Template # NRR-058

March 1, 2000

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BYRON AND BRAIDWOOD - ISSUANCE OF AMENDMENTS ON SPENT FUEL STORAGE RACKS (TAC NOS. MA5150, MA5149, MA5070, AND MA5071)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 112 to Facility Operating License No. NPF-37 and Amendment No. 112 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 105 to Facility Operating License No. NPF-72 and Amendment No. 105 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to Commonwealth Edison Company's application dated March 23, 1999, as supplemented on October 21 and December 15, 1999.

The amendments approve the installation of new Boral high density spent fuel storage racks at Byron and Braidwood stations. The amendments also approve an increase in the spent fuel pool storage capacity from 2,870 assemblies to 2,984 assemblies at each station.

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

Sincerely,

/RA/

George F. Dick, Jr., Sr. Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 112 to NPF-37
2. Amendment No. 112 to NPF-66
3. Amendment No. 105 to NPF-72
4. Amendment No. 105 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. NPF-37

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 23, 1999, as supplemented on October 21, 1999, and December 15, 1999, 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. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 112 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 1, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. NPF-66

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 23, 1999, as supplemented on October 21, 1999, and December 15, 1999, 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. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 112 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NOS. 112 AND 112

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

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

Remove Pages	Insert Pages
3.7.15-1	3.7.15-1
-	3.7.15-2
3.7.16-1	3.7.16-1
3.7.16-2	3.7.16-2
3.7.16-3	3.7.16-3
3.7.16-4	3.7.16-4
3.7.16-5	3.7.16-5
-	3.7.16-6
-	3.7.16-7
4.0-2	4.0-2
-	4.0-3

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be, as applicable:

a. ≥ 300 ppm for Holtec spent fuel pool storage racks; and

b. ≥ 2000 ppm for Joseph Oat spent fuel pool storage racks.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 Each spent fuel assembly stored in the spent fuel pool shall, as applicable:

a. Region 1 of Joseph Oat spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 4.7 weight percent U-235 or satisfy a minimum number of Integral Fuel Burnable Absorbers (IFBAs) for higher initial enrichments up to 5.0 weight percent U-235 to permit storage in any cell location.

b. Region 2 of Joseph Oat spent fuel pool storage racks

Have a combination of initial enrichment, burnup, and decay time within the Acceptable Burnup Domain of Figure 3.7.16-1, 3.7.16-2, or 3.7.16-3, as applicable for that storage configuration.

c. Interface Requirements for Joseph Oat spent fuel pool storage racks

Comply with the Interface Requirements within and between adjacent racks.

d. Region 1 of Holtec spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 5.0 weight percent U-235 to permit storage in any cell location.

e. Region 2 of Holtec spent fuel pool storage racks

Have a combination of initial enrichment and burnup within the Acceptable Burnup Domain of Figure 3.7.16-4.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly into a location which restores compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	-----------------NOTE-------------------- Item a is only applicable for storage of fuel assemblies in Region 1 Holtec spent fuel pool storage racks. Item b is only applicable for storage of fuel assemblies in Region 1 Joseph Oat spent fuel pool storage racks. --- Verify by administrative means the following requirements are met: a. Initial nominal enrichment of the fuel assembly is ≤ 5.0 weight percent U-235. <u>AND</u>	Prior to storing the fuel assembly in Region 1 (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	(continued) b. Initial nominal enrichment of the fuel assembly is ≤ 4.7 weight percent U-235 with less than the minimum number of IFBAs or ≤ 5.0 weight percent U-235 with the minimum number of IFBAs.	
SR 3.7.16.2	------------------NOTE-------------------- Figures 3.7.16-1, 3.7.16-2, and 3.7.16-3 are only applicable for storage of fuel assemblies in Region 2 Joseph Oat spent fuel pool storage racks. Figure 3.7.16-4 is only applicable for storage of fuel assemblies in Region 2 Holtec spent fuel pool storage racks. -- Verify by administrative means the combination of initial enrichment, burnup, and decay time, as applicable, of the fuel assembly is within the Acceptable Burnup Domain of Figure 3.7.16-1, 3.7.16-2, 3.7.16-3, or 3.7.16-4.	Prior to storing the fuel assembly in Region 2
SR 3.7.16.3	------------------NOTE-------------------- Only applicable for storage of fuel assemblies in Joseph Oat spent fuel pool storage racks. -- Verify by administrative means the interface requirements within and between adjacent racks are met.	Prior to storing the fuel assembly in the spent fuel pool

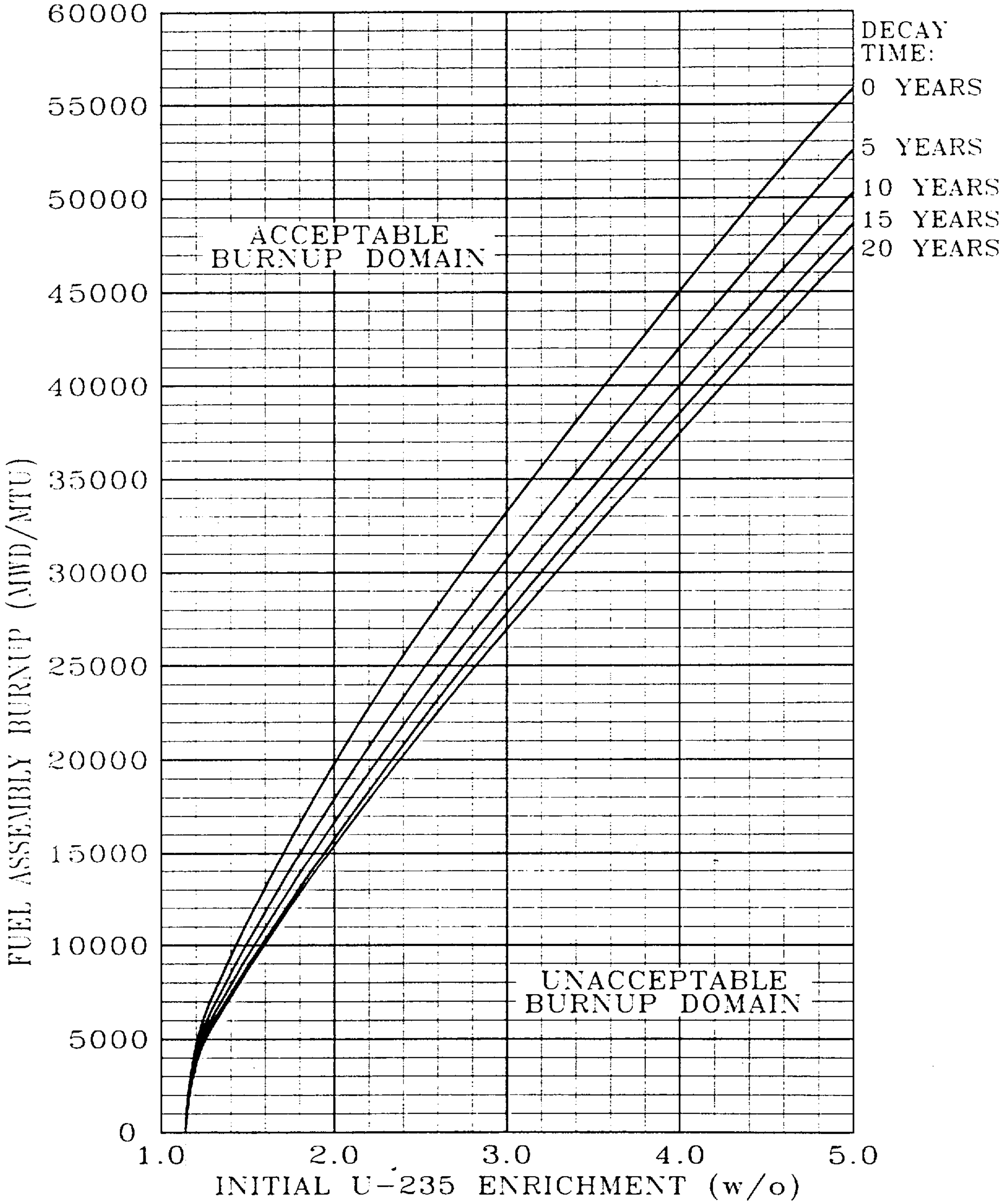

Figure 3.7.16-1 (page 1 of 1)
Region 2 All Cell Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

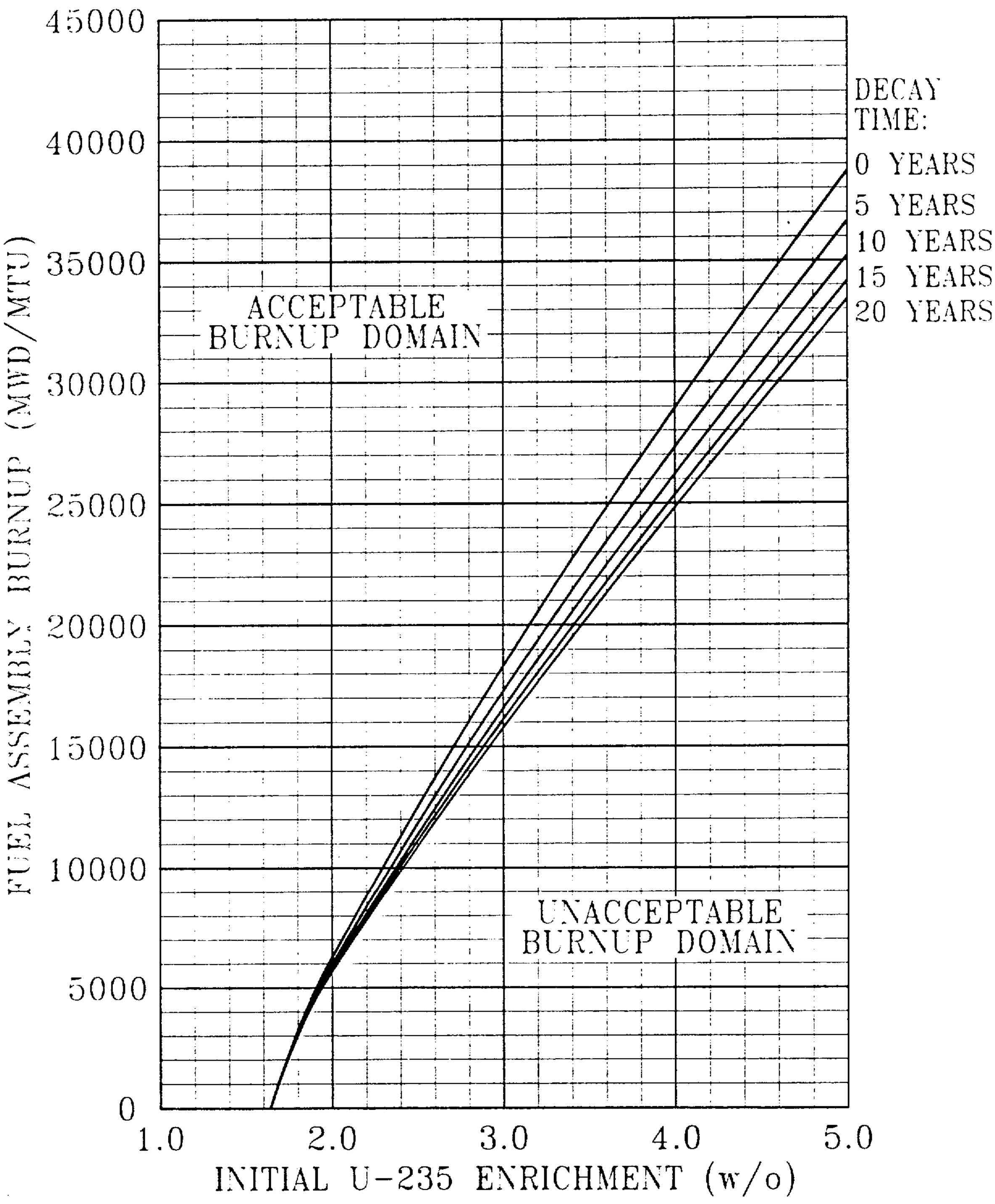

Figure 3.7.16-2 (page 1 of 1)
Region 2 3-out-of-4 Checkerboard Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

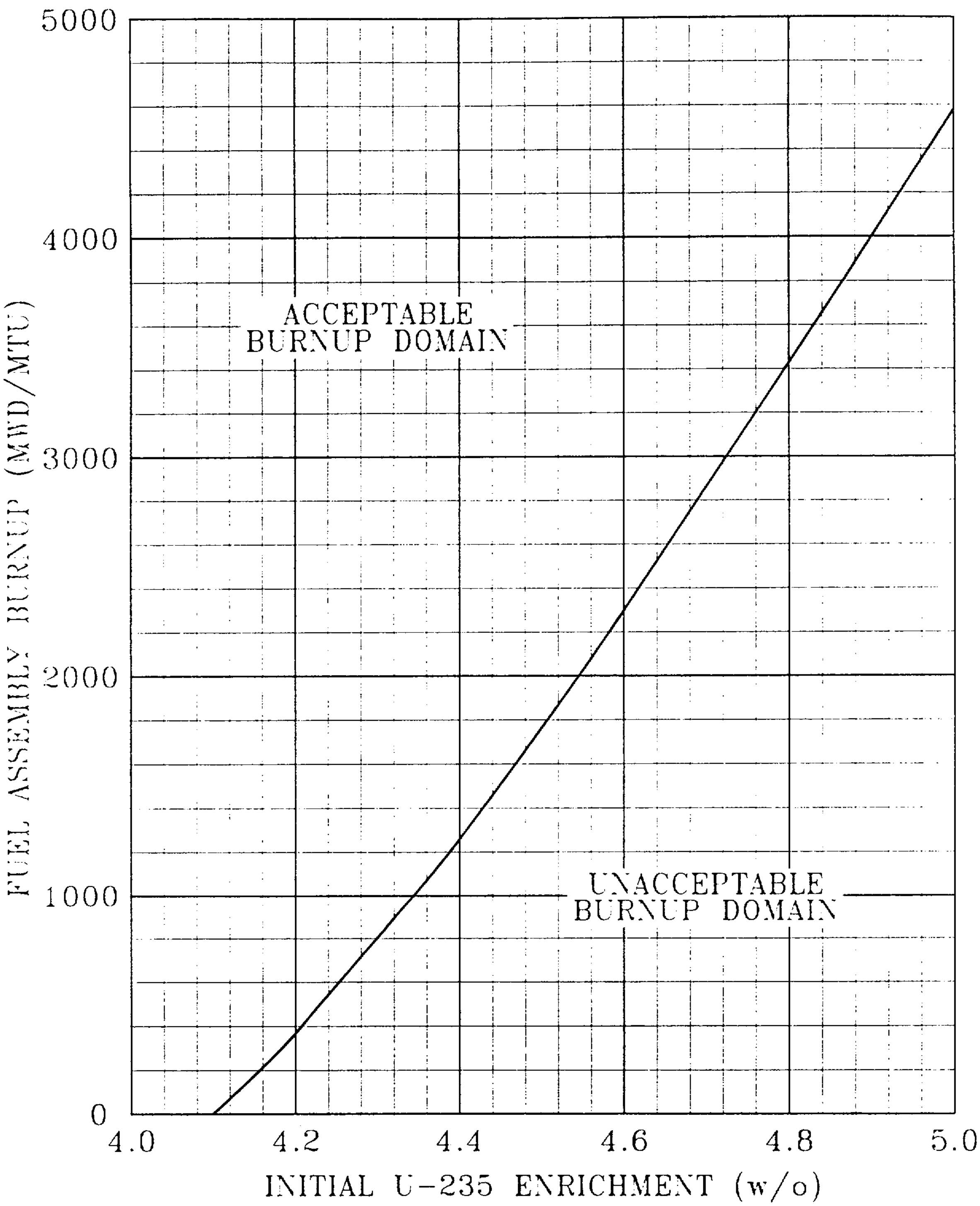

Figure 3.7.16-3 (page 1 of 1)
Region 2 2-out-of-4 Checkerboard Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

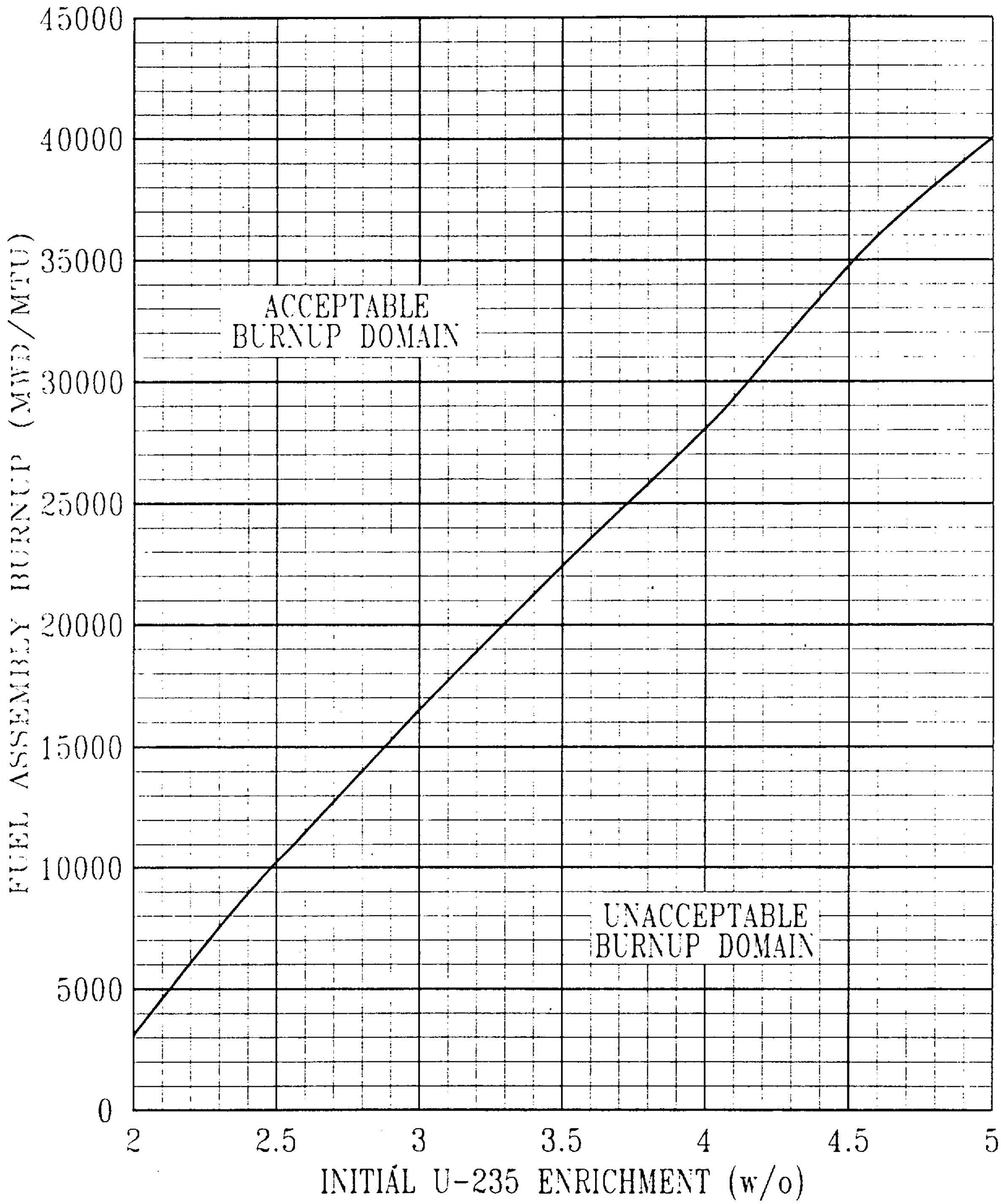

Figure 3.7.16-4 (page 1 of 1)
Region 2 Fuel Assembly Burnup Requirements
(Holtec Spent Fuel Pool Storage Racks)

DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained, as applicable, with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. For Joseph Oat spent fuel pool storage racks, $k_{eff} < 1.0$ if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology";

c. For Joseph Oat spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with water borated to 550 ppm, which includes an allowance for uncertainties as described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology";

d. For Joseph Oat spent fuel pool storage racks, a nominal 10.32 inch north-south and 10.42 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

e. For Joseph Oat spent fuel pool storage racks, a nominal 9.03 inch center to center distance between fuel assemblies placed in Region 2 racks.

f. For Holtec spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Holtec International Report HI-982094, "Criticality Analysis for Byron/Braidwood Rack Installation Project," Project No. 80944, 1998;

g. For Holtec spent fuel pool storage racks, a nominal 10.888 inch north-south and 10.574 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

h. For Holtec spent fuel pool storage racks, a nominal 8.97 inch center to center distance between fuel assemblies placed in Region 2 racks.

DESIGN FEATURES (continued)

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 410 ft, 0 inches.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 2984 fuel assemblies.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-72

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 23, 1999, as supplemented on October 21, 1999, and December 15, 1999, 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. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 105 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 1, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-77

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 23, 1999, as supplemented on October 21, 1999, and December 15, 1999, 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. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 105 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NOS. 105 AND 105

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3.7.15-1	3.7.15-1
-	3.7.15-2
3.7.16-1	3.7.16-1
3.7.16-2	3.7.16-2
3.7.16-3	3.7.16-3
3.7.16-4	3.7.16-4
3.7.16-5	3.7.16-5
-	3.7.16-6
-	3.7.16-7
4.0-2	4.0-2
-	4.0-3

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Boron Concentration

LCO 3.7.15 The spent fuel pool boron concentration shall be, as applicable:

a. ≥ 300 ppm for Holtec spent fuel pool storage racks; and

b. ≥ 2000 ppm for Joseph Oat spent fuel pool storage racks.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 Each spent fuel assembly stored in the spent fuel pool shall, as applicable:

a. Region 1 of Joseph Oat spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 4.7 weight percent U-235 or satisfy a minimum number of Integral Fuel Burnable Absorbers (IFBAs) for higher initial enrichments up to 5.0 weight percent U-235 to permit storage in any cell location.

b. Region 2 of Joseph Oat spent fuel pool storage racks

Have a combination of initial enrichment, burnup, and decay time within the Acceptable Burnup Domain of Figure 3.7.16-1, 3.7.16-2, or 3.7.16-3, as applicable for that storage configuration.

c. Interface Requirements for Joseph Oat spent fuel pool storage racks

Comply with the Interface Requirements within and between adjacent racks.

d. Region 1 of Holtec spent fuel pool storage racks

Have an initial nominal enrichment of ≤ 5.0 weight percent U-235 to permit storage in any cell location.

e. Region 2 of Holtec spent fuel pool storage racks

Have a combination of initial enrichment and burnup within the Acceptable Burnup Domain of Figure 3.7.16-4.

APPLICABILITY: Whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

-----------------------------------NOTE-----------------------------------

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly into a location which restores compliance.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	------------------NOTE------------------ Item a is only applicable for storage of fuel assemblies in Region 1 Holtec spent fuel pool storage racks. Item b is only applicable for storage of fuel assemblies in Region 1 Joseph Oat spent fuel pool storage racks. -- Verify by administrative means the following requirements are met: a. Initial nominal enrichment of the fuel assembly is ≤ 5.0 weight percent U-235. <u>AND</u>	Prior to storing the fuel assembly in Region 1 (continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	(continued) b. Initial nominal enrichment of the fuel assembly is ≤ 4.7 weight percent U-235 with less than the minimum number of IFBAs or ≤ 5.0 weight percent U-235 with the minimum number of IFBAs.	
SR 3.7.16.2	------------------NOTE-------------------- Figures 3.7.16-1, 3.7.16-2, and 3.7.16-3 are only applicable for storage of fuel assemblies in Region 2 Joseph Oat spent fuel pool storage racks. Figure 3.7.16-4 is only applicable for storage of fuel assemblies in Region 2 Holtec spent fuel pool storage racks. -- Verify by administrative means the combination of initial enrichment, burnup, and decay time, as applicable, of the fuel assembly is within the Acceptable Burnup Domain of Figure 3.7.16-1, 3.7.16-2, 3.7.16-3, or 3.7.16-4.	Prior to storing the fuel assembly in Region 2
SR 3.7.16.3	------------------NOTE-------------------- Only applicable for storage of fuel assemblies in Joseph Oat spent fuel pool storage racks. -- Verify by administrative means the interface requirements within and between adjacent racks are met.	Prior to storing the fuel assembly in the spent fuel pool

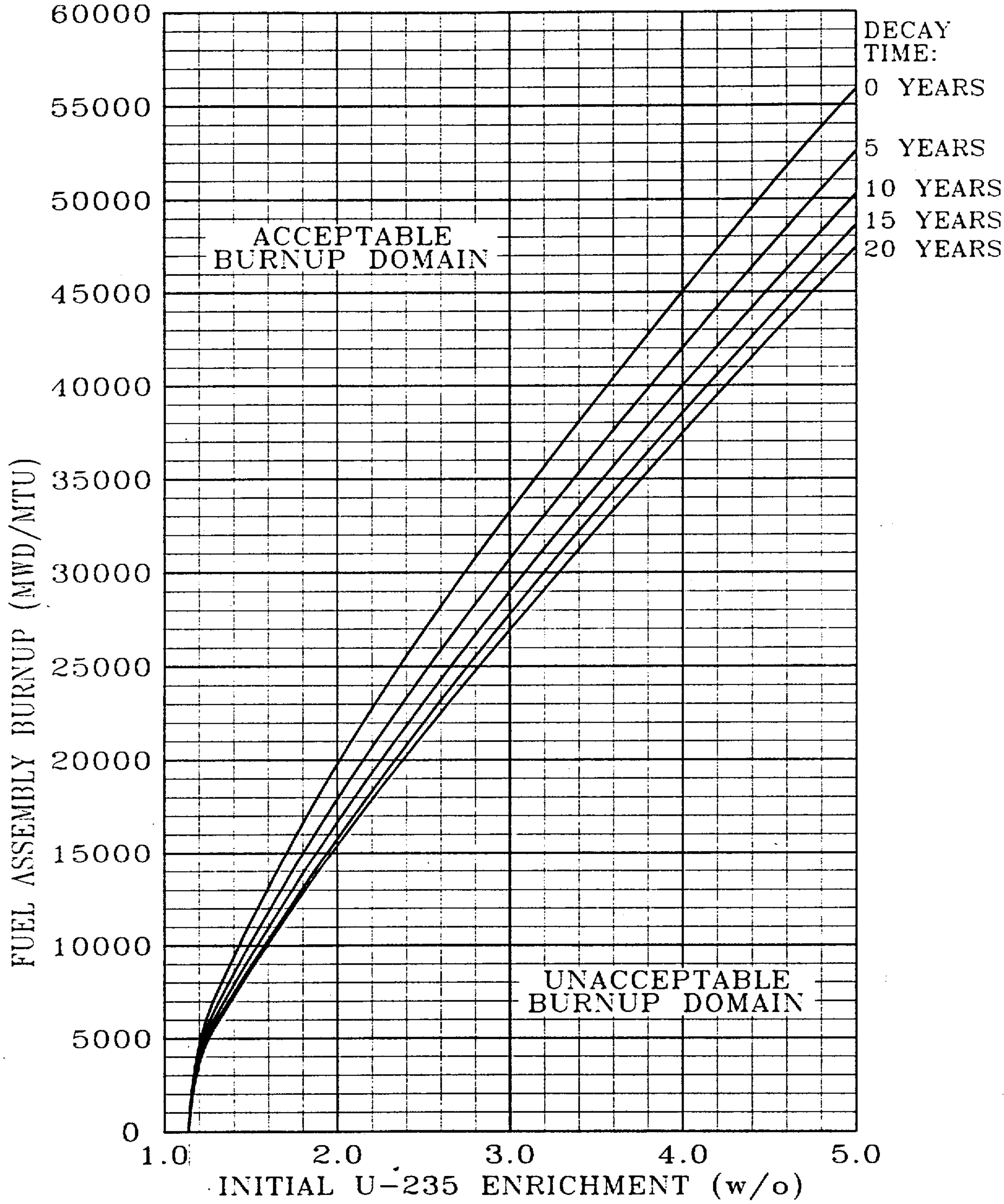

Figure 3.7.16-1 (page 1 of 1)
Region 2 All Cell Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

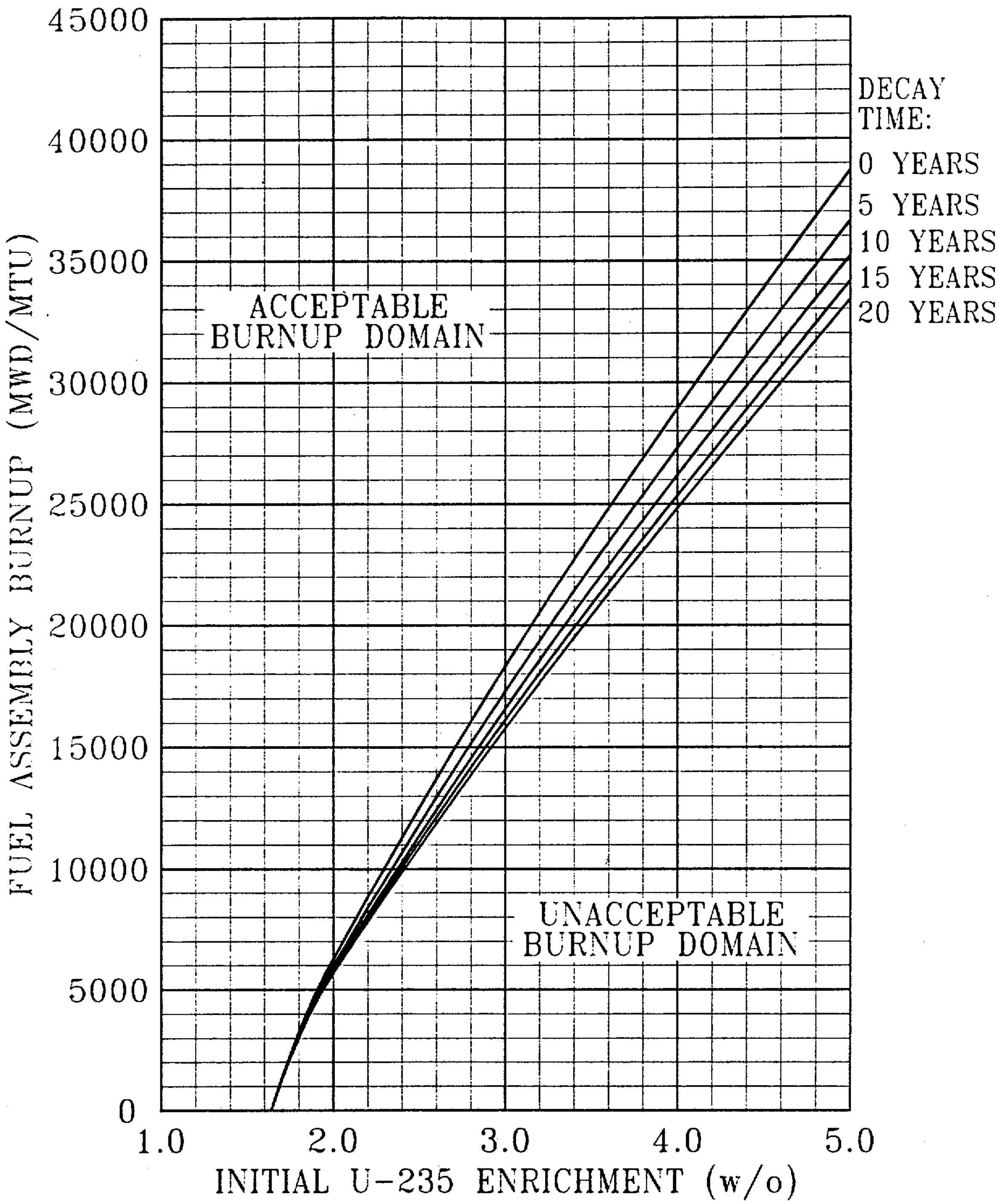

Figure 3.7.16-2 (page 1 of 1)
Region 2 3-out-of-4 Checkerboard Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

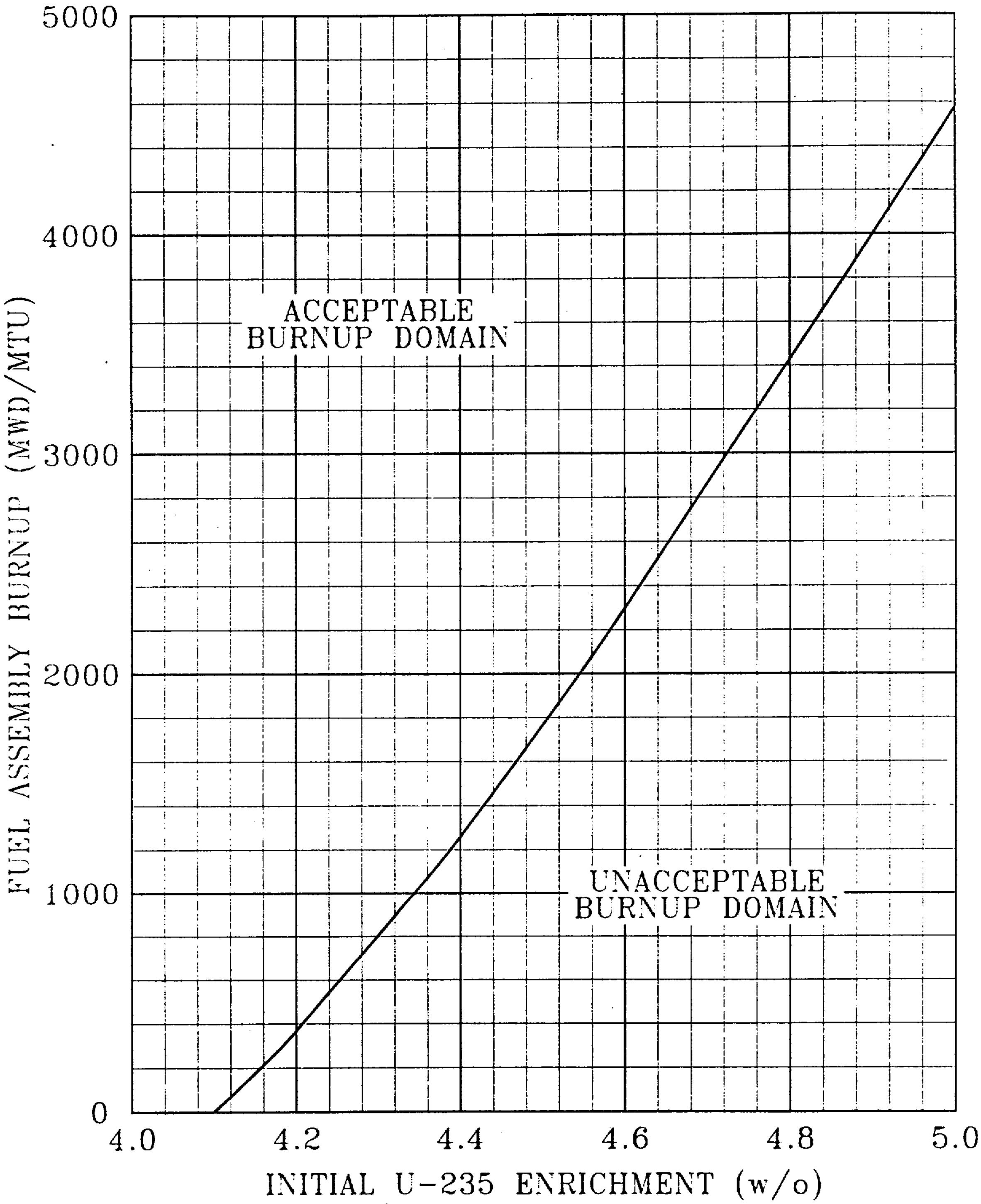

Figure 3.7.16-3 (page 1 of 1)
Region 2 2-out-of-4 Checkerboard Configuration Burnup Credit Requirements
(Joseph Oat Spent Fuel Pool Storage Racks)

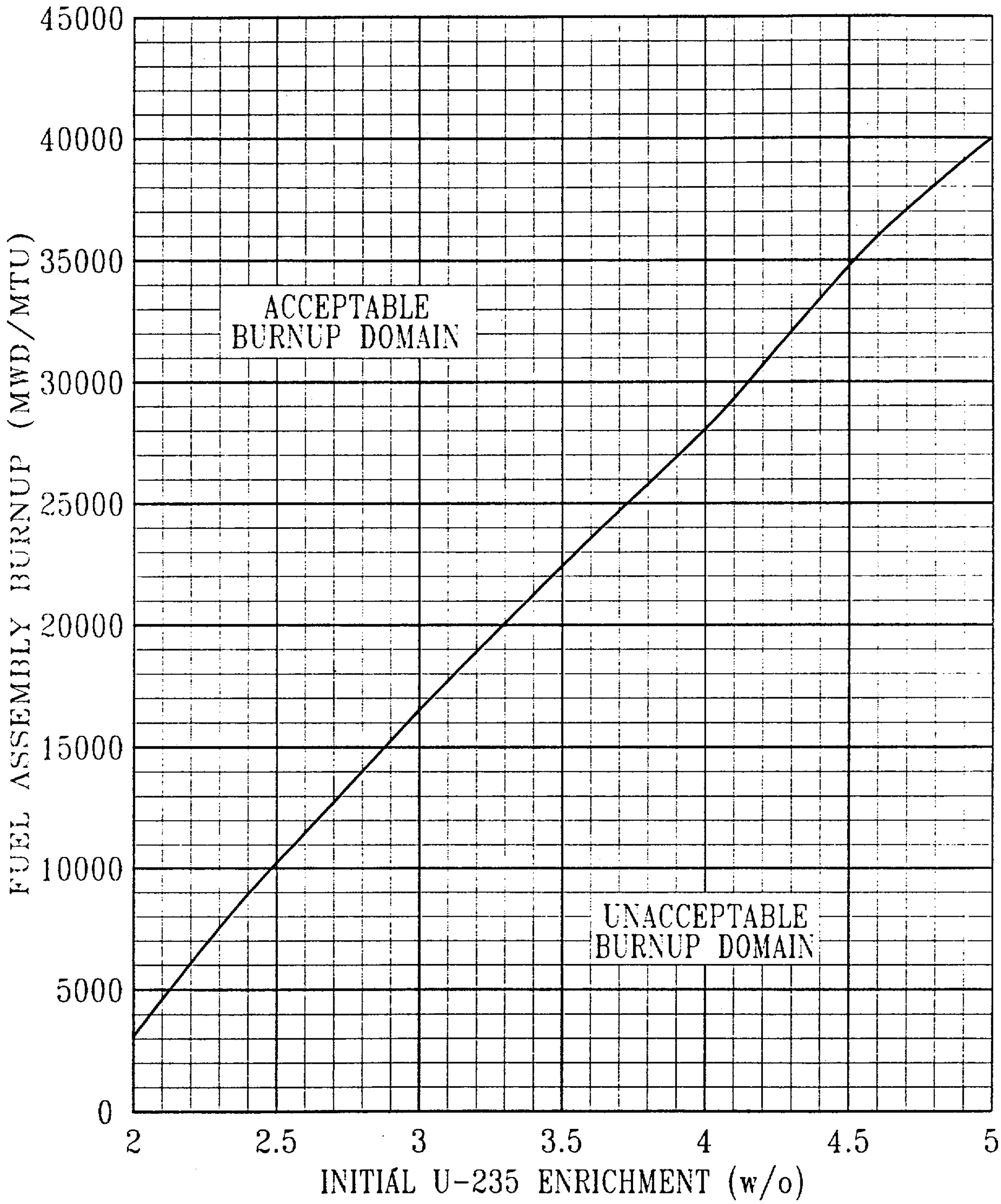

Figure 3.7.16-4 (page 1 of 1)
Region 2 Fuel Assembly Burnup Requirements
(Holtec Spent Fuel Pool Storage Racks)

DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained, as applicable, with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. For Joseph Oat spent fuel pool storage racks, $k_{eff} < 1.0$ if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology";

c. For Joseph Oat spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with water borated to 550 ppm, which includes an allowance for uncertainties as described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology";

d. For Joseph Oat spent fuel pool storage racks, a nominal 10.32 inch north-south and 10.42 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

e. For Joseph Oat spent fuel pool storage racks, a nominal 9.03 inch center to center distance between fuel assemblies placed in Region 2 racks.

f. For Holtec spent fuel pool storage racks, $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Holtec International Report HI-982094, "Criticality Analysis for Byron/Braidwood Rack Installation Project," Project No. 80944, 1998;

g. For Holtec spent fuel pool storage racks, a nominal 10.888 inch north-south and 10.574 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

h. For Holtec spent fuel pool storage racks, a nominal 8.97 inch center to center distance between fuel assemblies placed in Region 2 racks.

DESIGN FEATURES (continued)

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 410 ft, 0 inches.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 2984 fuel assemblies.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456, AND STN 50-457

1.0 INTRODUCTION

By letter dated March 23, 1999, Commonwealth Edison Company (ComEd, the licensee) requested license amendments to support the installation of new Boral high density spent fuel storage racks at Byron Station, Units 1 and 2 (Byron) and Braidwood Station, Units 1 and 2 (Braidwood) (Reference 1). The requested amendments proposed to change technical specification (TS) 3.7.15, "Spent Fuel Pool Boron Concentration"; TS 3.7.16, "Spent Fuel Assembly Storage"; TS 4.3.1, "Criticality"; and TS 4.3.3, "Capacity," to support installation of new Boral high-density spent fuel pool (SFP) storage racks at Byron and Braidwood. The change (rerack) will involve removing all 23 existing SFP storage racks at each station and replacing them with 24 new SFP storage racks. Installation of the new racks will increase the SFP capacity at each station from 2,870 assemblies to 2,984 assemblies.

A meeting was held on May 24, 1999, with the licensee to discuss the technical issues related to the proposed changes. A meeting summary was published on June 30, 1999 (Reference 2). Supplemental information that did not change the initial proposed no significant hazards consideration determination was provided in the licensee's letters of October 21, 1999 (Reference 3) and December 15, 1999 (Reference 4).

2.0 BACKGROUND

The Byron and Braidwood SFPs currently contain 23 storage racks which were manufactured by the Joseph Oat Company and which utilize Boraflex as the neutron absorber material. Degradation of Boraflex has caused water chemistry and clarity problems and has also resulted in the need to rely on soluble boron in the pool water to maintain the design basis (i.e., k_{eff} no greater than 0.95). The new storage racks are manufactured by Holtec International and utilize

Boral as the neutron absorber material. When installation of these new racks is complete, the Byron and Braidwood Stations spent fuel pools will each contain 24 racks which will increase the storage capacity at each plant from 2,870 assemblies to 2,984 assemblies.

The storage cells will be divided into two regions based upon rack type. Region 1 will contain four racks which use a "flux trap" design. In this design, the water gap between storage cells acts as a flux trap and provides an effective means of thermalizing neutrons so that they can be more readily absorbed by the Boral. The remaining 20 racks will not have flux traps and are referred to as Region 2.

3.0 EVALUATION

3.1 Criticality Considerations

3.1.1 Criticality Analysis Methods

The licensee's analysis of the reactivity effects of fuel storage in the Holtec racks to be installed in the Byron and Braidwood spent fuel pool was performed with the three-dimensional continuous energy MCNP4a Monte Carlo code. Independent verification calculations were performed with the three-dimensional discrete-energy NITAWL-KENO5a Monte Carlo code using the 238 group SCALE cross section library. Depletion analyses were made with the two-dimensional integral transport theory code, CASMO-4. CASMO-4 was also used for the determination of small reactivity increments due to manufacturing tolerances. 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 Byron and Braidwood spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and neutron absorber worth. A sufficient number of neutron histories (at least 1 million) were accumulated in each calculation to minimize the statistical uncertainty of the KENO5a calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Byron and Braidwood storage racks with a high degree of confidence.

3.1.2 Criticality Analysis Assumptions

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These assumptions included: (1) assuming unborated moderator at a temperature (4 degrees Celsius) that results in the highest reactivity, (2) neglecting radial neutron leakage (except in certain calculations where neutron leakage is inherent), (3) neglecting neutron absorption in minor structural members such as spacer grids, and (4) assuming the most conservative operating conditions for the depletion calculations. The design basis fuel assemblies include the Westinghouse Optimized Fuel Assembly (OFA), Vantage 5, and Vantage+ designs with a 17x17 fuel rod array containing UO_2 at a maximum initial enrichment of 5.0 weight percent (w/o) U-235.

Based on the above, the staff concludes that appropriately conservative assumptions were made.

3.1.3 Criterion for Criticality Prevention

General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50 states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. This requirement is met by conforming to the NRC acceptance criterion for criticality which states that the effective neutron multiplication factor (k_{eff}) in the spent fuel pool storage racks, if fully flooded by unborated water, shall be no greater than 0.95, including uncertainties at a 95/95 probability/confidence level.

For the Region 1 analysis, uncertainties due to the MCNP4a statistics and the MCNP4a method were statistically combined with mechanical tolerance uncertainties. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. A calculational bias derived from benchmark calculations was also included. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable. The licensee's analysis of the Region 1 storage racks showed that fuel with a nominal 5.0 w/o U-235 enrichment results in a maximum k_{eff} of 0.9422, including biases and 95/95 uncertainties.

For the Region 2 analysis, biases due to the calculational method and uncertainty in the depletion calculations were included. Uncertainties due to the MCNP4a statistics and the MCNP4a method, as well as burnup, where applicable, were statistically combined with mechanical tolerance uncertainties. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The concept of burnup reactivity equivalencing was used in order to store fuel with nominal enrichment up to 5.0 w/o U-235 in Region 2. This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for use in pressurized water reactor (PWR) fuel storage analysis. A series of reactivity calculations is performed to generate a set of enrichment versus burnup ordered pairs which yield an equivalent k_{eff} for fuel stored in the Byron and Braidwood Region 2 racks, as shown in TS Figure 3.7.16-4. The results of the burnup reactivity equivalencing shows that fuel with an initial U-235 enrichment of 5.0 w/o and irradiated to 40,000 megawatt days per metric ton uranium (MWD/MTU) results in a maximum k_{eff} of 0.9377 including biases and 95/95 uncertainties.

The results of these analyses, using the acceptable methods discussed above, meet the NRC criterion of k_{eff} no greater than 0.95, including all uncertainties at the 95/95 probability/ confidence level, and are, therefore, acceptable. The results show that Region 1 can accommodate fuel with enrichment as high as 5.0 w/o U-235 without restriction. Storage of fuel assemblies in Region 2 requires fuel burnup as a function of initial enrichment as given in TS Figure 3.7.16-4.

The new racks will not require interface restrictions because the minimum water gap between racks constitutes an effective neutron flux trap for the storage cells of neighboring Region 1 to Region 2 racks.

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 inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable areas in TS Figure 3.7.16-4, which could lead to an increase in reactivity. For such events, credit may be taken for the presence of at least 300 ppm of soluble boron in the pool required by TS 3.7.15 for the Holtec racks, 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. In fact, calculations show that for the most severe accident condition, a soluble boron concentration of 220 ppm boron would be adequate to maintain k_{eff} less than 0.95 for the Holtec racks.

3.1.4 Technical Specifications

Changes to TSs 3.7.15, 3.7.16, and 4.3 have been proposed as a result of the requested spent fuel pool reracking. Based on the above evaluation, the staff finds these changes acceptable as well as the associated Bases changes. During the installation of the new Holtec spent fuel storage racks, both Holtec racks and the existing Joseph Oat racks will be in the spent fuel pool at the same time. Therefore, the proposed TS changes address the requirements for both the new Holtec racks and the existing Joseph Oat racks. When shuffling fuel during the rack change-out, the current fuel assembly burnup, enrichment, and decay curve requirements applicable to the Joseph Oat racks, as well as the new burnup and enrichment curve requirements applicable to the Holtec racks, will be met. The soluble boron minimum concentration requirement of 2000 ppm currently required for the Joseph Oat racks will be maintained during the entire racks change-out process, thereby ensuring that k_{eff} will remain less than 0.95.

3.1.5 Summary

Based on the review described above, the staff finds that the criticality aspects of the proposed modifications to the Byron and Braidwood spent fuel pool storage racks are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling.

3.2 Spent Fuel Pool Cooling

The SFP cooling system removes decay heat from fuel stored in the SFP through the heat exchanger to the component cooling (CC) system. The essential service water system, in turn, removes heat from the CC water system. The SFP cooling system consists of two complete cooling trains and is a Safety Category I system that takes suction from the refueling water storage tanks and injects borated water into the SFP through the refueling water purification system. This is a Category I water makeup system.

Each cooling train includes one heat exchanger and pump, one purification loop with demineralizer and filter, and associated piping, valves, and flow indication instruments. Each cooling train is designed to maintain the bulk fluid temperature of the SFP below 140 degrees Fahrenheit for a normal one third of the reactor core discharge during refueling operations. Two additional sources of makeup water are provided to cool the SFP: (1) a backup Safety Category I makeup system, which takes suction from the Safety Category I fire protection system and injects water into the SFP, and (2) a non-Category I primary water makeup system, which takes suction from both primary water storage tanks and routes non-borated water through the SFP water filter, and the return header.

3.2.1 Decay Heat Load

The licensee performed decay heat load calculations in accordance with the provisions of NRC methodology Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Revision 2 (July 1981), to determine the maximum bulk pool temperature, to determine the time to boil after a loss of decay heat cooling for different fuel discharge conditions to ensure that SFP makeup is available, and to ensure that adequate time exists for corrective action.

To determine the bounding case for maximum decay heat evaluation, the licensee conservatively assumed the following:

- A full SFP condition, in which a total of 2,864 and 2,821 fuel assemblies would be accumulated from previous discharges in the Byron and Braidwood Station SFPs, respectively. Additional fresh full-core discharge of 193 fuel assemblies is added to increase the maximum fuel inventory to 3,057 and 3,014 fuel assemblies, respectively.

- An 18 months fuel cycle is used for both Byron and Braidwood stations.

- The building housing the SFP is assumed to have the maximum ambient air temperature of 104 degrees Fahrenheit and an increase in relative humidity to 100 percent which results in a conservative evaporative heat loss.

- The thermal heat exchanger performance of the SFP cooling system is assumed to be fouled to its design basis level to minimize heat rejection capacity.

- The CC water temperature is assumed to be at 105 degrees Fahrenheit.

- Thermal inertia of the SFP is limited to the quantity of water in the pool.

Three discharge scenarios were considered for bulk pool thermal-hydraulic evaluation: (1) a normal discharge of one-third of the reactor core (84 fuel assemblies) 100 hours after a reactor shutdown, (2) a normal full-core discharge (193 fuel assemblies) 100 hours after a reactor shutdown, and (3) an abnormal back-to-back discharge scenario in which a normal discharge is followed by a full-core discharge 17 days later, 100 hours after a reactor shutdown.

3.2.2 Normal Discharge Scenarios

Refueling operations at Byron and Braidwood are routinely performed with either an approximate one-third core offload or a full-core temporary offload in which approximately two-thirds of the fuel assemblies are returned to the reactor vessel, along with the new fuel upon the completion of the refueling operations. The Byron and Braidwood Updated Final Safety Analysis Reports (UFSARs) indicate that with the Joseph Oat rack configuration, the bulk pool temperature is maintained below 138 degrees Fahrenheit for the one-third core discharge and below 155 degrees Fahrenheit for the full-core discharge. The maximum heat load is approximately 35.6 x 10^6 Btu/hr and 56.5 x 10^6 Btu/hr, respectively.

The licensee evaluated these scenarios for the new Holtec rack configuration, which indicate that the bulk pool temperature is 138.32 degrees Fahrenheit for the one-third core discharge, and 157.13 degrees Fahrenheit for the full-core discharge, with a single train of cooling system in operation. The maximum heat load is 35.99 x 10^6 Btu/hr and 57.15 x 10^6 Btu/hr, respectively. The installation of new Holtec racks results in an increase of approximately 4 percent of the total fuel assemblies to be stored in the pool. As a consequence, the pool temperature for the full-core discharge increases to less than 1.4 percent above the 155 degrees Fahrenheit, and remains above 150 degrees Fahrenheit for less than 30 hours. The plant long-term cooling capability to maintain SFP temperature below 150 degrees Fahrenheit is provided by redundant safety-related SFP cooling systems, and the SFP temperature is monitored by operator rounds to assure that pool temperature is between 70 degrees Fahrenheit and 110 degrees Fahrenheit. The high temperature alarm is also annunciated in the main control room when pool temperature reached 149 degrees Fahrenheit.

Based on the review, the staff concludes that the new bulk pool temperature for the one-third core offload utilizing Holtec rack configuration remains similar to that of the previously licensed bulk pool temperature utilizing the Joseph Oat racks. The new bulk pool temperature increases less than 1.4 percent from the previously approved temperature of 155 degrees Fahrenheit for the full-core offload event. This is relatively minor and does not require modification of the current SFP cooling system design basis. Therefore, these changes are acceptable.

3.2.3 Abnormal Discharge Scenario

An abnormal back-to-back discharge scenario in which a normal discharge is followed by a full-core discharge 17 days after a full operation was also considered. The licensee performed an evaluation of this scenario utilizing a Holtec rack configuration, with one train of cooling system in operation to ensure that the bulk pool temperature is maintained below the boiling temperature of 212 degrees Fahrenheit. Standard Review Plan (SRP) (NUREG-0800), Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," indicates that for an abnormal discharge event, bulk pool boiling should not occur with two trains of cooling system in operation.

The licensee's evaluation indicates that peak bulk pool temperature reaches 166.50 degrees Fahrenheit, with the maximum decay heat load of 68.28 x 10^6 Btu/hr. The new pool temperature is still below the boiling temperature of 212 degrees Fahrenheit, with one train of cooling system in operation. The licensee stated that the pool temperature will be limited to 137.2 degrees Fahrenheit when both trains of cooling system are in operation.

On the basis of the review, the staff concludes that the licensee's calculation for an abnormal discharge scenario, with one train of cooling system in operation, is conservative, and the peak bulk pool temperature is still below the boiling temperature. Therefore, the change is acceptable.

3.2.4 Effects of Spent Fuel Pool Boiling

In the event that all forced SFP cooling becomes unavailable, the SFP water temperature will rise and eventually reach the normal bulk boiling temperature of 212 degrees Fahrenheit. The licensee determined that the minimum time to reach boiling is 8.43 hours and 3.82 hours for the normal one-third core discharge and the full-core discharge, respectively. Additionally, for an abnormal back-to-back discharge case, the minimum time to reach boiling is 2.63 hours. The pool inventory losses for the normal one-third core discharge, the full-core discharge, and the abnormal back-to-back discharge are computed to be 74 gpm, 118 gpm, and 141 gpm, respectively.

SFP makeup to match or exceed that of evaporative losses can be provided by the two 100 percent Safety Category I refueling water purification pumps that take suction from the two 450,000-gallon refueling water storage tanks, and inject borated water into the SFP with injection rates between 150 gpm and 250 gpm. Other makeup capabilities can be provided by the two Safety Category I fire protection systems with an injection rate of approximately 125 gpm each, and a non-Category I primary water makeup system, which takes suction from the two 500,000 gallon primary water storage tanks and routes non-borated water through the spent fuel pool water filter, and the return header with an injection rate of 120 gpm.

Additionally, various alarm indications associated with a loss of flow to the SFP cooling pump, a high pool temperature, and the SFP level are annunciated in the control room and direct the operator to appropriate response procedures to provide makeup water to the pool. The staff, therefore, concludes that makeup capabilities are adequately provided to mitigate the loss of SFP cooling events, and the operators would have adequate time to respond to the event before boiling could occur.

3.3 Materials Compatibility

The new maximum storage rack arrays are free-standing, self-supporting racks designed to stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

3.3.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-304L for all sheet metal stock and internally threaded support legs, ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100 degrees Fahrenheit) for externally threaded support spindle, and ASME Type 308L for weld material.

These materials used in the Holtec racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the spent fuel pool environment and are acceptable for use in this application.

3.3.2 Neutron Absorbing Material

The Holtec racks employ Boral as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the spent fuel pool environments where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed to vent the corrosion gases. The neutron absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

3.3.3 Summary

Based on its evaluation, the staff finds the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec International are compatible with the spent fuel pool environment at the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2. The type of degradation exhibited by the racks does not affect their neutron absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

3.4 Structural Evaluation

3.4.1 Storage Racks

The SFP storage racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee's contractor, Holtec International, performed the design, fabrication, and safety analysis of the new high density SFP storage racks. The design of these new racks at Byron and Braidwood Stations is similar to Holtec racks that have been reviewed and approved by the NRC and are presently in service at other nuclear power plants including Salem, Watts Bar, and Vogtle (Reference1). The proposed racks are freestanding and self-supporting equipment. They are not attached to the floor of the storage pool. At the time of the previous reracking, the seismic evaluation of the racks was performed using single rack (SR) three dimensional (3-D) simulations. However, for the current reracking, both SR and whole pool multi rack (WPMR) analyses were performed to simulate the dynamic behavior of the high density rack structures (Reference 1). The worst-case loads and stresses that result from both these analyses were used to determine the structural adequacy of the racks. Holtec used a computer program, DYNARACK, for the dynamic analysis to demonstrate the structural adequacy of the spent fuel rack design under the earthquake loading conditions.

The SFP structures at Byron and Braidwood Stations are identical with two exceptions. Specifically, the exceptions are the subsurface foundation material below the concrete basemats, and the seismic ground motion due to SSE. The Byron Station SFP basemat is founded directly on rock outcrop, while there is an intervening soil layer below the basemat at

Braidwood Station. For the previous reracking at Byron and Braidwood Stations approved in 1989 (which increased the SFP capacity from 1060 fuel assemblies to 2870 assemblies), the licensee used the envelope of the Byron spectra and Braidwood spectra obtained from the building seismic models at the fuel pool elevation as the design response spectra to develop the synthetic time histories for the design of the racks. However, for the current re-racking analysis at Byron and Braidwood Stations, the licensee had not clearly stated in Reference 1 that the same procedure was used to arrive at the target design spectra as seismic input for the dynamic analysis using the DYNARACK program. The information regarding the target design spectra was confirmed when the licensee submitted the additional information requested at the meeting held on May 24, 1999 (Reference 3). The synthetic time histories needed for the dynamic analyses in accordance with Section 3.7.1 of the SRP which calls for both the response spectrum and the power spectral density (PSD) corresponding to the generated acceleration time-history to envelope the target (design basis) counterparts were generated by the licensee.

The lateral motion of the rack due to seismic motion is resisted by the pedestal-to-pool slab interface friction, and is amplified or retarded by the fluid coupling forces produced by the close position of the rack to other structures. The construction of a 3-D single rack dynamic model for performing a seismic analysis consists of modeling the rack as a multi-degree-of-freedom (multi-DOF) structure in such a manner that the selected DOFs capture all macro-motion modes of the rack such as twisting, overturning, lift off, sliding, flexing, and combinations thereof. During a seismic event, the subject rack and the neighboring rack will both undergo dynamic motions which will be governed by the interaction among the inertia, fluid, friction, and rattling forces for each rack. In a single rack analysis, it is not possible to accurately model the hydrodynamic forces due to fluid coupling between two adjacent racks, because they depend on relative motions between the two racks (which are unknown in a single rack analysis). This limitation of not knowing the motion of the neighboring rack (to determine the relative motion) in a single rack analysis is overcome by using an artificial boundary condition referred to as the "out-of-phase" assumption. However, the licensee concluded that this method of accounting for the fluid coupling effects could not be shown to be conservative for closely spaced racks. Therefore, by building on the SR model, ComEd performed the WPMR analysis by simultaneously modeling all racks and accounting for the multi-body fluid coupling effects, as discussed in detail in Section 6.6.2 of Reference 1.

A nonlinear analytical model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the DYNARACK program, were used to simulate the three-dimensional dynamic behavior of the rack and the fuel assemblies including the frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration time histories) was generated from the design response spectra defined by the envelope of the Byron floor response spectra (FRS) and the Braidwood FRS obtained from the building seismic models at the fuel pool elevation. This is deemed to be conservative because the Byron SFP structure is founded on rock while the Braidwood SFP structure is resting on soil (References 3 and 4).

The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target PSD function compatible with the design response spectra as discussed in SRP Section 3.7.1.

Using the results of the DYNARACK analysis, ComEd performed the structural evaluation of spent fuel rack design, using the criteria given in NRC's, "O. T. Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications," dated April 14, 1978, which specifies the maximum required safety factor against rack overturning to be 1.5 and 1.1 under the operating basis earthquake (OBE) and SSE events, respectively (Reference 1, Section 6.7.1). The licensee performed the stress limit evaluation of the rack structure using the ASME Code, Section III, Subsection NF, for normal and upset conditions (Level A or Level B), and Section F-1334 (ASME Section III, Appendix F) for Level D condition. ComEd considered the applicable loads and their combinations in the seismic analysis of the rack modules, and performed parametric simulations for both the SR and WPMR analyses. The parameters varied in the different computer runs consisted of the rack/pool interface coefficient of friction, extent of storage locations occupied by spent fuel (ranging from nearly empty to full) and the type of seismic input (SSE or OBE).

For the parametric simulations, ComEd performed a total of twenty-five 3-D single rack model analyses and six WPMR model analyses (Reference 1, Section 6.8). The results of these analyses shown in Section 6.9.1 of Reference 1 indicate that the maximum rack displacement of 1.5 inches (for SSE condition) occurred at the top of the rack. Using this value, ComEd performed a rack overturning evaluation and found the factor of safety against overturning to be 61 which is much higher than the prescribed limit of 1.1 for SSE condition. ComEd also found that the maximum rack displacement at the baseplate elevation was 0.5975 inches. These results indicate that there are large safety margins against overturning of the racks as evidenced by the small rack movements and, consequently, the structural integrity and stability of the racks and fuel assemblies are maintained.

From the large number of computer runs of parametric evaluations, the licensee computed the maximum values of pedestal vertical and lateral loads, shear forces, displacements and stress factors and provided them in tabular form in Reference 1. The results show that: (1) all stresses are well below the corresponding "NF" limits, (2) there are no rack-to-wall or rack-to-rack impacts anywhere in the cellular region of the rack modules during SSE, and (3) the factor of safety against overturning is more than 60.

The licensee calculated the weld stresses of the rack at the connections (e.g., baseplate-to-cell, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions and demonstrated that all the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, Section III, Subsection NF, indicating that the weld connection design of the rack is adequate.

Based on: (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) the large factor of safety of the induced stresses of the rack when they are compared to the corresponding allowables provided in the ASME Section III, (3) a reasonable assurance that there is no rack-to-wall and rack-to-rack impacts, and (4) the licensee's overall structural integrity conclusions supported by both single rack analyses and whole-pool multi-rack analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.4.2 Spent Fuel Storage Pool

The Byron/Braidwood SFP structures, both built from monolithic reinforced concrete, are identical in dimensions and are designed as seismic Category I structures. The overall dimensions of each pool structure (including the walls and the basemat) are approximately 44.6 feet in width, 72 feet in length and 47 feet in depth; the reinforced concrete walls range from 5 feet to 6 feet in thickness while the thickness of the basemat is 6 feet (Reference 1, Section 8.2). The internal surface of the pool structure is lined with stainless steel plates to ensure water tight integrity. However, the pool liner is not a safety-related component, as its failure would not cause a rapid lowering of the water level in the pool (Reference 1, Section 8.6).

The pool structure was analyzed by using the finite element computer program, SLSAP (which is Sargent and Lundy's controlled version of the public domain code SAP), to demonstrate the adequacy of the pool structure under fully-loaded fuel racks with all storage locations occupied by fuel assemblies. The fully-loaded pool structure was subjected to the load combinations specified in the American Concrete Institute Code, ACI 318-71.

Table 8.5.3 of Reference 1 shows the predicted minimum factors of safety for the reinforced concrete ranging from 1.61 to 4.75 for bending moments of the concrete walls and slab (Reference 1, Section 8.5). In view of the calculated factors of safety, the staff concludes that the licensee's pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, and SSE loading conditions. Thus, the SFP design is acceptable.

3.4.3 Fuel Handling Accident

The following refueling accident cases were evaluated by the licensee: (1) two cases for the drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario), and (2) one case for the drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The impact region of one of the two deep drop events is located above the support pedestal. The maximum stresses produced by this impact in the liner and the concrete slab are well below the failure limits, thus, resulting in no damage to the SFP liner and the concrete slab. The second deep drop condition through an interior cell produces some deformation of the baseplate and localized severing of the baseplate/cell wall welds. However, the licensee determined that this drop event lowers the fuel assembly support surface by a maximum of 1.25 inches which is much less than the distance of 14 inches from the baseplate to the liner (Reference 1, Section 7.5.2) and, therefore, the licensee concluded that the pool liner will not be damaged.

The shallow drop event produces severe localized plastic deformation in the top of the storage cell, but the region of this plastic deformation is limited to 16 inches, which is below the design limit of 17 inches (Reference 1, Section 7.5.1). The staff reviewed the licensee's fuel drop analysis results in Reference 1 and concurs with its findings.

3.4.4 Summary

Based on the information provided by the licensee, the staff concludes that the structural analyses of the spent fuel storage rack modules and SFP are in compliance with the acceptance criteria specified in the Updated Final Safety Analysis Report (UFSAR) and current licensing practices.

3.5 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the modification of the Braidwood and Byron spent fuel racks with respect to occupational radiation exposure. As stated previously (Section 2.0), the licensee is planning to replace all 23 of the existing spent fuel pool storage racks at each station with 24 new spent fuel pool storage racks. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates the total occupational exposure for the reracking operation to be between 6 to 12 person-rem.

All of the operations involved in reracking will utilize detailed procedures prepared with full consideration of as low as reasonably achievable (ALARA) principles.

The Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP reracking operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels (including potential exposure to hot particles), and dosimetry requirements. Each member of the project team will attend an ALARA pre-plan meeting and each team member will be required to attend daily pre-job briefings on the scope of the work to be performed. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment including alarming dosimeters.

The licensee does not plan to use divers for this project. However, if it becomes necessary to utilize divers to remove any interferences which may impede the installation of the new spent fuel racks, the licensee will equip each diver with the appropriate monitoring equipment. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposure is maintained ALARA.

Based on review of the licensee's proposal, the staff concludes that the Braidwood and Byron spent fuel storage capacities can be increased in a manner that will ensure that doses to workers will be maintained ALARA. Therefore, the staff finds licensee's conclusion acceptable.

3.6 Solid Radioactive Waste

Spent fuel pool filtration resin replacement is determined primarily by the requirement for water clarity, and the resin is normally expected to be changed about once a year. Although there may be an increase in resin replacement during the SFP rack change-out, the licensee does not expect the resin change-out frequency of the SFP purification system to be increased as a result of the expanded storage capacity. Overall, the licensee does not expect that the additional fuel storage made available by the increase storage capacity will result in a significant change in the generation of solid radioactive waste. The staff finds the licensee's conclusion to be acceptable.

3.7 Accident Dose Evaluation

The licensee evaluated five spent fuel drop accidents, a spent fuel cask drop accident, and a change in the SFP water temperature. Because of the similarity between the new racks and the existing ones, and the small increase (4 percent) in the spent fuel capacity of the new racks, the consequences of the spent fuel and fuel cask drop accidents were either bounded by the previous accident analyses as incorporated in the plants' design bases or unaffected by the changeout of the SFP racks.

The change in temperature of the SFP water was evaluated for the potential increase in reactivity. Because the reactivity coefficient in the SFP is negative, a temperature increase will result in a decrease in reactivity. The initiators of this event are unaffected by the SFP rack replacement because there are no features of the design change affecting the SFP cooling system or that would prompt an SFP water temperature decrease.

As a consequence of the analyses, the licensee concluded and the staff agrees that increases in the capacity of the SFPs at Byron and Braidwood will not be accompanied by an associated increase in the radiological consequences of fuel-handling accidents. The potential offsite doses will not be increased over the values given in the UFSAR.

3.8 Heavy Loads Handling Evaluation

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines for licensees to assure safe handling of heavy loads by prohibiting load travel, to the extent practicable, over spent fuel assemblies, over the core, and over safety-related equipment. It also defines a heavy load as the combined weight of a single fuel assembly and its handling tool. Furthermore, it recommends in Section 5.1.1 that procedures be developed to cover load handling operations for heavy loads that could be handled over or in proximity to irradiated fuel.

The licensee's current heavy loads specification limits a heavy load to loads that exceed 2,000 pounds. Heavy loads are restricted from travel over spent fuel assemblies stored in the SFP. No changes to these specifications are proposed. The licensee proposes to use defense-in-depth guidelines as provided in NUREG-0612 to assure that heavy loads moved during the rerack operation, including spent fuel storage racks, are not moved over fuel in the SFP.

3.8.1 Hoisting System

The licensee stated that rack removal and installation will be performed in accordance with NUREG-0612 and ANSI N14.6 -1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." The rack change-out will be performed using the 125-ton Fuel Building double girder bridge crane. The licensee stated that the crane is designed in accordance with requirements of the Crane Manufacturers Association of America (CMAA), "Specification No. 70 for Electric Overhead Traveling Cranes," and American National Standard Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)." The maximum "impact weight" used by the licensee for a rack drop is 2300 lbs. which includes the rack, lift rig, rigging and the temporary hoist. Therefore, the crane capacity for the lift provides a large factor of safety.

A temporary hoist (lifting device) will be attached to the overhead crane to avoid submerging and contaminating the crane hook in the water in the SFP. The lifting rig will be remotely engaged and interposed between the crane hook and the rack and is specifically designed to lift the new spent fuel rack modules. It is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 -1978. It consists of four independently loaded lift rods and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. As stated by the licensee, both the stress design and the load testing of the lifting rig satisfies guidelines in Section 5.1.6(1) of NUREG-0612 and ANSI N14.6-1978, respectively. Accordingly, the lift rods are designed as follows: (1) with the appropriate stress design factor as specified in ANSI N14.6 (safety factor of 5 to 1), (2) load tested to 300 percent of the maximum weight to be lifted, and (3) after load testing, the integrity of the critical weld joints will be examined using a liquid penetrant. Non-customized lifting devices (i.e., slings) will be used in accordance with NUREG-0612 and ANSI B30.9-1971, "Slings." Therefore, the slings must be proof tested at a minimum of 1.5 times their rated capacity in accordance with Section 9.3.3 in ANSI B30.9.

The staff concludes that the crane coupled with the design and testing of the lifting rig and other lifting devices will enable the licensee to handle heavy loads with little to no risks to the safety of the rerack operation.

3.8.2 Load Paths and Heavy Loads Handling Accident Analysis

The licensee stated that all handling, installation and removal of the spent fuel storage racks will be performed in accordance with NUREG-0612 guidelines. Safe load paths will be developed for moving the racks into and out of the fuel building. Spent fuel in the pool will be shuffled into racks that are not in the travel path of a rack to be moved. Therefore, the new racks and lifting rig will not be carried over or near active fuel. Furthermore, the racks will be lifted such that the center of gravity of the lift points will be aligned with the center of gravity of the load. In addition, mechanical stops for the crane will be installed temporarily to prevent any crane travel or load movement over fuel. Also, the crane and bridge operators are to be trained in accordance with ANSI B30.2-1996 and plant specific training.

The licensee considered load handling accidents involving "shallow" and "deep" drops of a spent fuel assembly and its handling tool during the rerack operation. The shallow drop is a vertical drop to the top of the rack from a lift height of 36 inches. The deep drop is a straight vertical drop onto the baseplate of the rack module. Neither accident scenario would result in any damage to the spent fuel pool liner. As a result, a loss of inventory in the SFP would not occur due to the drop of a spent fuel pool storage rack. The licensee analyzed the potential for a cask drop accident and found that when the cask is moved by the crane, the crane and cask travel will not occur over the SFP because crane interlocks will limit the crane travel. Consequently, the probability and consequences of a cask drop are unaffected by the replacement of the existing racks.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: (1) operator errors, (2) rigging failures, (3) lack of adequate inspection, and (4) inadequate procedures. The licensee plans to implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. They will: (1) provide comprehensive training to the rerack installation crew, (2) use redundantly designed lifting rigs, (3) perform inspection and maintenance checks on the cranes and lifting devices prior to the rerack operation, and (4) use specific procedures that cover the entire rerack effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement.

The staff accepts the licensee's finding that SFP integrity would not be breached if a rack drop was to occur. Also, the staff agrees with the licensee that the use of the crane in conjunction with administrative procedures and controls focused on, but not limited to, the areas noted above will enable the licensee to maintain safety during the rerack operation.

3.8.3 Summary

Based on the evaluation of the licensee's submittal, the staff accepts the use of administrative controls in accordance with NUREG-0612 to improve the removal and installation of the racks in the SFP. The measures to be implemented will enable the licensee to move the racks while preventing any damage to spent fuel and the SFP structure if a crane failure or load drop was to occur. Therefore, the proposed considerations for moving heavy loads are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has previously been prepared and published in the Federal Register on February 29, 2000 (65 FR 10841).

Accordingly, based on the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect upon the quality of the human environment.

6.0 CONCLUSION

The Commission 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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7.0 REFERENCES

1. Letter, Commonwealth Edison Company to USNRC, "Request for an Amendment to Technical Specification to Support Installation of New Spent Fuel Pool Storage Racks at Byron and Braidwood Stations," dated March 23, 1999.

2. Memorandum, Stewart Bailey, NRC, "Summary of May 24, 1999, Meeting with Commonwealth Edison Company Regarding Re-rack of Spent Fuel Pools at Braidwood and Byron Stations," dated June 30, 1999.

3. Letter, Commonwealth Edison Company to USNRC, "Response to Request for Additional and Clarifying Information Regarding Holtec International Report HI-9820 83, Licensing Report for Spent Fuel Rack Installation at Byron and Braidwood Nuclear Stations," dated October 21, 1999.

4. Letter, Commonwealth Edison Company to USNRC, "Editorial Correction to Technical Specification Amendment Request to Support Installation of New Spent Fuel Pool Storage Racks at Byron and Braidwood Stations," dated December 15, 1999.

5. Appendix A to Letter from Leonard N. Olshan, NRC, to Henry E. Bliss, "Increase in the Spent Fuel Pool Capacity Through the Use of High Density Storage Racks, Byron Station, Units 1 and 2," dated March 17, 1989.