1988, the release fraction for high burnup fuel (up to 60,000 MWD/MTU) could be as high as 0.12 vice the previously assumed gap fraction of 0.10.

As noted in the staff's evaluation related to Amendments 136 and 121, calculated thyroid doses are increased to 13 rem (compared with the SER value of 11 rem) when the higher gap fraction is considered. This value of calculated thyroid dose remains a small fraction of 10 CFR Part 100 guideline values and is acceptable.

It is noted that this value is about 50 percent higher than that presented by the licensee in the licensing report associated with the current amendment request. It is believed that this discrepancy may be due to the fact that the licensee estimated core inventories using the ORIGIN-2 code instead of the values specified in TID 14844. In either case, however, calculated thyroid doses meet our criteria and are acceptable.

2.10 Technical Specifications

The following TS changes have been proposed as a result of the requested SFP reracking.

(1) Section 3.9.3 - The amount of time the reactor must be subcritical before refueling is changed from 100 hours to 168 hours.

(2) Section 3.9.15 - The applicability for the requirement to maintain 2400 ppm of boron in the SFP is modified from "whenever fuel assemblies with enrichment greater than 3.95 wt. % U-235 and with burnup less than 5,500 MWD/MTU are in the fuel storage pool" to "at all times."

(3) Section 5.6.1.1.b - The nominal center-to-center distance between fuel assemblies placed in the storage racks is decreased from 10.5 inches to 8.97 inches.

(4) Section 5.6.1.1.c - The current storage configuration restrictions for storage of Westinghouse fuel with nominal enrichment greater than 3.95 w/o U-235 and burnup less than 5550 MWD/MTU are superseded. The present two regions are replaced with three regions. The definitions of the regions are no longer dependent on fuel vendor (Westinghouse vs Exxon/ANF).

Two options for storage patterns are provided as Figures 5.6-1 and 5.6-2 of the proposed TS. A graphical representation of the regional definitions is provided as Figure 5.6-3 of the proposed TS.

(5) Section 5.6.1.2 - The maximum nominal fuel assembly enrichment for fuel stored in the SFP racks is increased from 3.50 to 4.95 w/o U-235 for Exxon/ANF 15 x 15 assemblies, and from 4.23 to 4.95 w/o U-235 for Exxon/ANF 17 x 17 assemblies.

(6) Section 5.3.1 - The enrichment limit for Unit 2 fuel assemblies is modified to indicate that the 4.95 w/o value is a nominal value. The Unit 1 enrichment is modified from 4.0 w/o U-235 to a nominal value of 4.95 w/o U-235.

(7) Section 5.6.4 - The authorized storage capacity of the SFP is increased from 2050 to 3613 assemblies.

In addition, License Condition 2.C.(5) for Unit 1 and License Condition 2.C.(3)(s) for Unit 2 are modified to increase the authorized storage capacity of the SFP from 2050 to 3613 assemblies. As currently worded, the license conditions allow storage of new or irradiated assemblies in any combination. The words "in any combination" is being deleted to reflect the new storage configuration restrictions. Clarification is being made to indicate that the 4.95 w/o U-235 enrichment limit is a nominal value.

Finally, an additional license condition (2.C.(8) for Unit 1 and 2.C.(3)(u) for Unit 2) is added to hold in abeyance the provisions of Specification 3/4.9.7 to allow movement of loads over the spent fuel pool during the reracking project. This condition also ensures that the criteria of NUREG-0612, "Controls of Heavy Loads at Nuclear Power Plants," are in effect when moving loads over the spent fuel pool.

Based on the above evaluation, the staff finds the proposed expansion of the spent fuel from 2050 assemblies to 3613 assemblies and all associated TS changes are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on December 23, 1992, (57 FR 61100). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: January 14, 1993

ATTACHMENT TO LICENSE AMENDMENT NO.152

TO 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 vertical lines indicating the area of change.

REMOVE	INSERT
3/4 9-3	3/4 9-3
3/4 9-18	3/4 9-18
5-4	5-4
5-5	5-5
5-6	5-6
5-7	5-7
----	5-7a
5-8	5-8
5-9	5-9

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 168 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 168 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 168 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

STORAGE POOL BORON CONCENTRATION*

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

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.

*Shared system with Cook Nuclear Plant - Unit 1

DESIGN FEATURES

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 nominally 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 plus or minus 100 cubic feet at a nominal Tavg 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 8.97-inch center-to-center distance between fuel assemblies, placed in the storage racks.

c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

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

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

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

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

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

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

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations and graphically depicted in Figure 5.6-3.

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

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

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

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

Where E = Initial Peak Enrichment

FIGURE 5.6-1: Normal Storage Pattern (Mixed Three Zone)

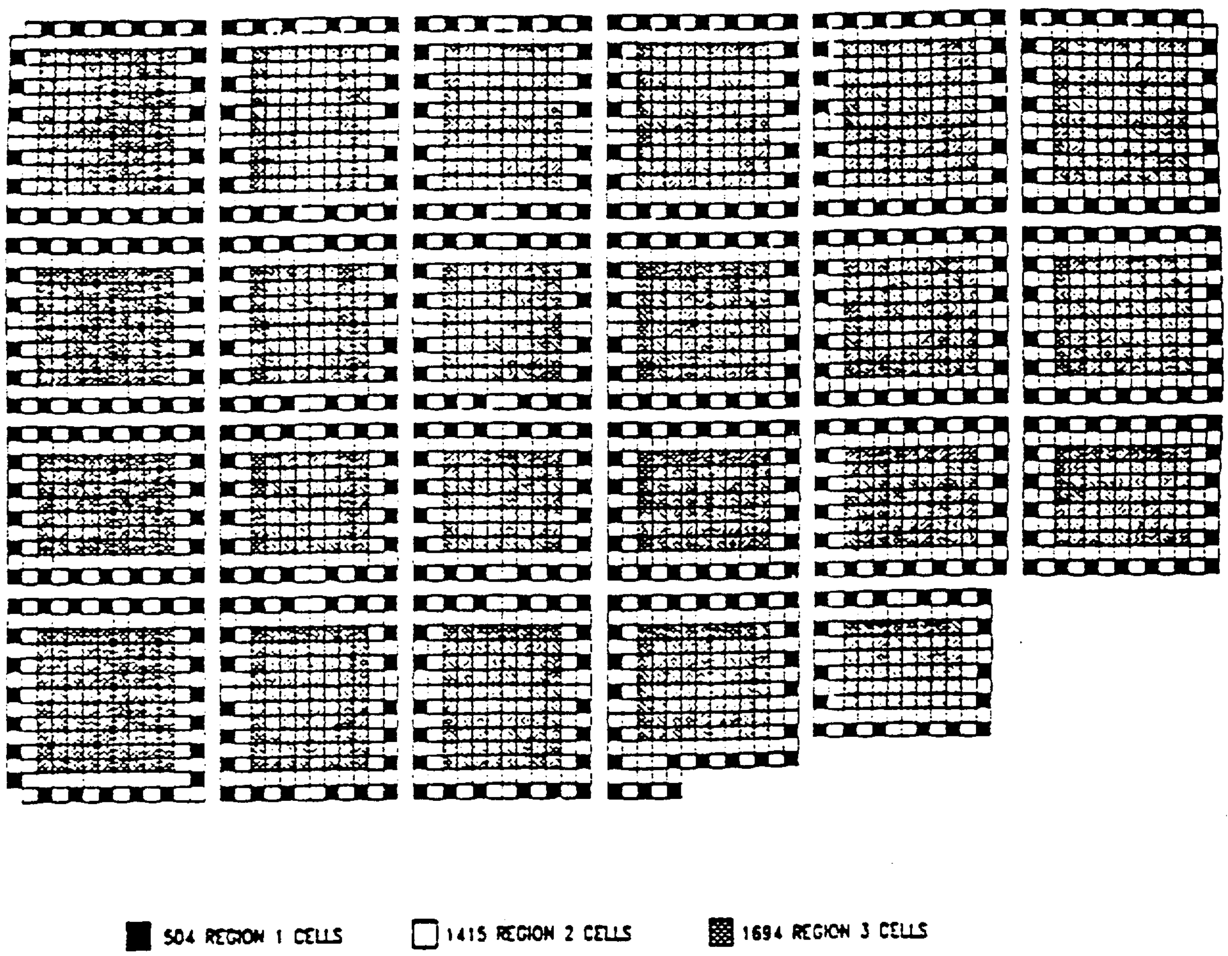
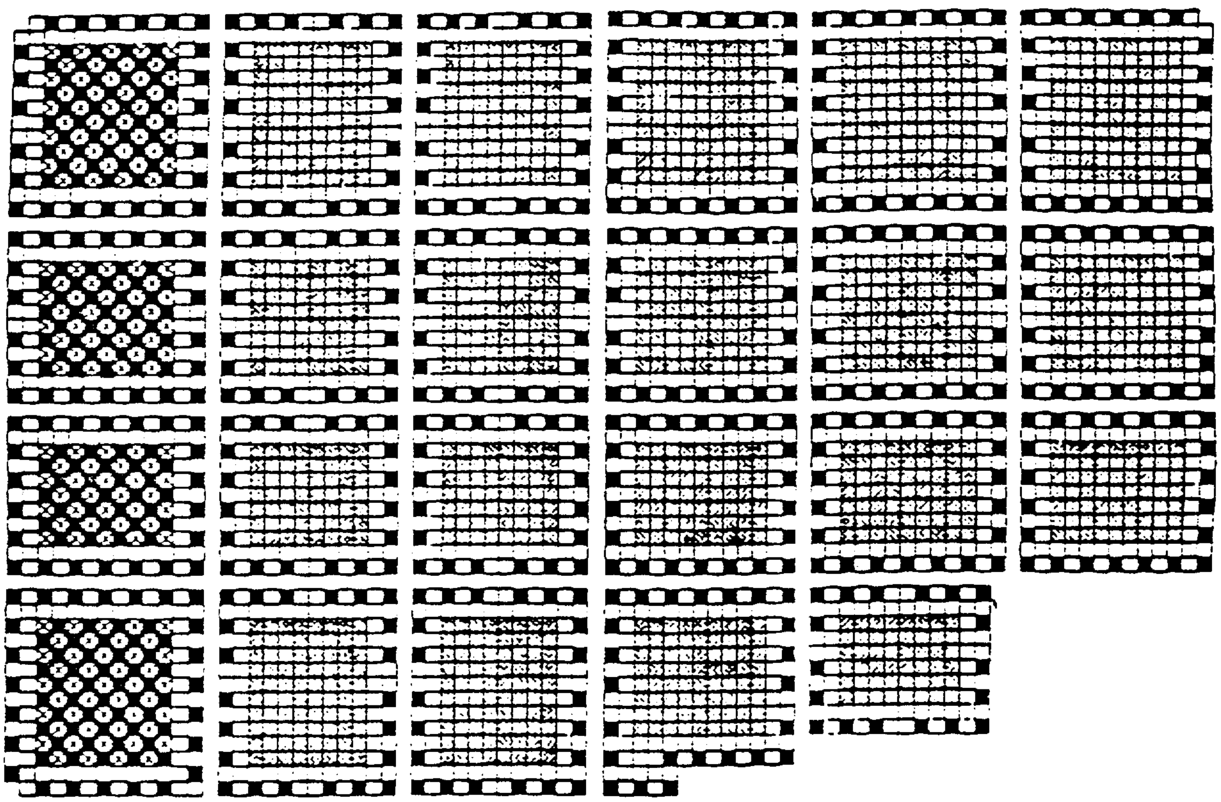
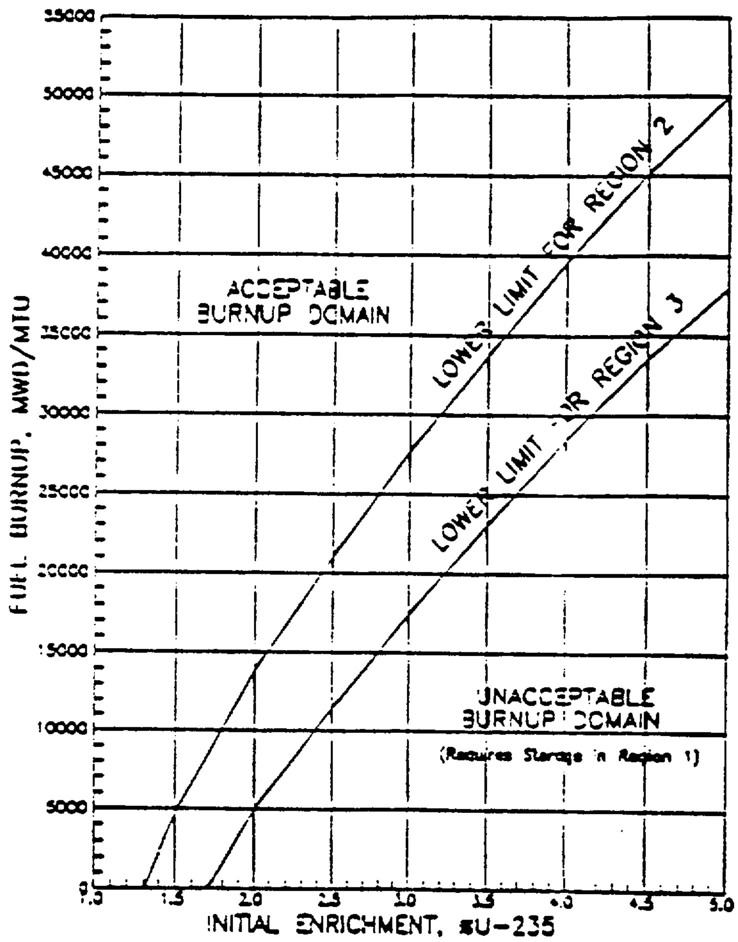


Figure 5.6-2: Interim Storage Pattern (Checkerboard)



158 EMPTY LOCATIONS   661 REGION 1 CELLS   1415 REGION 2 CELLS   1379 REGION 3 CELLS

Figure 5.6-3: Acceptable Burnup Domain in Regions 2 & 3



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	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

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 3613 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 on 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. 152
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 July 26, 1991 as supplemented June 7, 1991 February 4 and April 1, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 152 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, the license is amended by changes to paragraph 2.C.(3)(s) and adding paragraph 2.C.(3)(u) to read as follows:*

   (s) Spent Fuel Pool Storage

   The licensee is authorized to store D. C. Cook, Unit 1 and Unit 2 fuel assemblies, new or irradiated, up to a total of 3613 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 a nominal enrichment greater than 4.95% Uranium-235.

   (u) The provisions of Specification 3/4.9.7 are not applicable for loads being moved over the pool for the duration of the spent fuel pool reracking project. Control of loads moving over the spent fuel pool during the spent fuel pool reracking project shall comply with the criteria of NUREG-0612, "Controls of Heavy Loads at Nuclear Power Plants." Administrative controls shall be in place to prevent any load not rigged in compliance with the criteria of NUREG-0612 from passing over the spent fuel pool with the crane interlocks, required by T/S 3/4.9.7, disengaged.

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

FOR THE NUCLEAR REGULATORY COMMISSION

LBMarsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachments:
1. Changes to the Technical Specifications
2. Page 9 of License*

Date of Issuance: January 14, 1993

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*Page 9 is attached, for convenience, for the composite license to reflect the change.

(r) Amendment No. 6, 06-16-78

Deleted - Amendment No. 68, 04-08-85

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, up to a total of 3613 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 a nominal enrichment greater than 4.95% Uranium-235.

Admt. No. 104, 11/14/88
Admt. No. 152, 1/14/93

(3)(t) Amendment No. 13, 10-16-79

Deleted - Amendment No. 63, 04-27-84

(u) The provisions of Specification 3/4.9.7 are not applicable for loads being moved over the pool for the duration of the spent fuel pool reracking project. Control of loads moving over the spent fuel pool during the spent fuel pool reracking project shall comply with the criteria of NUREG-0612, "Controls of Heavy Loads at Nuclear Power Plants." Administrative controls shall be in place to prevent any load not rigged in compliance with the criteria of NUREG-0612 from passing over the spent fuel pool with the crane interlocks, required by T/S 3/4.9.7, disengaged.

Admt. No. 152, 1/14/93

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations and graphically depicted in Figure 5.6-3.

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

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

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

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

Where E = Initial Peak Enrichment

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

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

FIGURE 5.6-1: Normal Storage Pattern (Mixed Three Zone)

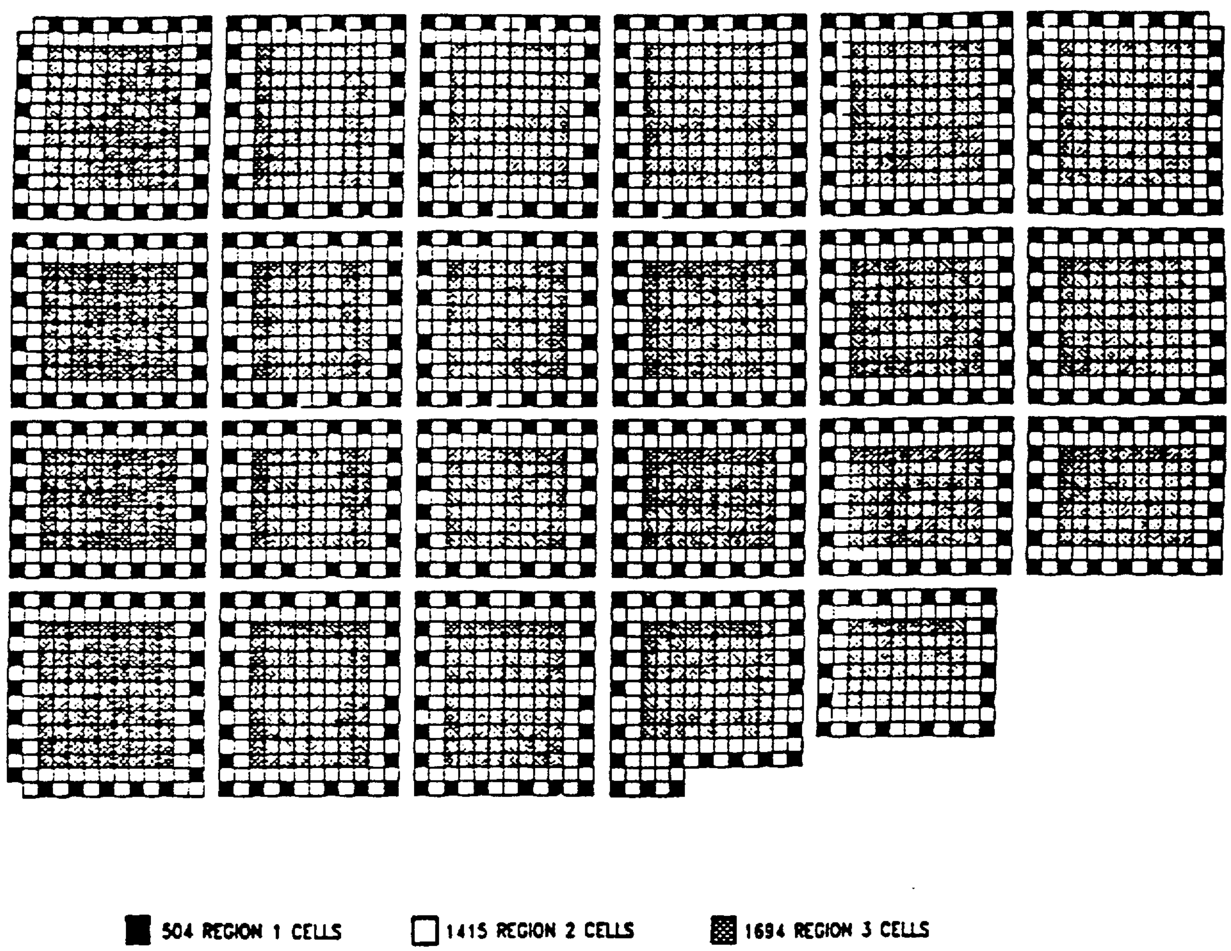


Figure 5.6-2: Interim Storage Pattern (Checkerboard)

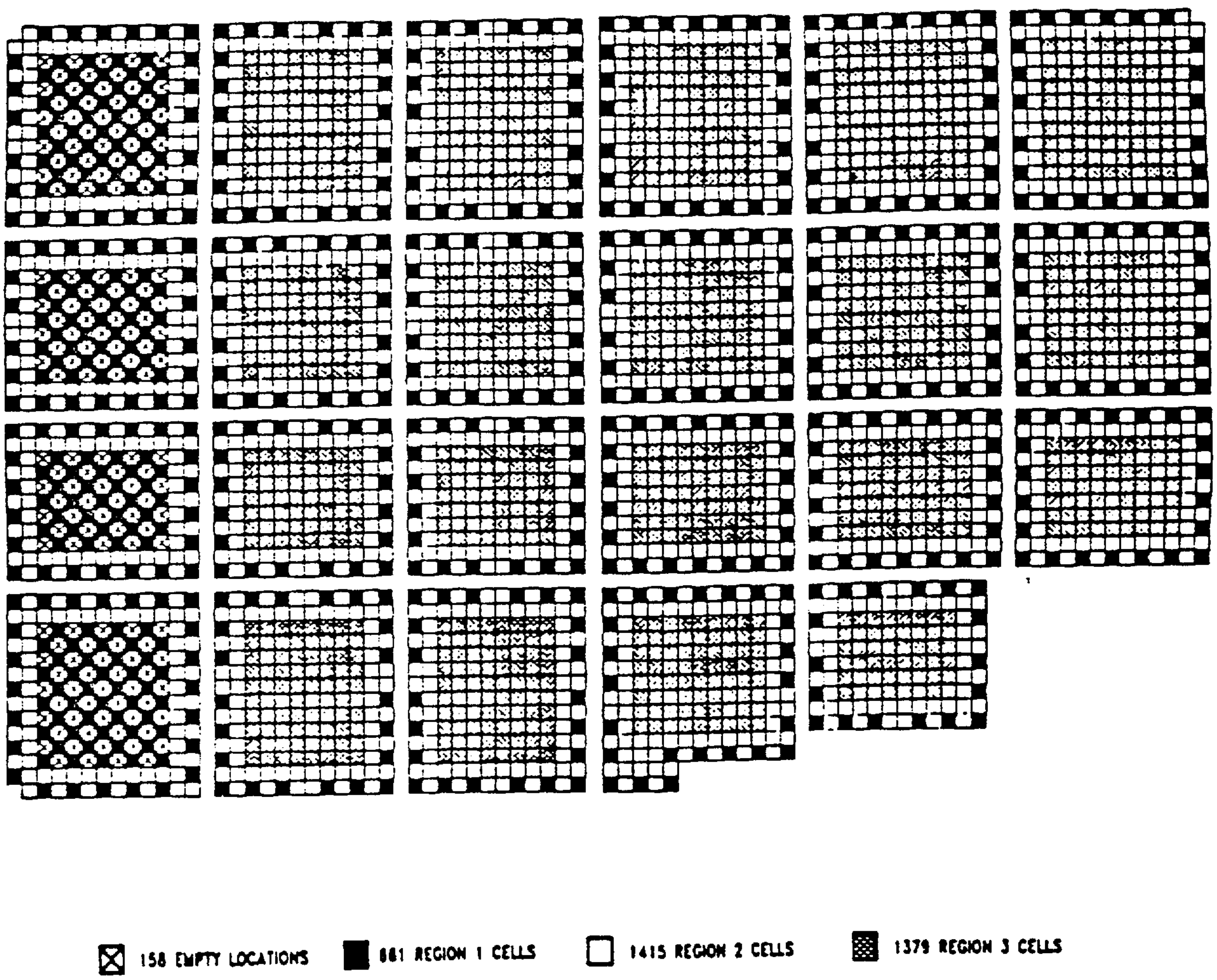
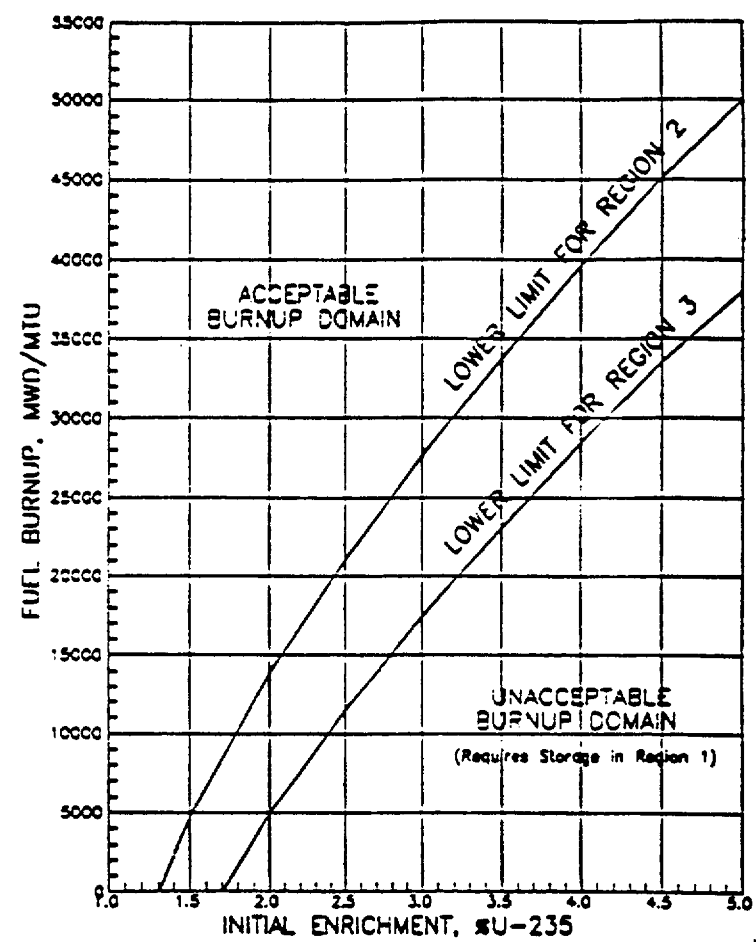


Figure 5.6-3: Acceptable Burnup Domain in Regions 2 & 3



DESIGN FEATURES

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. % 235U
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 3613 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.