March 12, 1992

Docket Nos. 50-315
and 50-316

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. 163 AND 147 TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74 (TAC NOS. M79834 AND M79835)

The Commission has issued the enclosed Amendment No. 163 to Facility Operating License No. DPR-58 and Amendment No. 147 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 February 15, 1991, as supplemented October 8, 1991, and January 14, 1992.

These amendments revise Technical Specification 5.6.1.1 "Criticality - Spent Fuel," for both units. Specifically, the current requirement to store Westinghouse fuel assemblies with fuel enrichments of greater than 3.95 weight percent U-235 and burnup of less than 5,500 MWD/MTU in Region I of the spent fuel pool in a 3-out-of-4 array (one storage cell in each symmetrical array empty) is modified to allow an array of highly reactive fuel "checkerboarded" with adequately burnt fuel with no empty storage locations. Additionally, minor administrative changes, i.e., page renumbering, correcting Table titles are being made.

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,

/s/

John Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

9204130167 920312
PDR ADOCK 05000315
P PDR

Enclosures:
1. Amendment No. 163 to DPR-58
2. Amendment No. 147 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDIII-1	:PM:PD31	:D:PDIII-1	:OGC	:
NAME	:PShuttleworth	:JStang:jkd	:LMarsh	:	
DATE	:3/4/92	: 3/ /92	: / /92	: 3/5/92	:

OFFICIAL RECORD COPY Document Name: AMEND COOK 79834/35

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

March 12, 1992

Docket Nos. 50-315
and 50-316

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. 163 AND 147 TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74 (TAC NOS. M79834 AND M79835)

The Commission has issued the enclosed Amendment No. 163 to Facility Operating License No. DPR-58 and Amendment No. 147 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 February 15, 1991, as supplemented October 8, 1991, and January 14, 1992.

These amendments revise Technical Specification 5.6.1.1 "Criticality - Spent Fuel," for both units. Specifically, the current requirement to store Westinghouse fuel assemblies with fuel enrichments of greater than 3.95 weight percent U-235 and burnup of less than 5,500 MWD/MTU in Region I of the spent fuel pool in a 3-out-of-4 array (one storage cell in each symmetrical array empty) is modified to allow an array of highly reactive fuel "checkerboarded" with adequately burnt fuel with no empty storage locations. Additionally, minor administrative changes, i.e., page renumbering, correcting Table titles are being made.

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,

John Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 163 to DPR-58
2. Amendment No. 147 to DPR-74
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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U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

Township Supervisor
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Post Office Box 818
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Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
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Nuclear Facilities and Environmental
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Mr. S. Brewer
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

DATED: March 12, 1992

AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK
AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DRP-74-D. C. COOK

Docket File
NRC & Local PDRs
PDIII-1 Reading
D.C. Cook Plant File
B. Boger
J. Zwolinski
L. Marsh
P. Shuttleworth
T. Colburn
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (8), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
R. Jones, 8/E/23
C. McCracken, 8/D/1
N. Wagner, 8/D/1
L. Kopp, 8/E/23
ACRS (10)
GPA/PA
OC/LFMB
B. Clayton, R-III

cc: Plant Service list

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111

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. 163
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 February 15, 1991, as supplemented October 8, 1991, and January 14, 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. 163 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 12, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 163

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 marginal lines indicating the area of change.

REMOVE	INSERT
XVI	XVI
5-5	5-5
5-5a	---
5-6	5-6
5-6a	---
5-7	5-7
5-8	5-8
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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 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 types of enrichments greater than 3.95 weight percent U-235 and burnup less than 5,550 MWD/MTU in a checkerboard pattern configuration alternating Category 1 and Category 2 fuel as shown in Figure 5.6-1.

Westinghouse Category 1 and Category 2 fuel definitions are given in Figure 5.6-2.

DESIGN FEATURES (cont'd)

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL (cont'd)

2. A separate region within the spent fuel storage racks, defined as Region 2, shall be established for storage of Westinghouse fuel of an enrichment less than or equal to 3.95 weight percent U-235 or an enrichment greater than 3.95 weight percent U-235 but with a burnup greater than or equal to 5,550 MWD/MTU, and Exxon/ANF fuel of an enrichment less than or equal to 4.23 weight percent U-235 for a 17 x 17 assembly or less than or equal to 3.50 weight percent U-235 for a 15 x 15 assembly.

3. The boundary between the Regions 1 and 2 mentioned above shall be such that the checkerboard pattern storage requirement of Region 1 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

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.

DESIGN FEATURES (cont'd)

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 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.

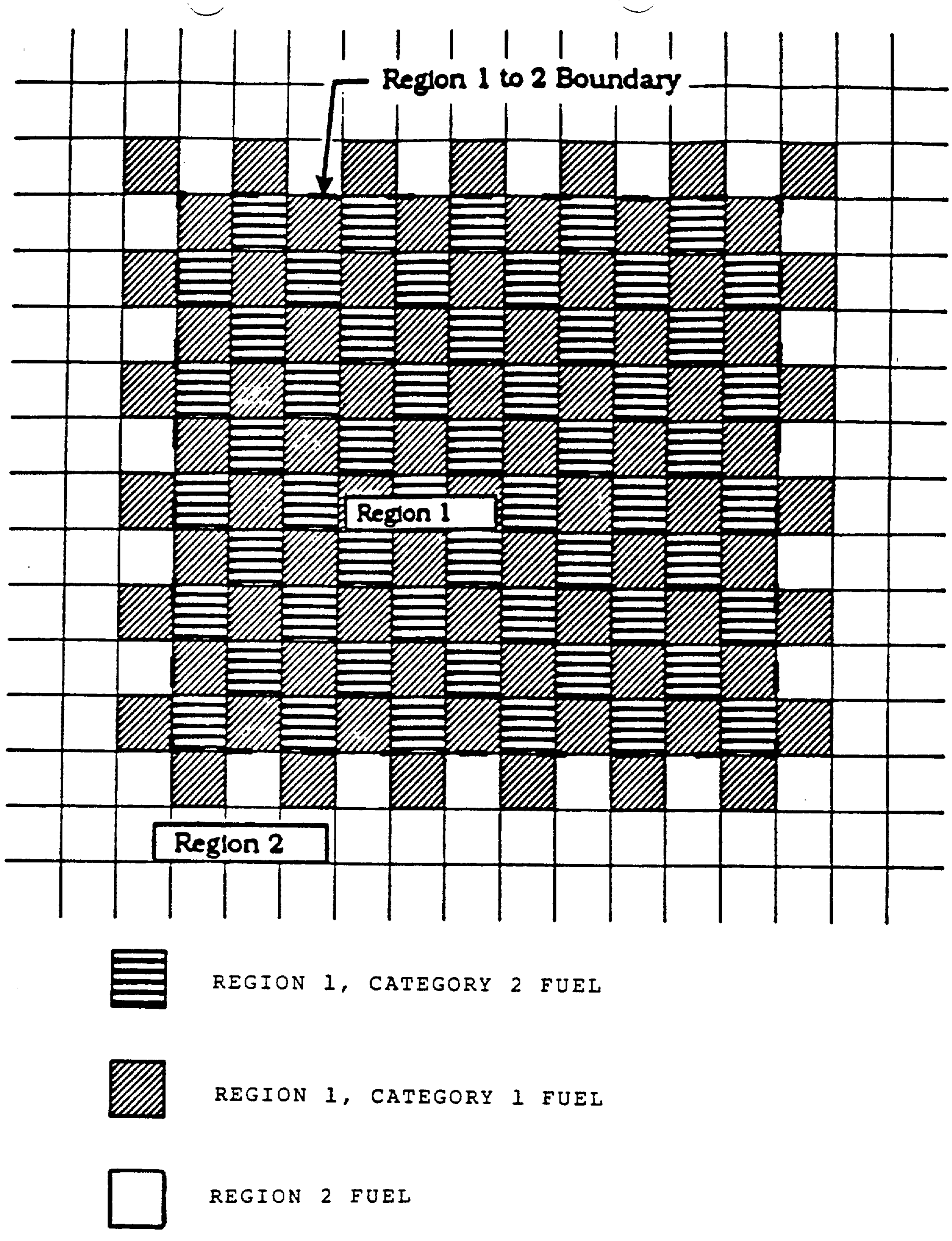

FIGURE 5.6-1

DONALD C. COOK NUCLEAR PLANT
SCHEMATIC FOR SFP INTERFACE BOUNDARY BETWEEN REGIONS 1 AND 2

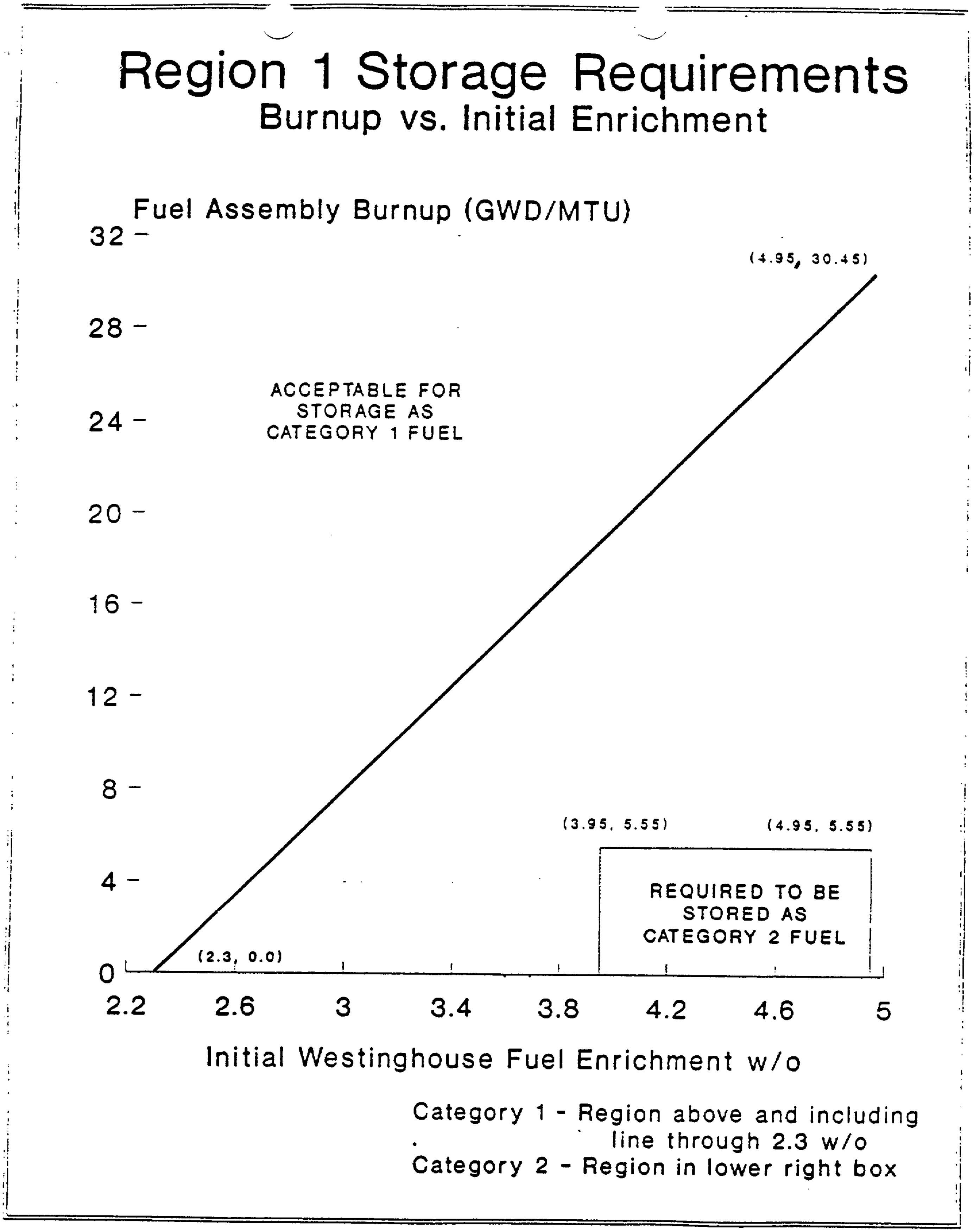

FIGURE 5.6-2

DONALD C. COOK NUCLEAR PLANT
SFP REGION 1 BURNED FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U-235
ENRICHMENT CURVE

TABLE 5.9-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at less than or equal to 100°F/hr and 200 cooldown cycles at less than or equal to 100°F/hr (pressurizer cooldown at less than or equal to 200°F/hr).	Heatup cycle - T_{avg} from less than or equal to 200°F to greater than or equal to 547°F. Cooldown cycle - T_{avg} from greater than or equal to 547°F to less than or equal to 200°F.
	80 loss of load cycles.	Without immediate turbine or reactor trip.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	200 large step decreases in load.	100% to 5% of RATED THERMAL POWER with steam dump.

TABLE 5.9-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe greater than 6 inches equivalent diameter.
	Operating Basis Earthquakes	400 cycles - 20 earthquakes of 20 cycles each.
	50 leak tests.	Pressurized to 2500 psia.
	5 hydrostatic pressure tests	Pressurized to 3107 psig.
Secondary System	1 steam line break	Break in a steam line greater than 5.5 inches equivalent diameter.
	5 hydrostatic pressure tests	Pressurized to 1356 psig.

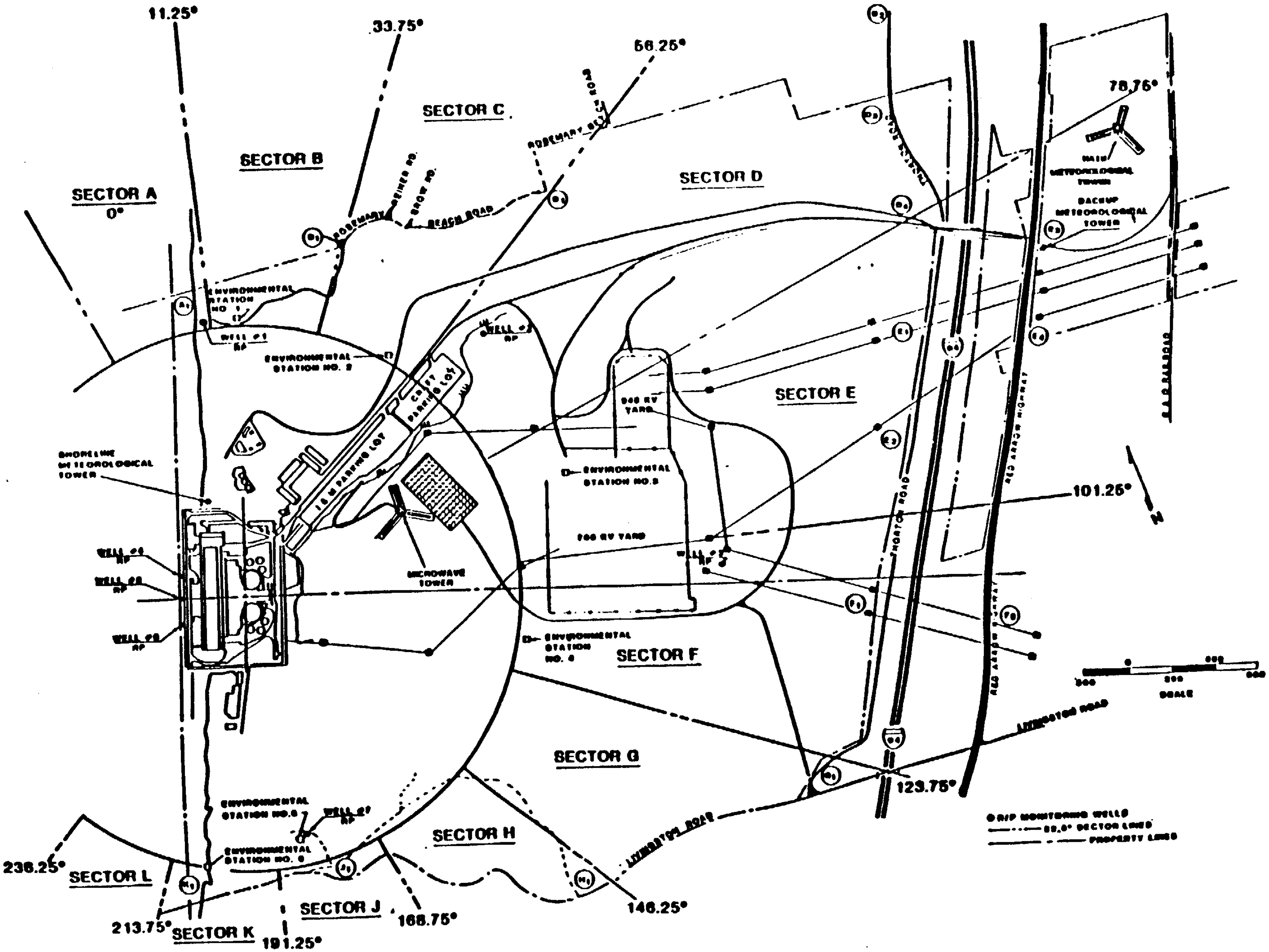

FIGURE 5.1-3: SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

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. 147
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 February 15, 1991, as supplemented October 8, 1991, and January 14, 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. 147 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

For

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 12, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 147

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
XVI	XVI
5-5	5-5
5-5a	---
5-6	5-6
5-7	5-7
5-8	5-8
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Criticality - New Fuel 5-6

Drainage 5-7

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VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 $\pm$ 100 cubic feet as a nominal Tavg 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 types of enrichment greater than 3.95 weight percent U-235 and burnup less than 5,550 MWD/MTU in a checkerboard pattern configuration alternating Category 1 fuel and Category 2 fuel as shown in Figure 5.6-1.

   Westinghouse Category 1 and Category 2 fuel definitions are given in Figure 5.6-2.

   2. A separate region within the spent fuel storage racks, defined as Region 2, shall be established for storage of Westinghouse fuel of an enrichment less than or equal to 3.95 weight percent U-235 or an enrichment greater than 3.95 weight percent U-235 but with a burnup greater than or equal to 5,550 MWD/MTU, and Exxon/ANF fuel of an enrichment less than or equal to 4.23 weight percent U-235 for a 17 x 17 assembly or less than or equal to 3.50 weight percent U-235 for a 15 x 15 assembly.

   3. The boundary between the Regions 1 and 2 mentioned above shall be such that the checkerboard pattern storage requirement of Region 1 shall be carried into Region 2 by at least one row as shown in Figure 5.6-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL (cont'd)

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	4.95
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

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.

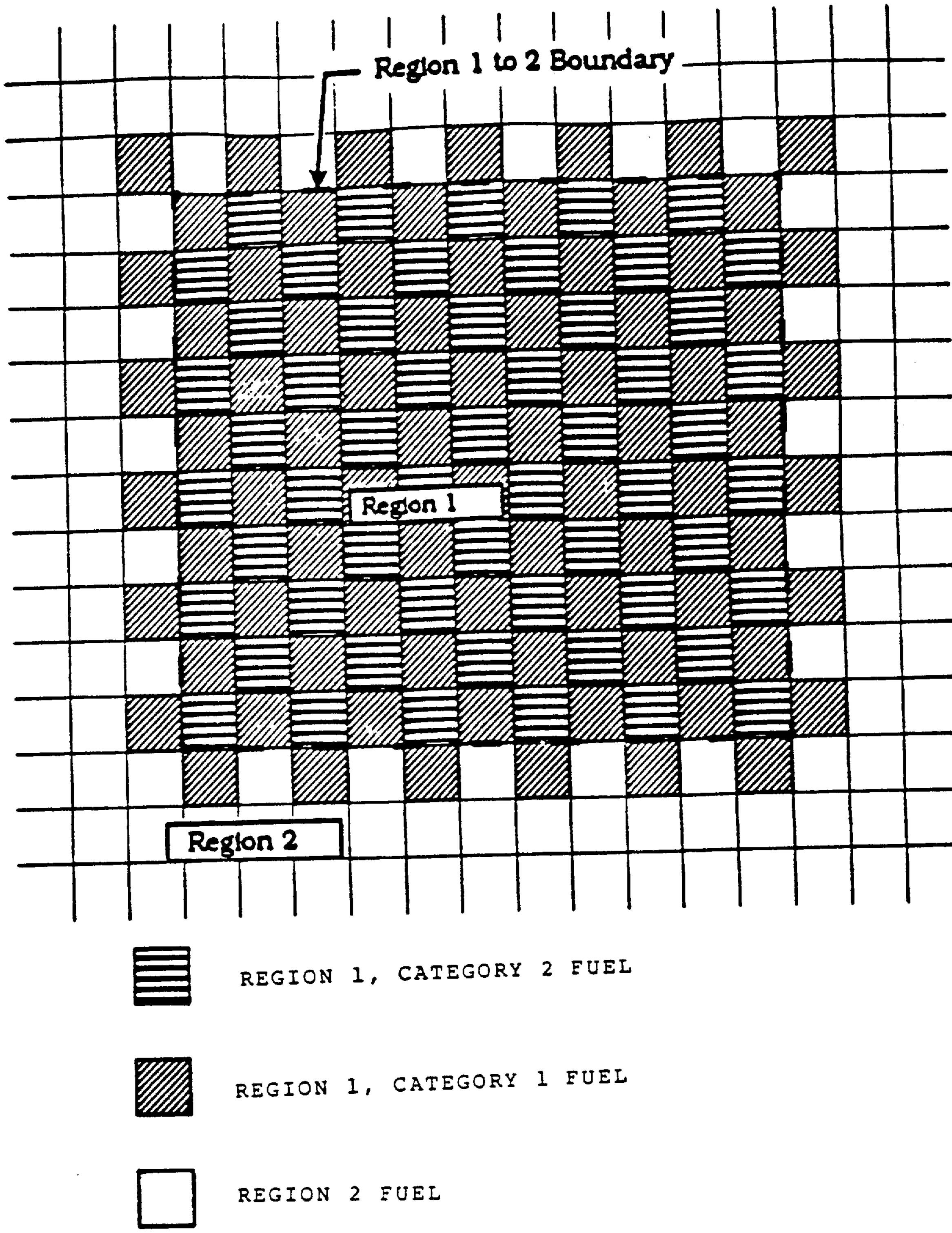

FIGURE 5.6-1
DONALD C. COOK NUCLEAR PLANT
SCHEMATIC FOR SFP INTERFACE BOUNDARY BETWEEN REGIONS 1 AND 2

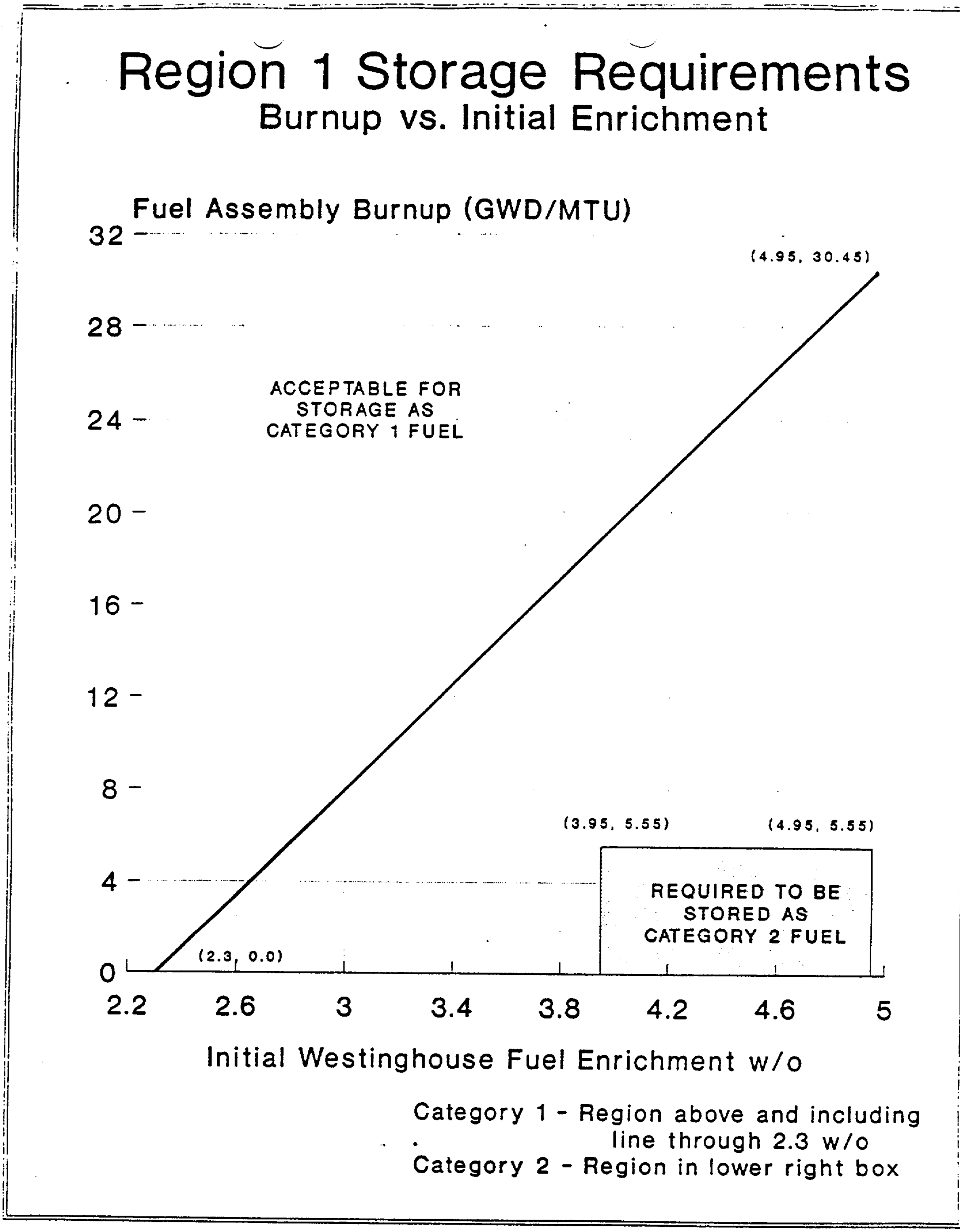

FIGURE 5.6-2

DONALD C. COOK NUCLEAR PLANT
SFP REGION 1 BURNED FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U-235 ENRICHMENT CURVE

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at less than or equal to 100°F/hr and 200 cooldown cycles at less than or equal to 100°F/hr (pressurizer cooldown at less than or equal to 200°F/hr).	Heatup cycle - T_{avg} from less than or equal to 200°F to greater than or equal to 547°F. Cooldown cycle - T_{avg} from greater than or equal to 547°F to less than or equal to 200°F.
	80 loss of load cycles.	Without immediate turbine or reactor trip.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	200 large step decreases in load.	100% to 5% of RATED THERMAL POWER with steam dump.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe greater than 6 inches equivalent diameter.
	Operating Basis Earthquakes	400 cycles - 20 earthquakes of 20 cycles each.
	50 leak tests.	Pressurized to 2500 psia.
	5 hydrostatic pressure tests	Pressurized to 3107 psig.
Secondary System	1 steam line break	Break in a steam line greater than 5.5 inches equivalent diameter.
	5 hydrostatic pressure tests	Pressurized to 1356 psig.

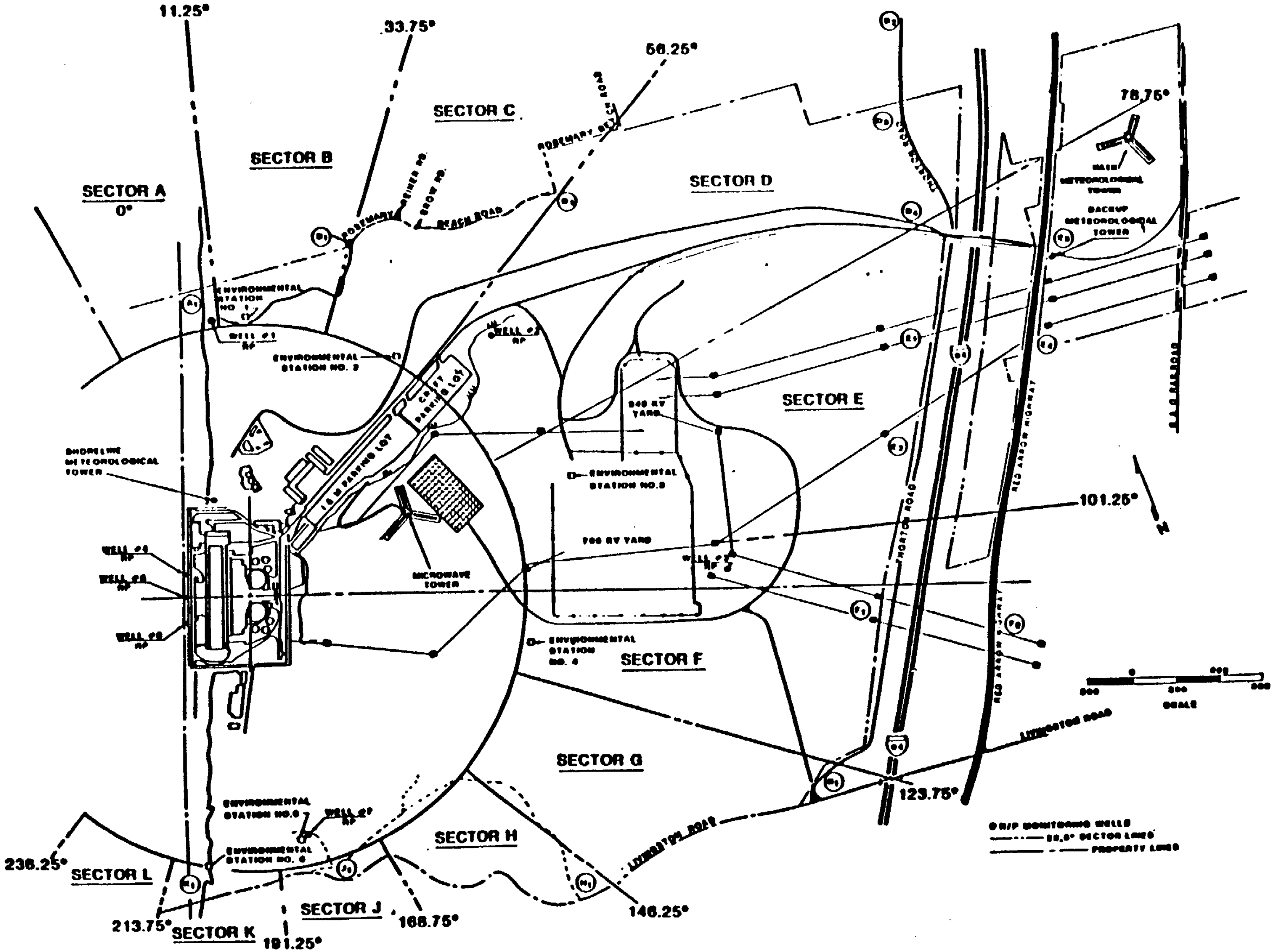

FIGURE 5.1-3: SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 147 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 February 15, 1991, Indiana Michigan Power Company (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, Unit Nos. 1 and 2. The proposed amendments would revise TS 5.6.1.1 "Criticality - Spent Fuel" to allow storage of spent fuel assemblies in a 2-out-of-4 "checkerboard" array pattern alternating highly reactive fuel with adequately burnt fuel and leaving no open storage locations. Currently, the TS require Westinghouse spent fuel stored in Region I of the spent fuel pool with enrichments of greater than 3.95 weight percent (w/o) U-235 and burnup of less than 5,500 MWD/MTU (megawatt-days per metric ton of uranium) to be stored in a 3-out-of-4 array with the fourth storage location in each symmetrical pattern empty. In addition, Figure 5.6-1 would be revised and Figure 5.6-2 would be added. The proposed amendments also make some minor administrative changes and corrections such as correcting the number in the title of a TS Table. By letter dated October 8, 1991, the licensee submitted additional information in response to staff questions raised during a conference call on October 1, 1991. By letter dated January 14, 1992, the licensee provided information concerning the environmental impact of the proposed amendment. These submittals did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Criticality Analysis

The D. C. Cook spent fuel pool storage racks were previously analyzed for the storage of Westinghouse 15x15 and 17x17 fuel assemblies as two separate regions. Region 1 was analyzed for nominal enrichments up to 4.95 w/o U-235 assuming a 3-out-of-4 assembly storage configuration. Region 2 was analyzed for nominal enrichments up to 3.95 w/o U-235 using all available storage locations. Enrichments greater than 3.95 w/o were also allowed in Region 2 provided restrictions on burnup were met. The current analysis considers an alternate fuel assembly storage pattern for Region 1 which uses all available storage locations by checkerboarding burned and fresh fuel assemblies together within the same region.

The criticality analysis was performed with the KENO-IV Monte Carlo computer code. However, since KENO-IV does not have the capability to deplete fuel assemblies, the PHOENIX transport theory code was used for burnup-dependent reactivity calculations. The use of KENO-IV and PHOENIX for fuel storage criticality analysis has been validated by comparison to fuel storage critical experiments. The use of PHOENIX for fuel storage burnup credit criticality analysis has been validated by comparison to experiments where the isotopic fuel composition was examined following discharge from a reactor. These benchmarks indicate that the analytical methods adequately reproduce the experimental values. The staff concludes that these methods and models are acceptable.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level (95/95 probability/confidence) that the effective multiplication factor (k-eff) of the fuel assembly array will be no greater than 0.95 when fully moderated by unborated water. A full loading in the D. C. Cook storage racks of fresh Westinghouse fuel assemblies enriched to greater than 3.95 w/o U-235 could violate this acceptance criterion. Therefore, credit was taken for the reactivity decrease due to burnup in order to load Westinghouse assemblies with initial U-235 enrichments greater than 3.95 w/o fuel assemblies enriched to greater than 3.95 w/o U-235 (but less than 4.95 w/o) can be stored with no restrictions provided they have achieved an accumulated burnup of at least 5,550 MWD/MTU. In order to store fuel assemblies enriched to greater than 3.95 w/o U-235 which do not meet this burnup requirement, restrictions must be placed on allowable storage configurations which effectively increase the center-to-center spacing of the assemblies. In D. C. Cook, this is accomplished by requiring a checkerboard configuration for any fuel assemblies with initial enrichment greater than 3.95 w/o U-235 and burnup less than 5,550 MWD/MTU.

The licensee has analyzed the reactivity of the spent fuel storage racks for an infinite array of 4.95 w/o U-235 fresh fuel loaded in a checkerboard configuration alternating with 4.95 w/o fuel having accumulated burnup of 31,000 MWD/MTU. Based on reactivity equivalencing, the latter is equivalent to 2.3 w/o U-235 fuel at zero burnup.

The resulting k-eff was 0.9455 including all appropriate biases and uncertainties at a 95/95 probability/confidence level. This meets the NRC acceptance criterion and is, therefore, acceptable. The licensee has shown that the checkerboard arrangement specified for the storage of burned and fresh fuel is valid for the storage of Westinghouse 15x15 Standard (STD) and Optimized Fuel Assemblies (OFA), and 17x17 STD, OFA and VANTAGE 5, as well as ANF 15x15 and 17x17 fuel assemblies. Plant procedures will be relied upon to determine whether or not an assembly satisfies the burnup criterion. In order to allow for uncertainties in the determination of assembly burnup, a reactivity uncertainty which increases linearly with burnup to 0.01 delta k at 30,000 MWD/MTU was applied to the PHOENIX calculational results in the development of the Region 1 burnup requirements. The staff finds this adequately conservative and acceptable.

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

1) Fuel assemblies containing the highest authorized enrichment plus 0.05 w/o U-235 to account for enrichment variability and are at the most reactive point in life.

2) No credit is taken for burnable absorbers in the fuel rods.

3) No credit is taken for the build up of fission product poison material.

4) The moderator is pure water at a temperature of 68 ^{o}F.

5) No credit is taken for spacer grids or sleeves.

6) The array is infinite in lateral extent.

7) The minimum poison material loading of 0.02 grams B-10 per square centimeter is used.

8) All available fuel storage cells are utilized.

In addition, the reactivity effects of mechanical and material thickness tolerances, a method bias determined from the benchmark comparisons, a bias to account for poison particle self-shielding, and a bias to account for the reactivity difference between the Region 1 burned checkerboard locations loaded with Westinghouse 17x17 STD fuel at 2.3 w/o U-235 versus Westinghouse 15x15 fuel assemblies at 2.3 w/o U-235 have been incorporated as well as calculational uncertainties at a 95/95 probability/confidence level. The input assumptions and treatment of uncertainties is consistent with staff guidance and is acceptable.

It is possible to postulate events which could lead to an increase in storage rack reactivity, such as dropped or misloaded fuel assemblies. However, for such events, credit may be taken for the 2400 ppm of boron in the spent fuel pool water by application of the double contingency principle of ANSI N16.1-1975. This states that one is not required to assume two unlikely, independent, concurrent events to provide for protection against a criticality accident. The reduction in k-eff caused by the borated water more than offsets any reactivity addition caused by credible accidents. The staff finds this acceptable since TS 3.9.15 requires the boron concentration in the spent fuel pool to be greater than 2400 ppm by measurement when fuel assemblies with enrichments greater than 3.95 w/o U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool.

Revisions to the D. C. Cook TS were proposed in order to change the storage pattern in the spent fuel pool. The proposed TS changes are as follows:

1) TS 5.6.1.1.c.1 identifies Region 1 of the spent fuel pool for storage of Westinghouse fuel types with nominal fuel assembly enrichments of 3.95 w/o or greater. Fuel stored in Region 1 must be stored in a checkerboard configuration alternating Westinghouse Category 1 and Category 2 fuel, as shown in TS Figure 5.6-1. Category 1 and Category 2 fuel are defined in TS Figure 5.6-2.

2) TS 5.6.1.1.c.2 identifies Region 2 of the spent fuel pool. This Region is for the storage of Westinghouse fuel with an enrichment less than or equal to 3.95 w/o U-235 or with an enrichment greater than 3.95 w/o U-235 but with a burnup greater than or equal to 5,550 MWD/MTU, and ANF fuel of an enrichment less than or equal to 4.23 w/o U-235 for a 17x17 assembly or less than or equal to 3.50 w/o U-235 for a 15x15 assembly.

3) TS 5.6.1.1.c.3 addresses the boundary condition between Regions 1 and 2 and requires that the checkerboard pattern for Region 1 must be carried into Region 2 by at least one row. This is illustrated in revised TS Figure 5.6-1.

2.2 Spent Fuel Pool Cooling

As stated previously, the licensee proposes to load Region 1 in a checkerboard fashion of fuel assemblies whose enrichment and burnup are limited so as to permit filling all the locations now empty.

2.2.1 Spent Fuel Pool Cooling System (SFPCS) Design Basis

In response to a staff question, the licensee discussed the design basis for the SFPCS in the October 8, 1991 submittal, stating:

(1) the original FSAR design basis was to have each SFPCS train capable of maintaining the pool temperature at approximately 120°F with the normal heat load (2/3 core) and at approximately 150°F with one 2/3 cores present. The current design basis, as stated in the FSAR, was to maintain the pool temperature below 120°F with the normal heat load generated by 1857 assemblies and below 130°F when one complete core is unloaded in addition to the 1857 fuel assemblies already stored there. A full core (193 assemblies) in addition to 1857 stored assemblies fills the pool to completion (2050 fuel assembly storage locations).

This design basis (for two SFPCS trains in operation) was found to be acceptable by the staff in an SER dated October 16, 1979. In this SER the staff also found the licensee's plan to change the number of SFP storage assembly locations from 500 to 2050 to be acceptable. This basis was found acceptable by the staff up to the present time which includes the last safety evaluation concerning storage of fuel in the SFP. This evaluation, dated November 10, 1988, provided for storage of fuel with an average exposure of 50,000 MWD/MTU and fuel enrichments as great as 4.95 w/o U-235. However, the licensee was required to limit the enrichment of fuel to 4.23 w/o U-235 because of non-thermal/hydraulic concerns within the SFP.

In view of the foregoing, the staff finds the design basis for pool coolant of less than 130°F for a full core offload and less than 120°F for a normal offload to be acceptable.

2.2.2 Heat Generation in SFP

2.2.2.1 Licensee Calculations

The licensee referred to the analyses conducted in the October 11, 1988 submittal where average burnups as great as 50,000 MWD/MTU were considered. In that case the licensee calculated the decay heat for an offloaded core from Unit 2 (with a power level of 3411 MWT as contrasted to 3250 MWT for Unit 2) to be 35.5 MBTU/HR (million BTU/HR). While the licensee did not report the total, it is assumed that the licensee added the heat loads for the rest of the spent fuel assemblies used to fill the pool, 6.9 MBTU/HR, for an approximate total of 42 MBTU/HR. The corresponding normal offload rate was reported as 19 MBTU/HR. The licensee also noted that a calculation performed in 1983, for internal use, used values of 42 and 21 MBTU/HR, respectively.

2.2.2.2 Independent Staff Calculations

The staff estimated the full core offload heat generation rate to be 46.9 MBTU/HR when using the method specified in standard review plan (SRP) Section 9.1.3. This value decreases slightly to 45.7 MBTU/HR when using a value of 156 hours, as used by the D. C. Cook licensee, instead of the 150 hours specified in the SRP. The value for a normal core offload, using the SRP technique becomes 19.2 MBTU/HR utilizing the normal offload of 88 SFAs (spent fuel assemblies) for Unit 2. If the staff uses the technique of adding the heat generation rates for a full Unit 2 core discharge at 156 hours using the technique in SRP Section 9.1.3, 37.3 MBTU/HR, to the value of 6.9 MBTU/HR for the remainder of the SFAs stored in the SFP, a value of 44.2 MBTU/HR is arrived at; for the normal 88 SFA discharge, the SRP method is used to arrive at a normal offload heat generation rate of 16.9 MBTU/HR. To this is added 6.9 MBTU/HR, to arrive at a value of 23.8 MBTU/HR for a normal offload. The results for a full core offload and a normal offload are compared below:

Table 1

Calculated Heat Generation Rates (MBTU/HR)

Method	Full Core Offload	Normal Reloading Offload
Licensees	42	21
Staff-SRP	46.9	19.2
Staff-Addition (offload at 150 hrs +6.9 MBTU/HR)	44.2	23.8

The staff finds these proposed changes to TS 5.6.1.1 acceptable.

2.2.2.3 Conclusions Heat Generation Rates

The staff finds the heat generation rates adopted by the licensee to be acceptable on the following basis:

a) the values shown do not differ sufficiently to indicate that one is markedly different so as to suggest a gross difference in approach,

b) the licensee's methodology has been found acceptable in the past and has not changed for the present calculation, and

c) the different values do not materially affect the results shown below, as discussed in areas 2.3 "SFP Coolant Temperatures," and 2.4 "Fuel Element Cladding Temperatures."

2.2.3 SFP Coolant Temperatures

2.2.3.1 Licensee Calculations Calculated

The licensee reported maximum pool coolant temperatures, as follows:

Maximum pool temperature, 120°F
(both trains operating)
Maximum pool temperature, 130°F
(full core offload, both trains operating)
Maximum pool temperatures, 165°F
(full core offload, with one pump out)

2.2.3.2 Staff Calculations

The licensee reported the design data for one train of the SFPCS to be: 14.9 MBTU/HR for a CCW (component cooling water system) inlet temperature of 95°F and outlet of 105°F; for the SFP coolant the temperatures are 120°F inlet and 106.9°F outlet. From this, it is apparent that, with both trains operating, the heat which the SFPCS is capable of removing (29.8 MBTU/HR) is much greater than even the greatest calculated heat generation rate shown for a normal offload, 23.8 MBTU/HR -- thus a temperature lower than 120°F will be obtained for the SFP coolant entering the SFP heat exchanger. Again, using the design data, the staff calculated temperatures of SFP coolant entering the heat exchanger at the various SFP heat generation rates discussed above for a full core offload, as follows:

Table 2

SFP Coolant Temperatures Into

Heat Exchanger vs Pool Heat Generation

	(Two Trains Operating)		
SFP Heat generation Rate, MBTU	42	44	47
SFP Coolant Inlet temperature	130.2	132.5	134.5

These calculations assumed that, for the temperature changes involved, the overall heat transfer coefficient for the SFP heat exchanger remains constant.

2.2.3.3 Conclusions

It appears that, for the lowest heat generation rate shown above, 42 MBTU/HR the coolant temperature is not significantly above the licensee's design basis of "below" 130°F, as calculated by the staff; the licensee reported a maximum coolant temperature of 130°F. For the other heat generation rates shown the coolant temperature attained is much lower than the SRP limit of below boiling (212°F). For the normal offload the inlet SFP coolant temperature would be less than 120°F, as noted above.

In view of the foregoing, the staff finds the fuel pool coolant temperatures calculated by the licensee to be acceptable.

2.2.4 Pool Coolant Heating Time (Without Cooling)

2.2.4.1 Licensee Calculations

The licensee assumed a full core offload from one unit together with a simultaneous normal refueling load from the other unit to develop the heating rate, 53.7 MBTU/HR for the spent fuel pool. This resulted in a heatup rate of 14.7°F/HR, thus, it takes 5.5 hours to reach 212°F (boiling) from 130°F.

2.2.4.2 Conclusions

The conservatism taken by using a very high heat generation rate, together with the assumption of no evaporative heat transfer to the pool atmosphere nor heat transfer to the pool walls is considered sufficient. In addition, these values, 5.5 hours, and 14.7°F/hr rise, seem to be conservative with regard to results from other fuel pools, and, thus, an acceptable value. The time is also found acceptable on the basis of time allowable for operators to secure some mode of water addition to the pool in the remote chance that a full core offload and normal reload offload will occur simultaneously.

2.2.5 Spent Fuel Cladding Temperatures

2.2.5.1 Licensee Calculations

The licensee calculated linear heat generation rates of 53.96 KWT per Unit 2 SFA and 51.85 KWT per Unit 1 SFA. The licensee used these heat generation rates as input to the code XCOBRA-IIIC and determined the flow vs pressure drop for each channel type, 15x15 for Unit 1 and 17x17 for Unit 2. Three runs were conducted for each channel type: (1) a run with inlet coolant at 150°F, (2) a run with inlet coolant at 150°F with 90% channel blockage, and (3) a run with the inlet temperature at 212°F. The results, flow vs. pressure drop, were then put into an iterative computer program which was used to determine total flow through each rack and SFA cell in that rack. The flow thus determined was used in a final COBRA run to give the water temperature at 12 axial locations along the SFA. A separate calculation was used to determine the clad temperature from the surrounding water temperature. The maximum fuel cladding temperatures found were: (1) 187°F for the case of 150% water inlet, (2) 205°F for the 90% blockage case, and (3) 236°F for the case where all cooling had been lost and the coolant inlet temperature to the SFAs was 212°F.

2.2.5.2 Staff Calculations

None performed.

2.2.5.3 Conclusion

The methodology used by the licensee was deemed adequate to the solution of fuel clad temperatures in the fuel pool. In addition, the maximum cladding temperature thus uncovered was similar to values determined by licensees for other pools. The maximum temperature thus determined 236°F is sufficiently low so as to assure no damage to SFAs being stored in the SFP.

The staff found the licensee's design basis for SFP coolant temperatures calculated SFP heat generation rates, SFP coolant temperatures, pool coolant heatup time and spent fuel clad temperatures, acceptable. It is noted that the licensee is planning to rerack the pool so as to change the number of fuel assembly storage locations from the present level of 2050 to 3613 locations. Thus, this present safety evaluation is, in view of the licensee's submitted proposal for increased storage (dated July 26, 1991) of a temporary nature.

With this in mind the staff finds acceptable the storage of fuel in the D.C. Cook spent fuel pool with fuel enrichments as high as 4.95 w/o U-235 and average burnups as high as 50,000 MWD/MTU, with no change in number of locations (2050).

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 March 12, 1992 (57 FR 8785).

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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