December 4, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS - BYRON AND BRAIDWOOD STATIONS (TAC NOS. M99170, M99171, M99168 AND M99169)

Dear Ms. Johnson:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 94 to Facility Operating License No. NPF-37 and Amendment No. 94 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 86 to Facility Operating License No. NPF-72 and Amendment No. 86 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 June 30, 1997, as supplemented on September 25, 1997.

The amendments grant partial credit for boron in the spent fuel pools to maintain the subcriticality.

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,

Original Signed By:

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 94 to NPF-37
2. Amendment No. 94 to NPF-66
3. Amendment No. 86 to NPF-72
4. Amendment No. 86 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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*concurrence provided by memo dated 9/30/97; no major revisions

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

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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. 94
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 June 30, 1997, as supplemented on September 25, 1997, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No.94 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

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 4, 1997

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. 94
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 June 30, 1997, as supplemented on September 25, 1997, 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 1;

   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.94 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

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 4, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 94 AND 94

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. Pages indicated by an asterisk are provided for convenience only.

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XVII	XVII
XVIII	XVIII
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The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation and injecting simulated process data to verify OPERABILITY of alarm and/or trip functions.

3/4.9.11 WATER LEVEL/BORON CONCENTRATION - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks. The dissolved boron concentration of the water in the storage pool shall be maintained at greater than or equal to 2000 ppm.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the water level requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. With the boron concentration requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and immediately take action to restore the dissolved boron concentration to within its limit as soon as possible.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

4.9.11.a Boron concentration in the storage pool shall be determined to be greater than or equal to 2000 ppm at least once per 48 hours.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.11 WATER LEVEL/BORON CONCENTRATION - STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

The restrictions on soluble boron concentration in the storage pool water ensure the spent fuel rack k_{eff} will be maintained less than or equal to 0.95 with a 95-percent confidence level.

3/4.9.12 FUEL HANDLING BUILDING EXHAUST FILTER PLENUM

The limitations on the Fuel Handling Building Exhaust Filter Plenum ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO, except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, ZIRLO, or stainless steel or by vacancies may be made if justified by a cycle specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of less than 3.20 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading or previous cycle loading. The enrichment of any reload fuel design shall be determined to be acceptable for storage in either the spent fuel pool or the new fuel vault. Such acceptance criteria shall be based on the results of the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162 and "Criticality Analysis of the Byron/Braidwood Fresh Fuel Racks," June 1989.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, silver-indium-cadmium, or a mixture of both types. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the UFSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

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

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,257 cubic feet at a nominal T_{avg} of 588.4°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

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

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

b. A 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 with Credit for Soluble Boron," Revision 1, November 1996;

c. A k_{eff} < 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 with Credit for Soluble Boron," Revision 1, November 1996;

d. A nominal 10.32 inch north-south and 10.42 inch east-west center-to-center distance between fuel assemblies placed in the Region 1 racks;

e. New or spent assemblies with sufficient Integral Fuel Burnable Absorbers present in each fuel assembly, as described in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162, which may be allowed unrestricted storage in the Region 1 racks;

f. A nominal 9.03 inch center-to-center distance between fuel assemblies placed in the Region 2 racks;

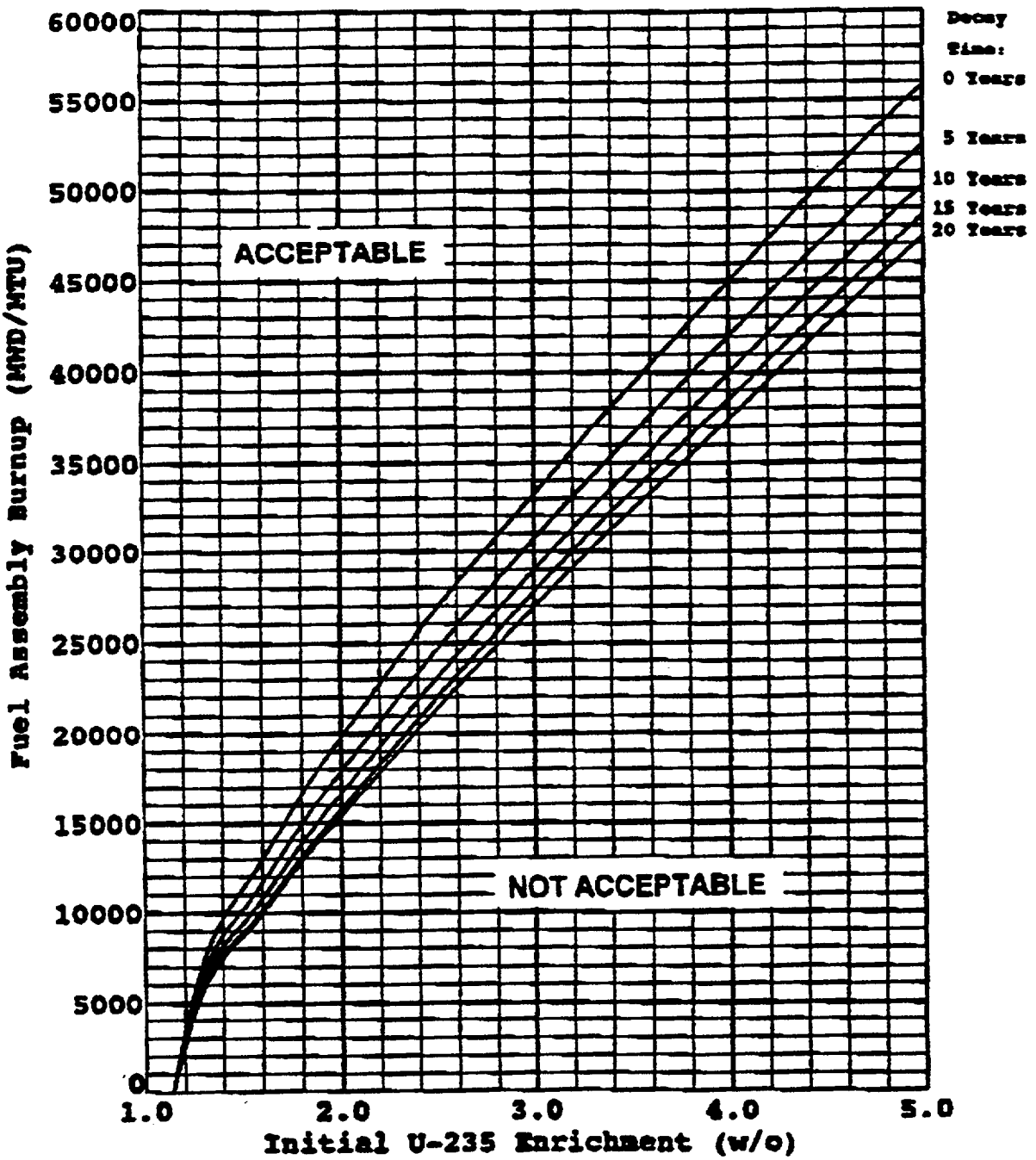
g. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment, and decay time in the acceptable region of Figures 5.6-1, 5.6-2 or 5.6-3, as applicable, may be stored in the Region 2 racks in the applicable checkerboard configuration, as described in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162; and

h. Interface requirements within and between adjacent racks as described in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

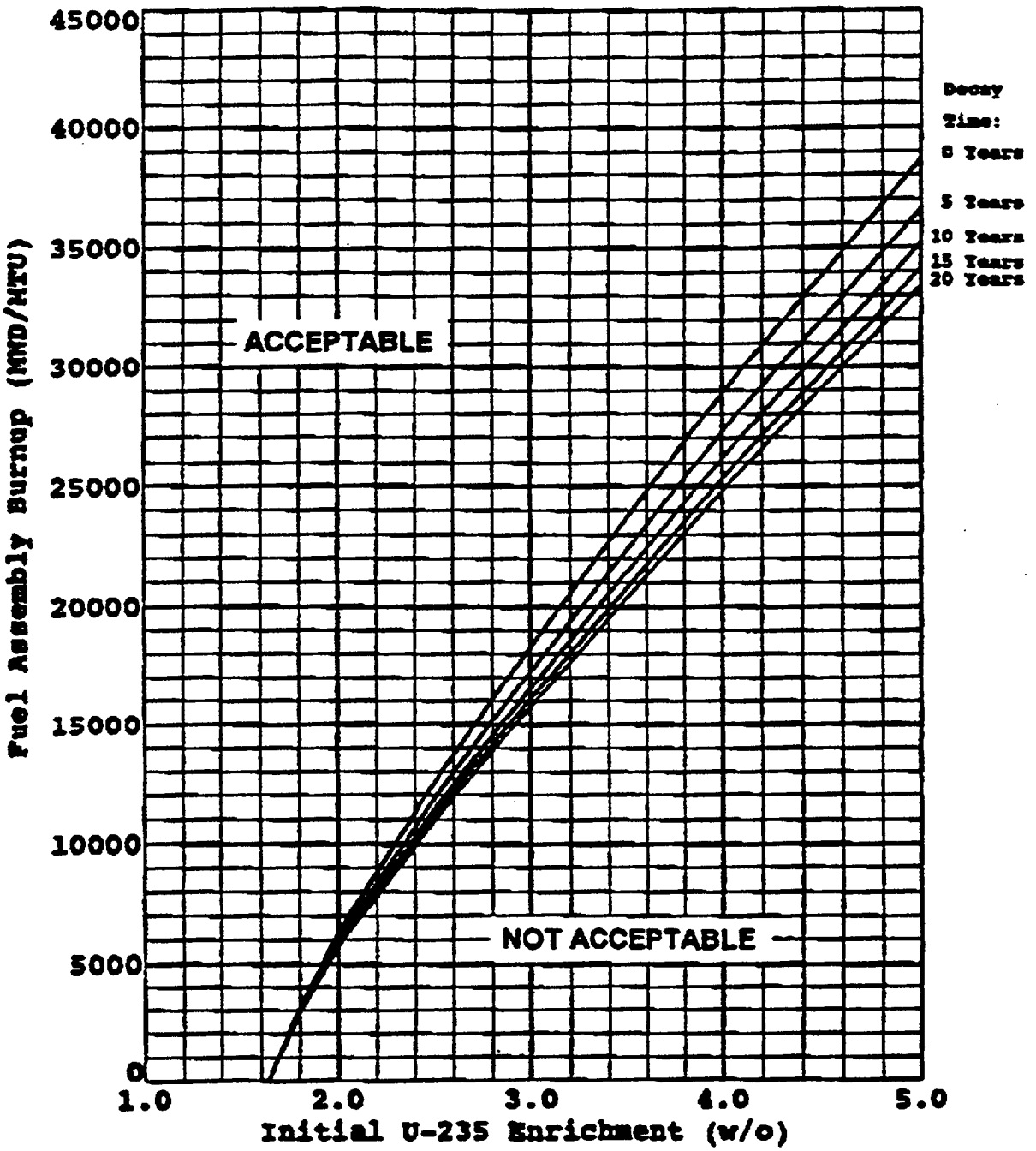
5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 423 feet 2 inches.



Note: The use of linear interpolation between the minimum burnups is acceptable.

FIGURE 5.6-1

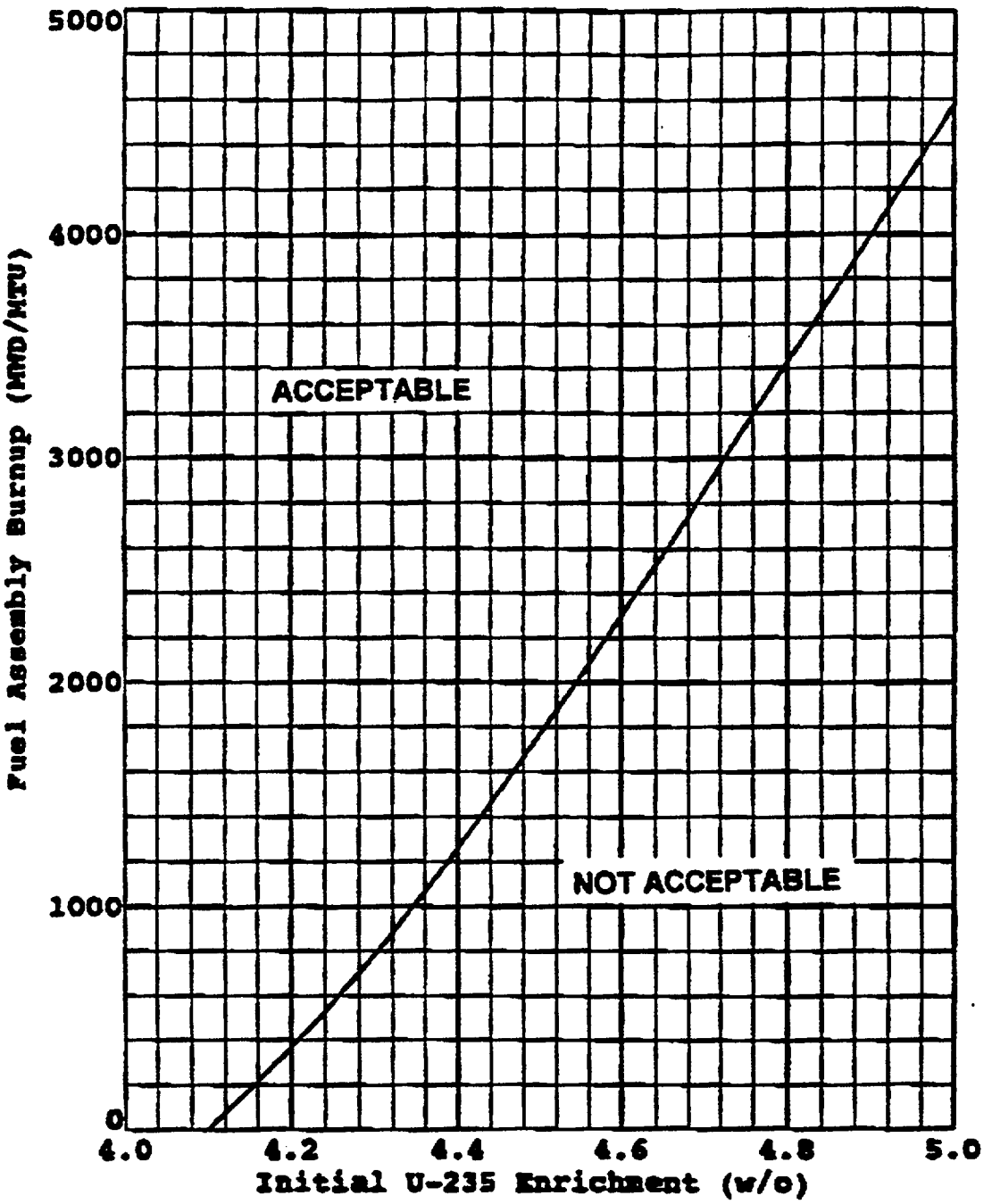
MINIMUM BURNUP VERSUS INITIAL ENRICHMENT FOR REGION 2
ALL CELL CONFIGURATION STORAGE



Note: The use of linear interpolation between the minimum burnups is acceptable.

FIGURE 5.6-2

MINIMUM BURNUP VERSUS INITIAL ENRICHMENT FOR REGION 2
3-OUT-OF-4 CHECKERBOARD CONFIGURATION



Note: The use of linear interpolation between the minimum burnups is acceptable.

FIGURE 5.6-3

MINIMUM BURNUP VERSUS INITIAL ENRICHMENT FOR REGION 2
2-OUT-OF-4 CHECKERBOARD CONFIGURATION

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

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. 86
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 June 30, 1997, as supplemented on September 25, 1997, 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:

9712170431 971204
PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No.86 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

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 4, 1997

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. 86
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 June 30, 1997, as supplemented on September 25, 1997, 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 1;

   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.86 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

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 4, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 86 AND 86

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. Pages indicated by an asterisk are provided for convenience only.

Remove Pages	Insert Pages
I	I
XII	XII
XVII	XVII
XVIII	XVIII
XX	XX
1-1*	1-1*
1-2	1-2
3/4 9-13	3/4 9-13
B 3/4 9-3	B 3/4 9-3
5-3*	5-3*
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- -	5-5c
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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation and injecting simulated process data to verify OPERABILITY of alarm and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

3/4.9.11 WATER LEVEL/BORON CONCENTRATION - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks. The dissolved boron concentration of the water in the storage pool shall be maintained at greater than or equal to 2000 ppm.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the water level requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. With the boron concentration requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and immediately take action to restore the dissolved boron concentration to within its limit as soon as possible.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

4.9.11.a Boron concentration in the storage pool shall be determined to be greater than or equal to 2000 ppm at least once per 48 hours.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.11 WATER LEVEL/BORON CONCENTRATION - STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

The restrictions on soluble boron concentration in the storage pool water ensure the spent fuel rack k_{eff} will be maintained less than or equal to 0.95 with a 95-percent confidence level.

3/4.9.12 FUEL HANDLING BUILDING EXHAUST FILTER PLENUM

The limitations on the Fuel Handling Building Exhaust Filter Plenum ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

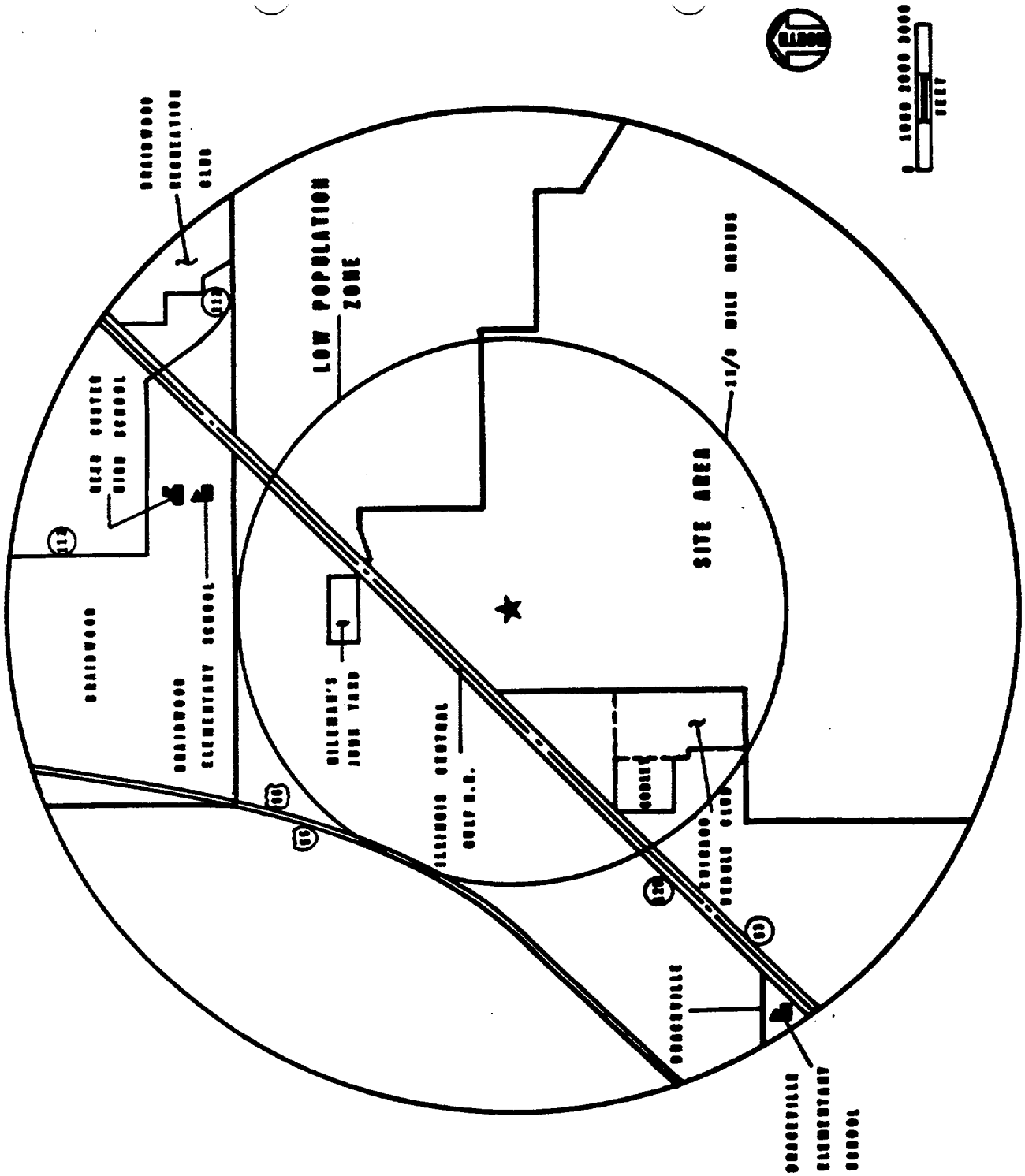


FIGURE 5.1-2

LOW POPULATION ZONE (L.P.Z.)
PUBLIC FACILITIES AND INSTITUTIONS WITHIN 1.125 MILES OF THE SITE

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO, except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, ZIRLO, or stainless steel or by vacancies may be made if justified by a cycle specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of less than 3.20 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading or previous cycle loading. The enrichment of any reload fuel design shall be determined to be acceptable for storage in either the spent fuel pool or the new fuel vault. Such acceptance criteria shall be based on the results of the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162 and "Criticality Analysis of the Byron/Braidwood Fresh Fuel Racks," June 1989.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, silver-indium-cadmium, or a mixture of both types. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the UFSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

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

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,257 cubic feet at a nominal T_{avg} of 588.4°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

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

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

b. A $k_{eff} < 1.0$ 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 with Credit for Soluble Boron," Revision 1, November 1996;

c. A $k_{eff} \leq 0.95$ 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 with Credit for Soluble Boron," Revision 1, November 1996;

d. A nominal 10.32 inch north-south and 10.42 inch east-west center-to-center distance between fuel assemblies placed in the Region 1 racks;

e. New or spent assemblies with sufficient Integral Fuel Burnable Absorbers present in each fuel assembly, as described in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162, which may be allowed unrestricted storage in the Region 1 racks;

f. A nominal 9.03 inch center-to-center distance between fuel assemblies placed in the Region 2 racks;

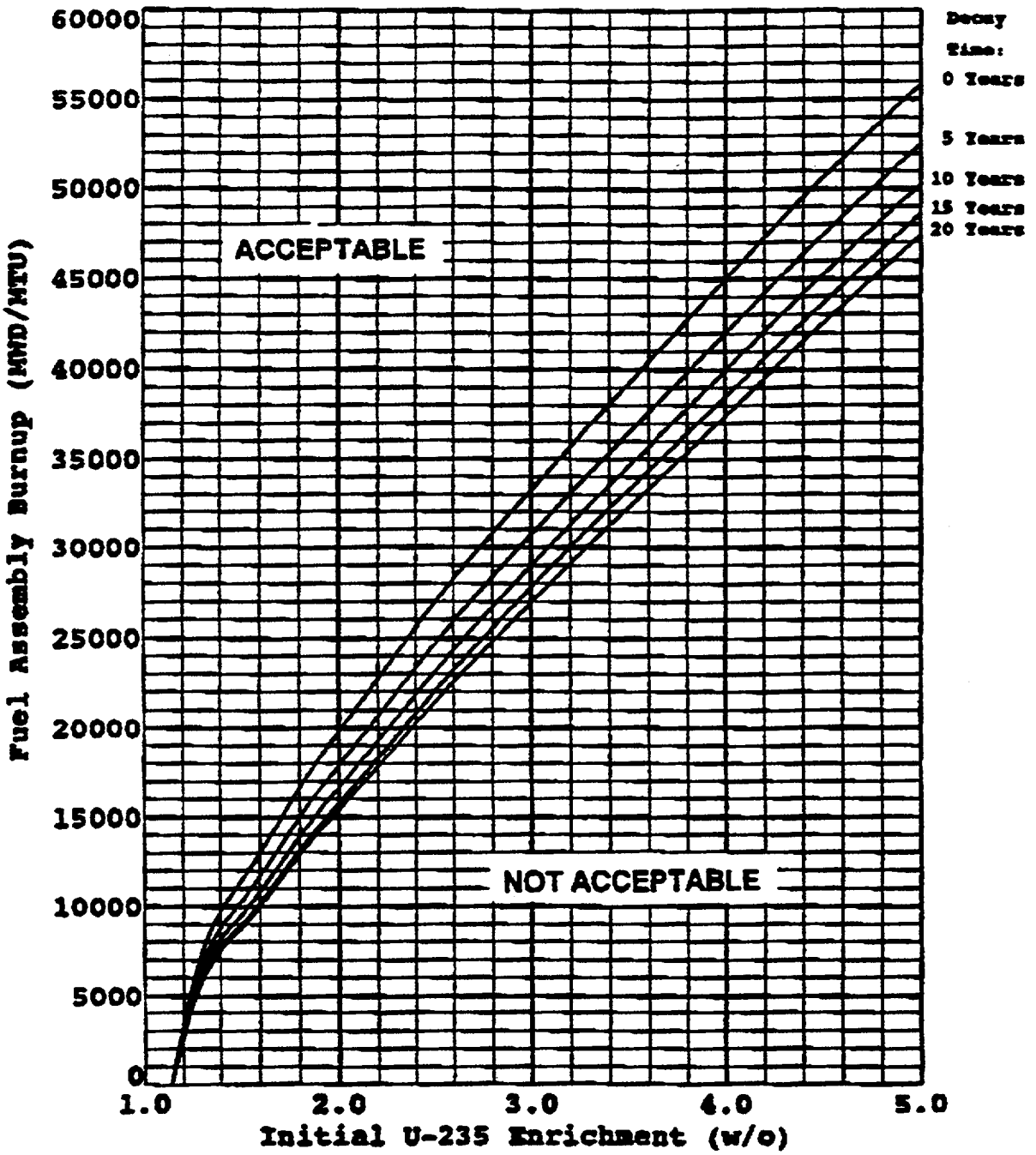
g. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment, and decay time in the acceptable region of Figures 5.6-1, 5.6-2, or 5.6-3, as applicable, which may be stored in the Region 2 racks in the applicable checkerboard configuration, as described in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162; and

h. Interface requirements within and between adjacent racks as described in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

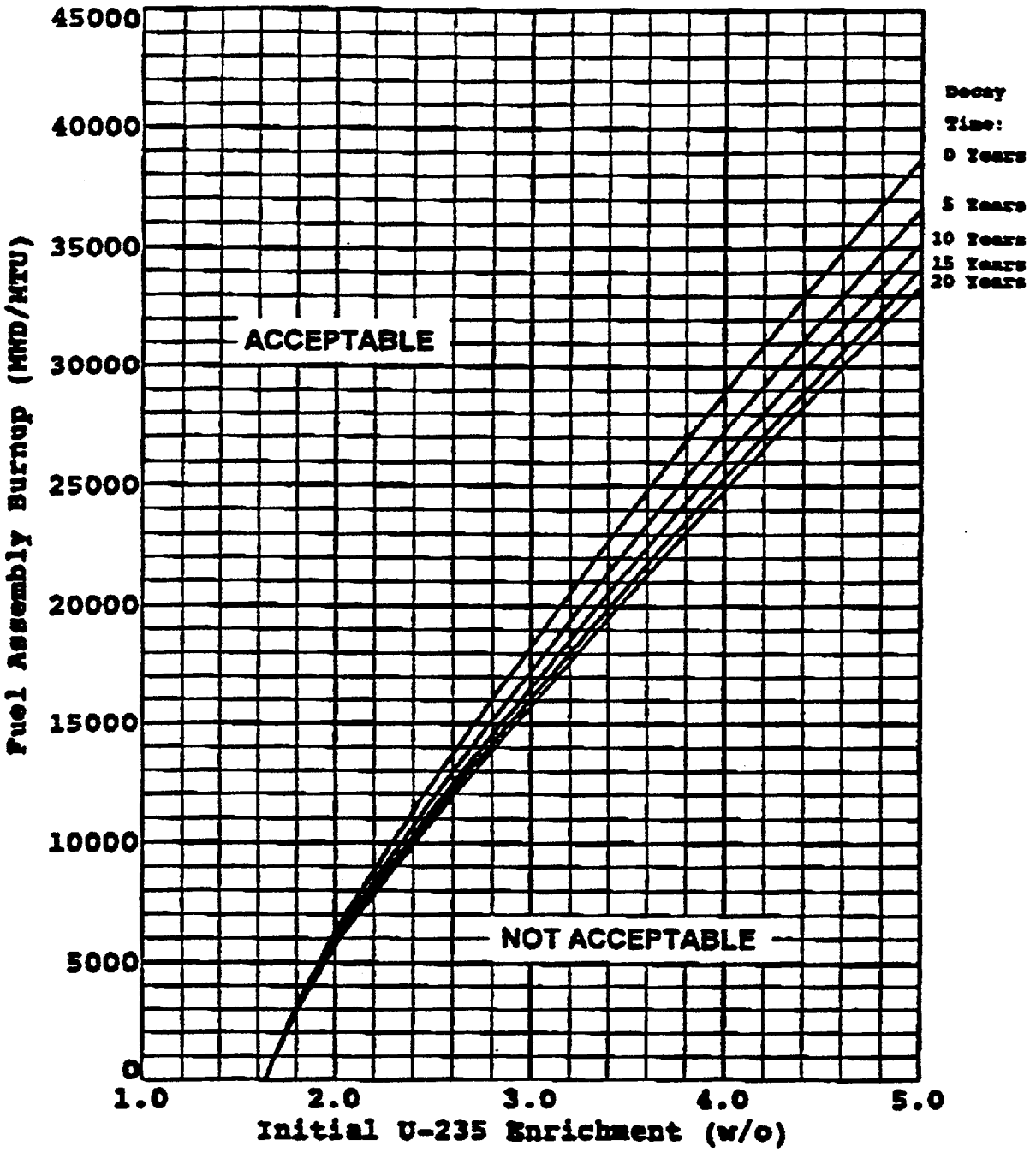
5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 423 feet 0 inches.



Note: The use of linear interpolation between the minimum burnups is acceptable.

FIGURE 5.6-1

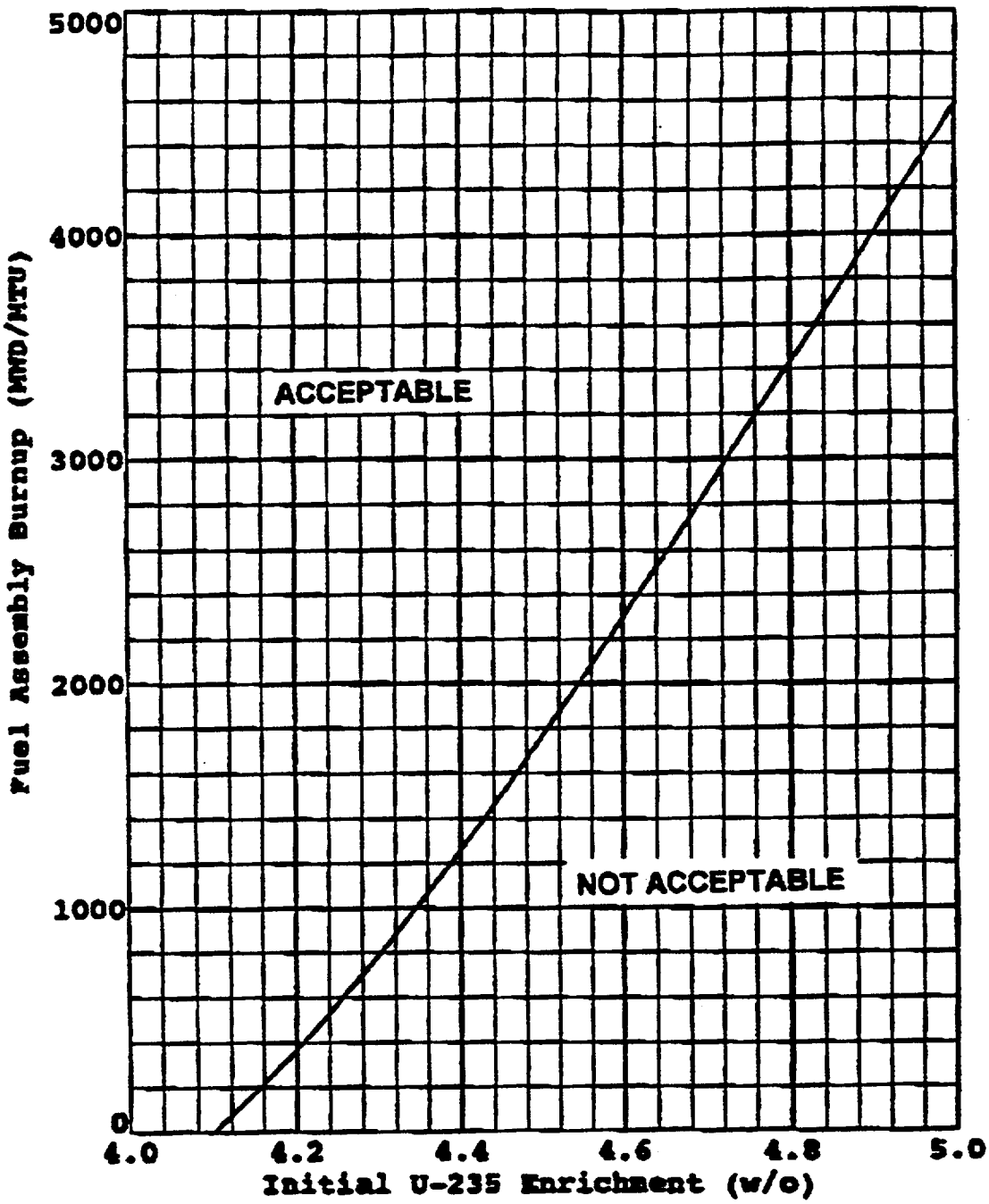
MINIMUM BURNUP VERSUS INITIAL ENRICHMENT FOR REGION 2
ALL CELL CONFIGURATION STORAGE



Note: The use of linear interpolation between the minimum burnups is acceptable.

FIGURE 5.6-2

MINIMUM BURNUP VERSUS INITIAL ENRICHMENT FOR REGION 2
3-OUT-OF-4 CHECKERBOARD CONFIGURATION



Note: The use of linear interpolation between the minimum burnups is acceptable.

FIGURE 5.6-3

MINIMUM BURNUP VERSUS INITIAL ENRICHMENT FOR REGION 2
2-OUT-OF-4 CHECKERBOARD CONFIGURATION

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO.94 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO.86 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO.86 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

On April 2, 1997, the staff issued License Amendment Nos. 86 for Byron Station, Units 1 and 2, and License Amendment Nos. 78 for Braidwood Station, Units 1 and 2, granting partial credit for boron in the spent fuel pools to maintain the subcriticality. The amendments were granted because Commonwealth Edison Company (ComEd) had noted degradation of the Boroflex panels in the spent fuel storage cells. However the amendments were to be in effect only until December 31, 1997, while development of a long-term solution was underway. In a letter of June 30, 1997 (Reference 1), supplemented by letter of September 25, 1997 (Reference 2), ComEd requested changes to the Byron and Braidwood, Units 1 and 2, Technical Specifications (TS) to allow the use of credit for soluble boron in the spent fuel pool criticality analyses. These criticality analyses were performed using the methodology developed by the Westinghouse Owners Group (WOG) and described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" (Reference 3).

2.0 EVALUATION

2.1 Criticality Analysis

The Byron and Braidwood spent fuel storage racks were analyzed using the Westinghouse methodology which has been reviewed and approved by the NRC (Reference 3). This methodology takes partial credit for soluble boron in the fuel storage pool criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1) k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level as described in WCAP-14416-NP-A; and

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2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in WCAP-14416-NP-A.

The analysis of the reactivity effects of fuel storage in the Byron and Braidwood spent fuel racks was performed with the three-dimensional Monte Carlo code, KENO-Va, with neutron cross sections generated with the NITAWL-II and XSDRNPM-S codes using the 227 group ENDF/B-V cross-section library. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, PHOENIX-P, which uses a 42 energy group nuclear data library. The analytical methods and models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the Byron and Braidwood racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include close proximity storage and strong neutron absorbers. 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.

The Byron and Braidwood spent fuel pools are composed of three different types of storage racks, designated as Region 1, Region 2, and Failed Assembly cells. Region 1 contains three distinct cell types denoted as Type 1, Type 2, and Type 3. All cells contain 7.75 inch wide Boraflex sheets of 0.075 inch thickness. The Type 1 interior cells also contain Boral sheets outside of the Boraflex on all four sides. The Type 2 cells are side peripheral cells with Boral sheets outside of the Boraflex only on the three interior sides. Type 3 cells are corner peripheral cells with Boral sheets outside of the Boraflex only on the two interior sides. The Region 2 racks contain Boraflex sheets on all four sides, but no Boral. The six Failed Assembly cells contain no neutron absorber but rely upon the 21-inch lattice spacing to meet criticality requirements.

The spent fuel storage racks have previously been qualified for storage of various Westinghouse 17x17 fuel assembly types with maximum enrichments up to 5.0 weight percent (w/o) U-235. The maximum enrichment is based on a nominal value of 4.95 w/o U-235 plus a manufacturing tolerance of 0.05. The spent fuel rack Boraflex absorber panels were considered in this previous analysis. Because of the Boraflex deterioration that has been observed in many spent fuel pools, the Byron and Braidwood spent fuel storage racks have been reanalyzed neglecting the presence of Boraflex to allow storage of all 17x17 fuel assemblies with nominal enrichments up to 5.0 w/o U-235 (enrichment tolerance of ±0.05 w/o U-235) using credit for checkerboarding, burnup, burnable absorbers, and soluble boron.

For the Region 1 and Region 2 storage racks, the moderator was assumed to be pure water at a temperature of 68 degrees Fahrenheit and a density of 1.0 gm/cc and the array was assumed to be infinite in lateral extent. Uncertainties due to tolerances in fuel enrichment and density, storage cell inner diameter, storage cell pitch, stainless steel thickness, assembly position, calculational uncertainty, and methodology bias uncertainty were accounted for. For the Region 1 analysis, a ±0.007 inch tolerance about the nominal Boral sheet thickness of 0.075 inches was also used. These uncertainties were appropriately determined at the 95/95

probability/confidence level. A methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of the normal range of spent fuel pool water temperatures (50 degrees Fahrenheit to 160 degrees Fahrenheit) were included. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

For Region 1, the enrichment required to maintain k_{eff} less than 1.0 with all cells filled with Westinghouse 17x17 OFA fuel assemblies (most reactive type) and no soluble boron in the pool water was found to be 4.70 w/o U-235. This resulted in a nominal k_{eff} of 0.98264. The 95/95 k_{eff} was then determined by adding the temperature and methodology biases and the statistical sum of independent tolerances and uncertainties to the nominal k_{eff} values, as described in Reference 3. This resulted in a 95/95 k_{eff} of 0.99944. Since this value is less than 1.0 and was determined at a 95/95 probability/confidence level, it meets the NRC criterion for precluding criticality with no credit for soluble boron and is acceptable.

Storage of fuel assemblies with nominal enrichments between 4.70 and 5.0 w/o U-235 is achievable by crediting the reactivity decrease associated with Integral Fuel Burnable Absorbers (IFBAs). IFBAs consist of a boron neutron absorbing material applied as a thin coating on the outside of the UO_2 fuel pellet. Reactivity calculations were performed to determine the number of IFBA rods needed for nominally enriched 5.0 w/o fuel which yield a k_{eff} equivalent to or lower than that previously obtained for 4.70 w/o U-235 fuel. For this analysis, the fuel assembly is modeled at its most reactive point in cycle life. Each IFBA rod contains the minimum standard B-10 loading (1.50 mg/in) offered by Westinghouse for 17x17 OFA fuel. Uncertainties associated with IFBA credit include a 5 percent manufacturing tolerance and a 10 percent calculational uncertainty on the B-10 loading. The staff finds these uncertainties adequately conservative and acceptable. The lowest IFBA pattern (16 rods) offered by Westinghouse was found to be acceptable.

In order to assure that a 5 percent subcritical margin is maintained, credit for soluble boron was taken. The soluble boron credit calculations for Region 1 assumed the all cell storage configuration moderated by water borated to 400 ppm. As previously described, the individual tolerances and uncertainties and the temperature and methodology biases were added to the calculated nominal k_{eff} to obtain a 95/95 value. The resulting 95/95 k_{eff} was 0.94569 for OFA fuel enriched to 4.70 w/o U-235. Since k_{eff} is less than 0.95 with 400 ppm of boron and uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion for precluding criticality is satisfied. This is well below the minimum spent fuel pool boron concentration value of 2000 ppm required by TS 3.9.11 and is, therefore, acceptable.

In order to determine the enrichment required to maintain k_{eff} less than 1.0 in Region 2 with no credit for soluble boron or Boraflex, three configurations were analyzed. The first was all cell storage in which fuel assemblies with sufficiently low nominal enrichments were stored in every cell location. The second involved a 3-out-of-4 checkerboarding pattern with one empty cell and fuel assemblies with nominal enrichments up to 1.64 w/o U-235 in the other three cells. The third configuration used a 2-out-of-4 assembly checkerboard with two diagonally adjacent empty cells and two assemblies with nominal enrichments up to 4.10 w/o U-235 in the other diagonally adjacent cells. The criticality analyses considered all the fuel types currently stored in the spent fuel pool and in use at Byron and Braidwood. The reactivity of the Westinghouse OFA design

was found to bound other fuel types in use or stored at Byron and Braidwood (VANTAGE 5, VANTAGE+, and PERFORMANCE+).

The nominal k_{eff} for the all cell storage configuration was determined to be 0.96885 for Westinghouse OFA fuel enriched to 1.14 w/o U-235 with no credit for soluble boron or Boraflex. The 95/95 k_{eff} was then determined by adding the temperature and methodology biases and the statistical sum of independent tolerances and uncertainties to the nominal k_{eff} values, as described in Reference 3. This resulted in a 95/95 k_{eff} of 0.99631. For the 3-out-of-4 checkerboard configuration with one empty cell and the remaining cells containing assemblies having a nominal enrichment no greater than 1.64 w/o U-235 and no credit for soluble boron or Boraflex, the 95/95 k_{eff} was 0.99662. The 2-out-of-4 checkerboard configuration with two diagonally adjacent cells containing assemblies having a nominal enrichment no greater than 4.10 w/o U-235 and the remaining cells empty and no credit for soluble boron or Boraflex, the 95/95 k_{eff} was 0.99664. Since these values are less than 1.0 and were determined at a 95/95 probability/confidence level, they meet the NRC criterion for precluding criticality and are acceptable.

The soluble boron credit calculations assumed 1.14 w/o U-235 fuel in the all cell storage configuration moderated by water borated to 150 ppm. This is well below the minimum spent fuel pool boron concentration value of 2000 ppm required by TS 3.9.11 and is, therefore, acceptable. As previously described, the individual tolerances and uncertainties and the temperature and methodology biases were added to the calculated nominal k_{eff} to obtain a 95/95 value. The resulting 95/95 k_{eff} was 0.94743. Since k_{eff} is less than 0.95 with 150 ppm of boron and uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion for precluding criticality is satisfied.

For the checkerboard configurations described above, a soluble boron concentration of 200 ppm was found to result in a 95/95 k_{eff} of 0.93999 and 0.94663, for the 3-out-of-4 and the 2-out-of-4 checkerboard configurations, respectively. Therefore, these also meet the NRC acceptance criterion for precluding criticality and are acceptable.

The concept of reactivity equivalencing due to fuel burnup was used to achieve the storage of fuel assemblies with enrichments higher than 1.14 w/o U-235 for the all cell storage configuration and fuel assemblies with enrichments higher than 1.64 and 4.10 w/o U-235 for the 3-out-of-4 and 2-out-of-4 checkerboard configurations, respectively. The NRC has previously accepted the use of reactivity equivalencing predicated upon the reactivity decrease associated with fuel depletion. This analysis also includes spent fuel decay time credit, which results from the radioactive decay of isotopes in the spent fuel to daughter isotopes. The loss in reactivity due to the radioactive decay of the spent fuel results in reducing the minimum burnup needed to meet the reactivity requirements.

In the decay time methodology, the fission product isotopes are frozen at the concentrations existing at the time of discharge from the core, except for Xe-135, which is removed. These calculations are performed at different discharge burnups. The fuel is depleted using a high soluble boron letdown curve to enhance the buildup of plutonium making the fuel more reactive in the spent fuel storage racks. Credit is taken only for the decay of actinides, one of the major contributors being the decay of Pu-241 to Am-241. Calculations by Westinghouse from 100

hours after shutdown (at which time the major fission product Xe-135 has essentially decayed away) to 30 years following shutdown have shown that decay of the fission products has the effect of continuously reducing the reactivity of the spent fuel. However, no credit for fission product decay is used in the decay time credit. Based on these conservative assumptions, the staff concludes that the proposed use of decay time credit is acceptable.

To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ for storage of fuel assemblies in Region 2 with enrichments up to 5.0 w/o U-235, a series of reactivity calculations was performed to generate a set of enrichment versus fuel assembly discharge burnup ordered pairs which all yield an equivalent k_{eff} when stored in the Byron and Braidwood spent fuel storage racks. These are shown in TS Figures 5.6-1 through 5.6-3 and represent combinations of fuel enrichment and discharge burnup which yield the same rack k_{eff} as the rack loaded with 1.14, 1.64, and 4.10 w/o fuel (at zero burnup) for all cell, 3-out-of-4, and 2-out-of-4 storage configurations, respectively. Decay time credit is included for the all cell and 3-out-of-4 storage configurations. Uncertainties associated with burnup credit include a reactivity uncertainty of 0.01 Δk at 30,000 MWD/MTU applied linearly to the burnup credit requirement to account for calculational and depletion uncertainties and 5 percent on the calculated burnup to account for burnup measurement uncertainty. The NRC staff concludes that these uncertainties are acceptable. The amount of additional soluble boron, above the values required above, that is needed to account for these uncertainties is 400 ppm for the all cells configuration, 350 ppm for the 3-out-of-4 checkerboard configuration, and 50 ppm for the 2-out-of-4 checkerboard configuration. This results in a total soluble boron credit of 550 ppm for the all cells and 3-out-of-4 configurations and 250 ppm for the 2-out-of-4 configuration. These are well below the minimum spent fuel pool boron concentration value of 2000 ppm required by TS 3.9.11 and are, therefore, acceptable.

Although most accidents will not result in a reactivity increase, two accidents can be postulated for each storage configuration which would increase reactivity beyond the analyzed conditions. The first would be a loss of fuel pool cooling system and a rise in pool water temperature from 160 degrees Fahrenheit to 240 degrees Fahrenheit. The second would be a misload of an assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied. Calculations have shown that the misload assembly accident for a 2-out-of-4 checkerboard configuration results in the highest reactivity increase. The reactivity increase requires an additional 1400 ppm of soluble boron. However, for such events, the double contingency principle can be applied. This states that the assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident. Therefore, the minimum amount of boron required by TS 3.9.11 (2000 ppm) is more than sufficient to cover any accident and the presence of the additional boron above the concentration required for normal conditions and reactivity equivalencing (550 ppm maximum) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. Similar calculations for the six Failed Assembly spent fuel racks when all cells contain fuel enriched to 5.0 w/o U-235 resulted in a 95/95 k_{eff} of 0.96330 with no credit taken for soluble boron and 0.93543 when 100 ppm of soluble boron is included. An additional 100 ppm of soluble boron is required for accidents, resulting in a total boron credit of 200 ppm, well below the minimum spent fuel pool boron concentration value of 2000 ppm required by TS 3.9.11 and, therefore, acceptable.

In order to prevent an undesirable increase in reactivity, the boundaries between the different storage configurations were analyzed. For Region 1, the boundary between the rack modules must use a 3-out-of-4 checkerboard arrangement since the peripheral cell exterior sides do not contain Boral sheets. Several interface requirements apply within Region 2. The boundary between all cell storage and 2-out-of-4 or 3-out-of-4 checkerboard storage must either be separated by a vacant row of cells or configured such that the first row of carryover in the checkerboard storage zone uses 1.64 w/o fuel assemblies alternating with empty cells. The boundary between 2-out-of-4 and 3-out-of-4 checkerboard storage must either be separated by a vacant row of cells or the interface must be configured such that the first row of carryover in the 2-out-of-4 storage zone uses 4.10 w/o assemblies alternating with empty cells. Since the criticality analysis does not consider the Boraflex neutron absorber panels, the boundary between Region 1 and Region 2 must be configured such that one row of vacant cells is maintained between the regions.

The TS changes proposed as a result of the revised criticality analysis are consistent with the changes stated in the NRC Safety Evaluation (SE) for WCAP-14416-P (Reference 4). Westinghouse submitted a revised topical report, WCAP-14416-NP-A, Revision 1, which incorporated the changes stated in the NRC SE. Also, since the staff disagreed with the proprietary finding of the original WCAP-14416-P, Westinghouse's revised topical report was submitted as a nonproprietary version. Based on this consistency with the approved methodology and on the above evaluation, the staff finds these TS changes acceptable. The proposed associated Bases changes adequately describe these TS changes and are also acceptable.

2.2 Boron Dilution of the Spent Fuel Pool

In accordance with the NRC Safety Evaluation of the Westinghouse methodology described in WCAP-14416-NP-A (Reference 4), the licensee performed a boron dilution analysis to ensure that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. Potential events were quantified to show that sufficient time will be available to enable adequate detection and suppression of any dilution event.

Deterministic dilution event calculations were performed for Byron and Braidwood to define the dilution times and volumes necessary to dilute the spent fuel pool from the minimum TS boron concentration of 2000 ppm to a soluble boron concentration where k_{eff} of 0.95 would be approached (550 ppm). The volume required to dilute the spent fuel pool from the TS limit to 550 ppm is 612,000 gallons. The various events that were considered included dilution from the boron recovery system recycle holdup tanks, primary water system, chemical volume and control system blender, demineralized water system, fire protection system, reverse osmosis system, spent fuel pool demineralizers, station heating system, and component cooling water system, and other events that may affect the boron concentration of the pool, such as seismic events, pipe break, and loss of offsite power.

The licensee's evaluation concluded that the most limiting event was determined to be using the primary water connection to the spent fuel pool cleanup subsystem for either makeup or for transferring spent resin. It is the only unborated water authorized for use under normal plant conditions. A dilution event would occur if the process isolation valve was inadvertently left

open. It would take approximately 46 hours at a rate of 220 gpm to dilute the spent fuel pool to 550 ppm. It is reasonable to expect that alarms or operator rounds (every 8 hours) would detect the increase in spent fuel pool level prior to reaching 550 ppm and the event would be terminated by plant personnel. No single tank source is large enough at Byron to dilute the spent fuel pool below 550 ppm. At Braidwood, the only source large enough to dilute the spent fuel pool below 550 ppm (508 ppm) is the condensate storage tank (CST) which has a capacity of 650,000 gallons. The water from the CSTs also allow the highest dilution rate of 420 gpm. At this dilution rate, it would take 24 hours to dilute the spent fuel pool to 550 ppm. Again, it is reasonable to expect that this increase in the spent fuel pool level would be detected by alarms or by operator rounds and terminated by plant personnel. The evaluation of the dilution sources determined that two scenarios do not change the level of the spent fuel pool during dilution. These two scenarios involve the transfer of water from the reverse osmosis system and the spent fuel pool demineralizers. The limiting dilution would require 51 hours. The TS surveillance requirement of sampling every 48 hours would detect the dilution event in sufficient time. The staff finds that a boron dilution event to 550 ppm for the Byron and Braidwood plants is unlikely and the licensee's analysis is acceptable.

The licensee's evaluation concluded that an unplanned or inadvertent event which would result in the dilution of the spent fuel pool boron concentration from 2000 ppm to 550 ppm is not a credible event. The staff finds that the combination of the TS-controlled spent fuel pool concentration and 48-hour sampling requirement, the alarms, operator rounds, and other administrative controls should adequately detect a dilution event prior to k_{eff} reaching 0.95 (550 ppm) and, therefore, the analysis and proposed TS controls are acceptable for the boron dilution aspects of the request.

Additionally, the criticality analysis for the spent fuel storage pool show that k_{eff} remains less than 1.0 at a 95/95 probability/confidence level even if the pool were completely filled with unborated water. Therefore, even if the spent fuel storage pool were diluted to zero ppm, the fuel is expected to remain subcritical.

3.0 SUMMARY

Based on the review described above, the staff finds the criticality aspects of the proposed Byron and Braidwood license amendment request are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The analysis assumed credit for soluble boron, as allowed by WCAP-14416-NP-A, but no credit for the Boraflex neutron absorber panels. The amount of soluble boron required for each storage configuration analyzed is shown in attached Table 1.

The following storage configurations and U-235 enrichment limits for Westinghouse 17x17 OFA fuel assemblies were determined to be acceptable.

For Region 1:

> Assemblies with initial nominal enrichments no greater than 4.70 w/o U-235 can be stored in any cell location. Fuel assemblies with initial nominal enrichments greater than this and up to 5.0 w/o U-235 must contain at least 16 IFBA rods.

For Region 2:

Assemblies with initial nominal enrichments no greater than 1.14 w/o U-235 can be stored in any cell location. Fuel assemblies with initial nominal enrichments greater than this and up to 5.0 w/o U-235 must satisfy a minimum burnup requirement as shown in TS Figure 5.6-1.

Assemblies with initial nominal enrichments no greater than 1.64 w/o U-235 can be stored in a 3-out-of-4 checkerboard arrangement. Fuel assemblies with initial nominal enrichments greater than this and up to 5.0 w/o U-235 must satisfy a minimum burnup requirement as shown in TS Figure 5.6-2.

Assemblies with initial nominal enrichments no greater than 4.10 w/o U-235 can be stored in a 2-out-of-4 checkerboard arrangement. Fuel assemblies with initial nominal enrichments greater than this and up to 5.0 w/o U-235 must satisfy a minimum burnup requirement as shown in TS Figure 5.6-3.

For Failed Assembly Cells:

Assemblies with initial nominal enrichments no greater than 5.0 w/o U-235 can be stored in any cell location.

With respect to possible inadvertent dilution of boron from the spent fuel pool, the staff finds that the combination of the TS-controlled spent fuel pool concentration and 48-hour sampling requirement, the alarms, operator rounds, and other administrative controls should adequately detect a dilution event prior to k_{eff} reaching 0.95 (550 ppm) and, therefore, the analysis and proposed TS controls are acceptable for the boron dilution aspects of the request.

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

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 54868). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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.

Attachment: Summary of Soluable Boron Credit Requirements for Byron/Braidwood, Units 1 and 2

Principal Contributors: L. Kopp
D. Jackson

Date: December 4, 1997

7.0 REFERENCES

1. Letter, J. B. Hosmer, ComEd, to U. S. NRC, Subject: "Application for Amendment to Appendix A, Technical Specifications, for Facility Operating Licenses: Byron and Braidwood Nuclear Power Stations, Units 1 and 2, Boron Credit in the Spent Fuel Storage Pool," dated June 30, 1997.

2. Letter, J. B. Hosmer, ComEd, to U. S. NRC, Subject: "Supplement to the Application for Amendment Request for Appendix A, Technical Specifications, to Facility Operating Licenses: NPF-37, NPF-66, NPF-72, and NPF-77, Boron Credit in the Spent Fuel Pool," dated September 25, 1997.

3. Westinghouse Electric Corporation, WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Newmyer, W. D., dated November 1996.

4. Letter, T. E. Collins, NRC, to T. Greene, Westinghouse Owners' Group, Subject: "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Criticality Analysis Methodology," (TAC No. M93254), dated October 25, 1996.

TABLE 1

Summary of Soluble Boron Credit Requirements for Byron/Braidwood Units 1 and 2

Region	Storage Configuration	Soluble Boron Required for $k_{eff} \leq 0.95$ (ppm)	Soluble Boron Required for Reactivity Euivalencing (ppm)	Total Soluble Boron Credit Required Without Accidents (ppm)
1	All Cell Storage	400	0	400
2	All Cell Storage	150	400	550
2	3-out-of-4 Checkerboard	200	350	550
2	2-out-of-4 Checkerboard	200	50	250
Failed Assembly	All Cell Storage	100	n/a	100