June 12, 1997

Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CREDIT FOR SOLUBLE BORON IN SPENT FUEL POOL CRITICALITY ANALYSES (TAC NOS. M93072 AND M93073)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 129 to Facility Operating License No. DPR-42 and Amendment No. 121 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 28, 1995, as revised February 21, 1997.

The amendments revise the TS for the Prairie Island Nuclear Generating Plant to allow credit for soluble boron in spent fuel criticality analyses. The request is based on the NRC approval of the Westinghouse Owners Group generic methodology for crediting soluble boron given in Topical Report WCAP-14416-NP-A.

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

Sincerely,

ORIGINAL SIGNED BY

Beth A. Wetzel, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 129 to DPR-42
2. Amendment No. 121 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

DISTRIBUTION: See attached page

DOCUMENT NAME: G:\WPDOCS\PRAIRIE\PI93072.AMD *See previous concurrence

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DATED: June 12, 1997

AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company (the licensee) dated July 28, 1995, as revised February 21, 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. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 129 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel

Beth A. Wetzel, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 12, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

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

REMOVE	INSERT
TS-vi	TS-vi
TS-xi	TS-xi
TS-xiii	TS-xiii
TS.3.8-4	TS.3.8-4
TS.3.8-5	--
Figure TS.3.8-1	Figure TS.3.8-1
--	Figure TS.3.8-2
Table TS.4.1-2B (Page 1 of 2)	Table TS.4.1-2B (Page 1 of 2)
Table TS.4.1-2B (Page 2 of 2)	Table TS.4.1-2B (Page 2 of 2)
--	TS.4.20-1
TS.5.6-1	TS.5.6-1
TS.5.6-3	TS.5.6-3
Figure TS.5.6-1	Figure TS.5.6-1
Figure TS.5.6-2	Figure TS.5.6-2
--	Figure TS.5.6-3
--	Figure TS.5.6-4
--	Figure TS.5.6-5
--	Figure TS.5.6-6
--	Figure TS.5.6-7
--	Figure TS.5.6-8
--	Figure TS.5.6-9
--	Figure TS.5.6-10
--	Figure TS.5.6-11
--	Figure TS.5.6-12
B.3.8-2	B.3.8-2
B.3.8-3	B.3.8-3
B.3.8-4	B.3.8-4
--	B.3.8-5
--	B.4.20-1

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APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

3.8.C. Small Spent Fuel Pool Restrictions

No more than 45 recently discharged assemblies shall be located in the small pool (pool No. 1).

D. Spent Fuel Pool Special Ventilation System

1. Both trains of the Spent Fuel Pool Special Ventilation System shall be OPERABLE at all times (except as specified in 3.8.D.2 and 3.8.D.3 below).

2. With one train of the Spent Fuel Pool Special Ventilation System inoperable, fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure) are permissible during the following 7 days, provided the redundant train is demonstrated OPERABLE prior to proceeding with those operations.

3. With both trains of the Spent Fuel Pool Special Ventilation System inoperable, suspend all fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure).

4. The provisions of specification 3.0.C are not applicable.

E. Spent Fuel Pool Storage

1. Fuel Assembly Storage

   a. The combination of initial enrichment, burnup and decay time of each spent fuel assembly stored in the spent fuel pool shall be within the unrestricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, or fuel assemblies shall be stored in accordance with Specification 5.6.A.1.e.

   b. If the requirements of 3.8.E.1.a are not met, immediately initiate action to move any noncomplying fuel assembly to an acceptable location.

   c. The provisions of Specification 3.0.C are not applicable.

2. Spent Fuel Pool Boron Concentration

   a. The spent fuel pool boron concentration shall be ≥ 1,800 ppm when fuel assemblies are stored in the spent fuel pool.

   b. If the spent fuel pool boron concentration is not within limit, then immediately:

      1. Suspend movement of fuel assemblies in the spent fuel pool, and

      2. Initiate action to restore spent fuel pool boron concentration to within limit.

   c. The provisions of Specification 3.0.C are not applicable.

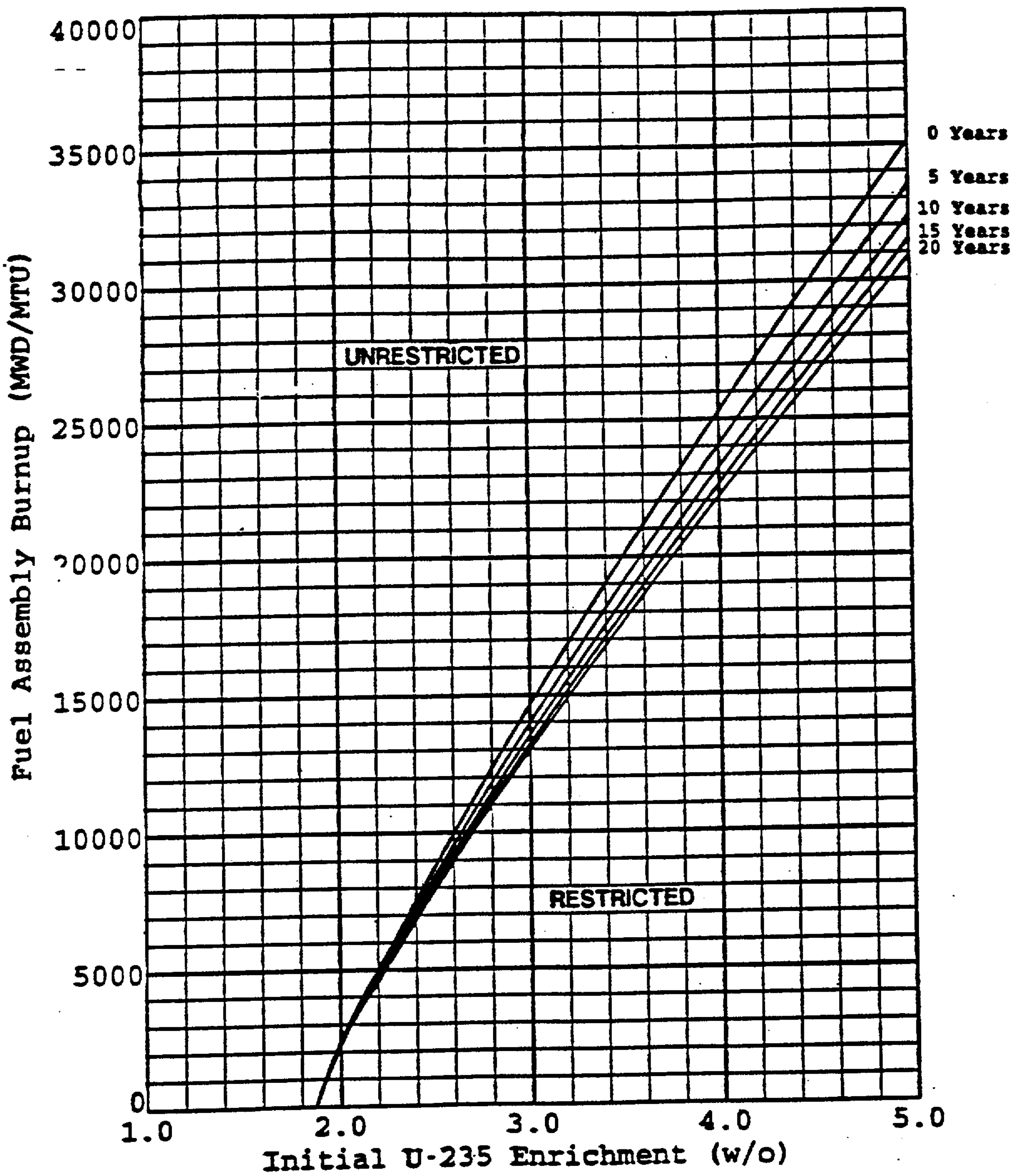


FIGURE TS.3.8-1 Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements - OFA Fuel

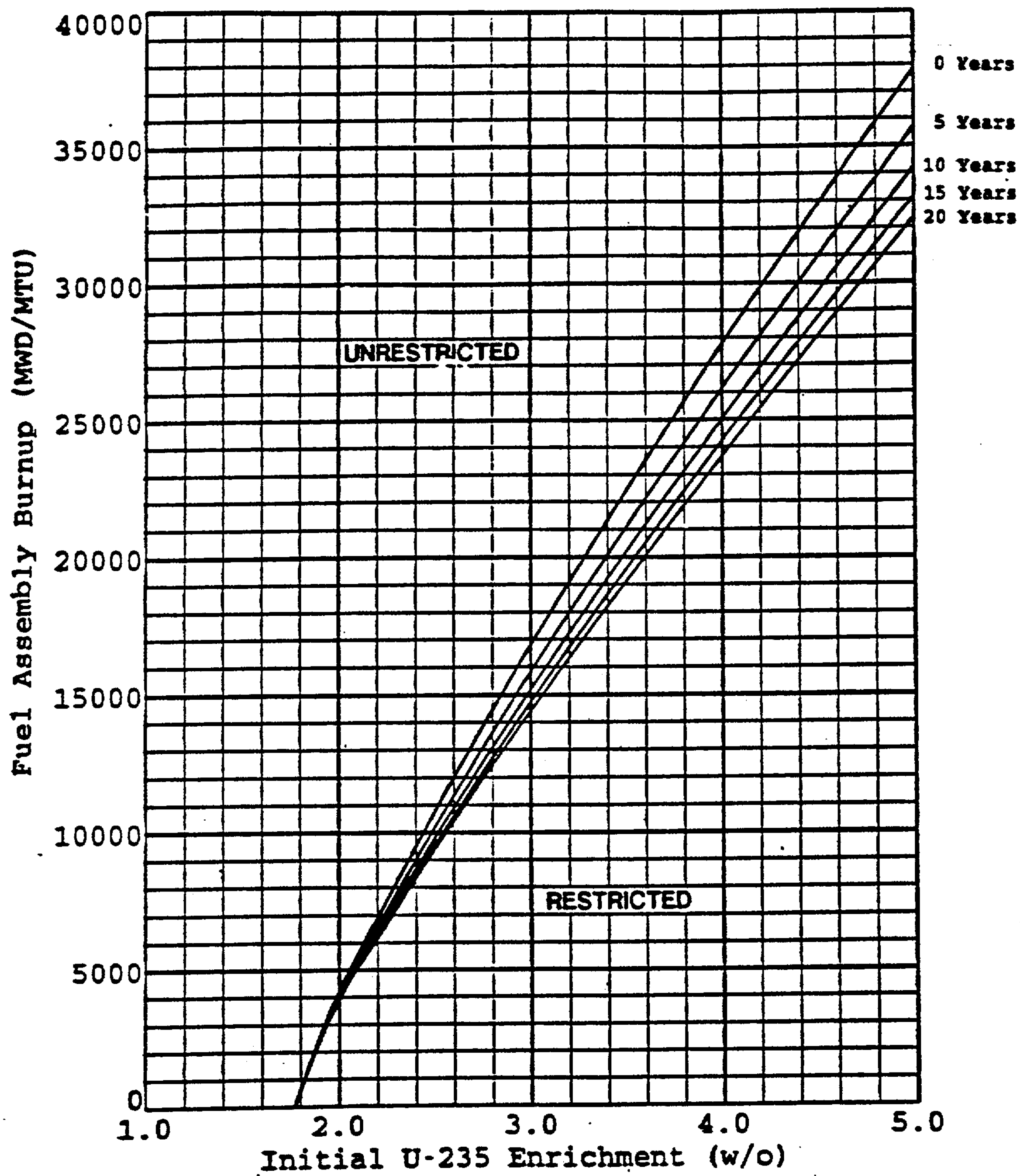


FIGURE TS.3.8-2 Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements - STD Fuel

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY
1. RCS Gross Activity Determination	5/week
2. RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3. RCS Radiochemistry $\bar{E}$ determination	1/6 months(1) (when at power)
4. RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5. RCS Radiochemistry (2)	Monthly
6. RCS Tritium Activity	Weekly
7. RCS Chemistry (Cl*, F*, O2)	5/Week
8. RCS Boron Concentration*(3)	2/Week (4)
9. RWST Boron Concentration	Weekly
10. Boric Acid Tanks Boron Concentration	2/Week
11. Caustic Standpipe NaOH Concentration	Monthly
12. Accumulator Boron Concentration	Monthly
13. Spent Fuel Pit Boron Concentration	Weekly

* Required at all times.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY
14. Secondary Coolant Gross Beta-Gamma activity	Weekly
15. Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)
16. Secondary Coolant Chemistry	
pH	5/week (6)
pH Control Additive	5/week (6)
Sodium	5/week (6)

Notes:

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.
2. To determine activity of corrosion products having a half-life greater than 30 minutes.
3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.
4. The maximum interval between analyses shall not exceed 5 days.
5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.
6. The maximum interval between analyses shall not exceed 3 days.

4.20 Spent Fuel Pool Storage Configuration

Applicability

This surveillance is applicable whenever fuel is stored in the spent fuel pool.

Objective

To verify that fuel assemblies in the spent fuel pool are stored in accordance with the requirements of Specification 3.8.E.1.a.

Specification

A spent fuel pool inventory verification shall be performed within 7 days of the completion of any fuel handling campaign which involves the relocation of fuel assemblies within the spent fuel pool or the addition of fuel assemblies to the spent fuel pool.

5.6 FUEL HANDLING

A. Criticality Consideration

1. The spent fuel storage racks are designed (Reference 1) and shall be maintained with:

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

   b. $K_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 3;

   c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 750 ppm, which includes an allowance for uncertainties as described in Reference 3;

   d. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the unrestricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, may be allowed unrestricted storage in the spent fuel racks; and

   e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the restricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, will be stored in compliance with Figures TS.5.6-1 through TS.5.6-12.

2. The new fuel storage racks are designed (Reference 1) and shall be maintained with:

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

   b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 2; and

   c. $K_{eff} \leq 0.98$ if accidentally filled with a low density moderator which resulted in optimum low density moderation conditions.

3. Fuel will not be inserted into a spent fuel cask in the pool, unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask. The criticality analysis for the TN-40 spent fuel storage cask was based on fresh fuel enriched to 3.85 weight percent U-235.

B. Spent Fuel Storage Structure

The spent fuel storage pool is enclosed with a reinforced concrete building having 12- to 18-inch thick walls and roof (Reference 1). The pool and pool enclosure are Class I (seismic) structures that afford protection against loss of integrity from postulated tornado missiles. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The bottoms of the slots are above the tops of the active fuel in the fuel assemblies which will be stored vertically in specially constructed racks.

D. Spent Fuel Storage Capacity

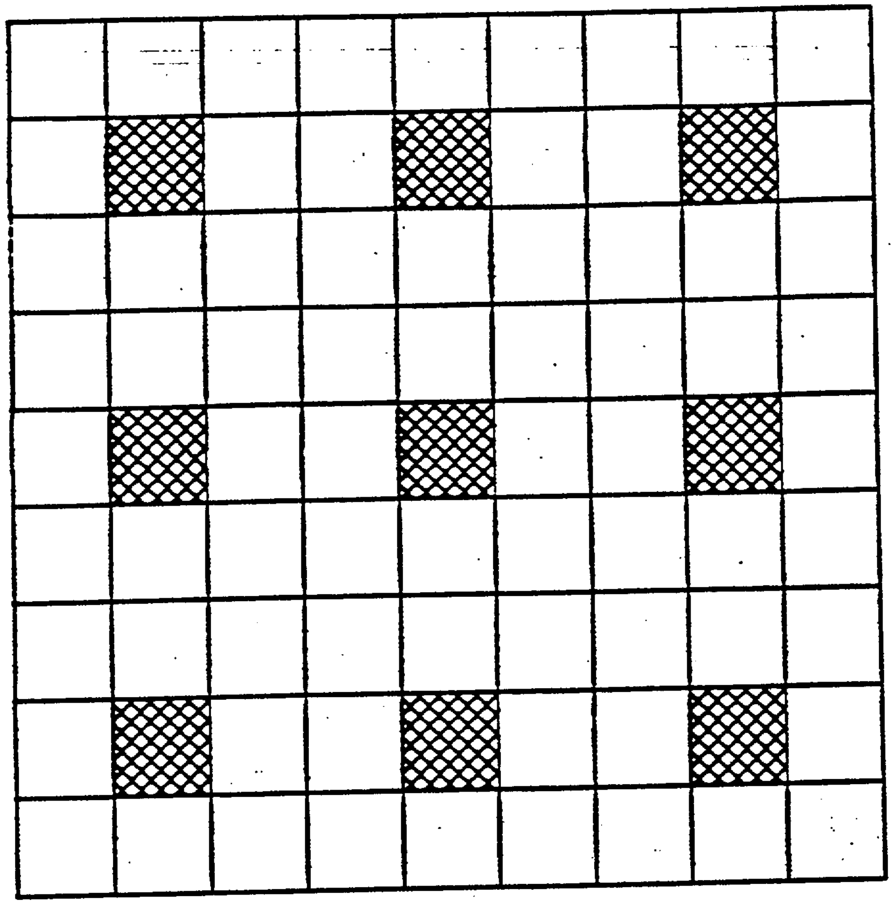
The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, provides up to 1582 storage locations. The southeast corner of the small pool (pool no. 1) also serves as the cask lay down area. During times when the cask is being used, four racks are removed from the small pool. With the four storage racks in the southeast corner of pool 1 removed, a total of 1386 storage locations are provided. To allow insertion of a spent fuel cask, total storage is limited to 1386 assemblies, not including those assemblies which can be returned to the reactor.

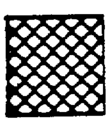
Reference

1. USAR, Section 10.2

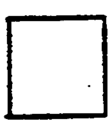
2. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

3. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.



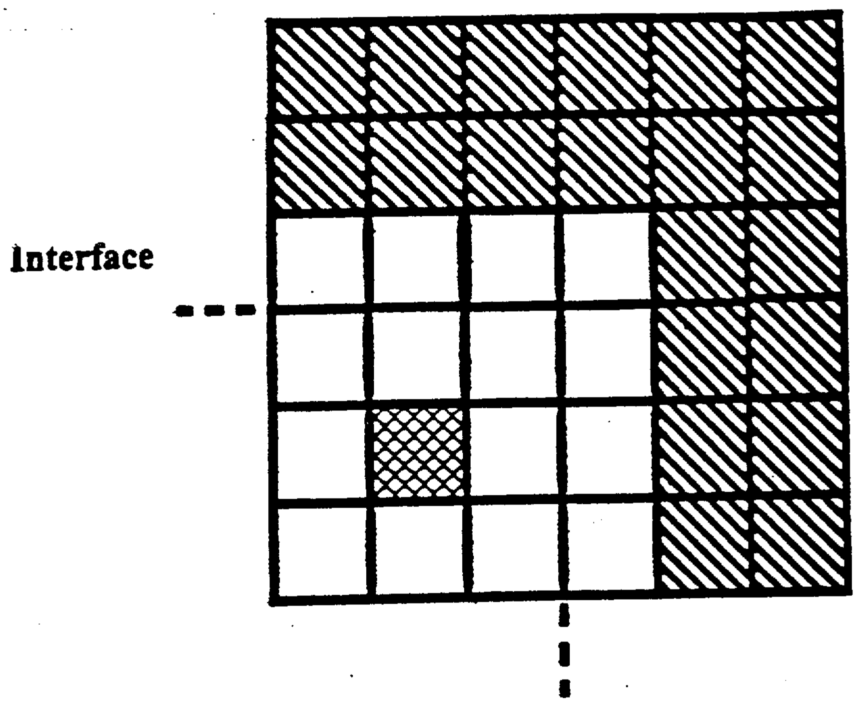


Fresh Fuel: Must be less than or equal to nominal 4.95 w/o ^{235}U
No restrictions on burnup



Burned Fuel: Must satisfy minimum burnup requirements of Figures TS.5.6-3 through TS.5.6-12 depending on number of GAD rods in fresh fuel

FIGURE TS.5.6-1 Spent Fuel Pool Burned/Fresh Checkerboard Cell Layout



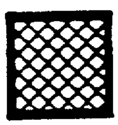
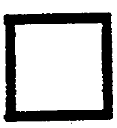
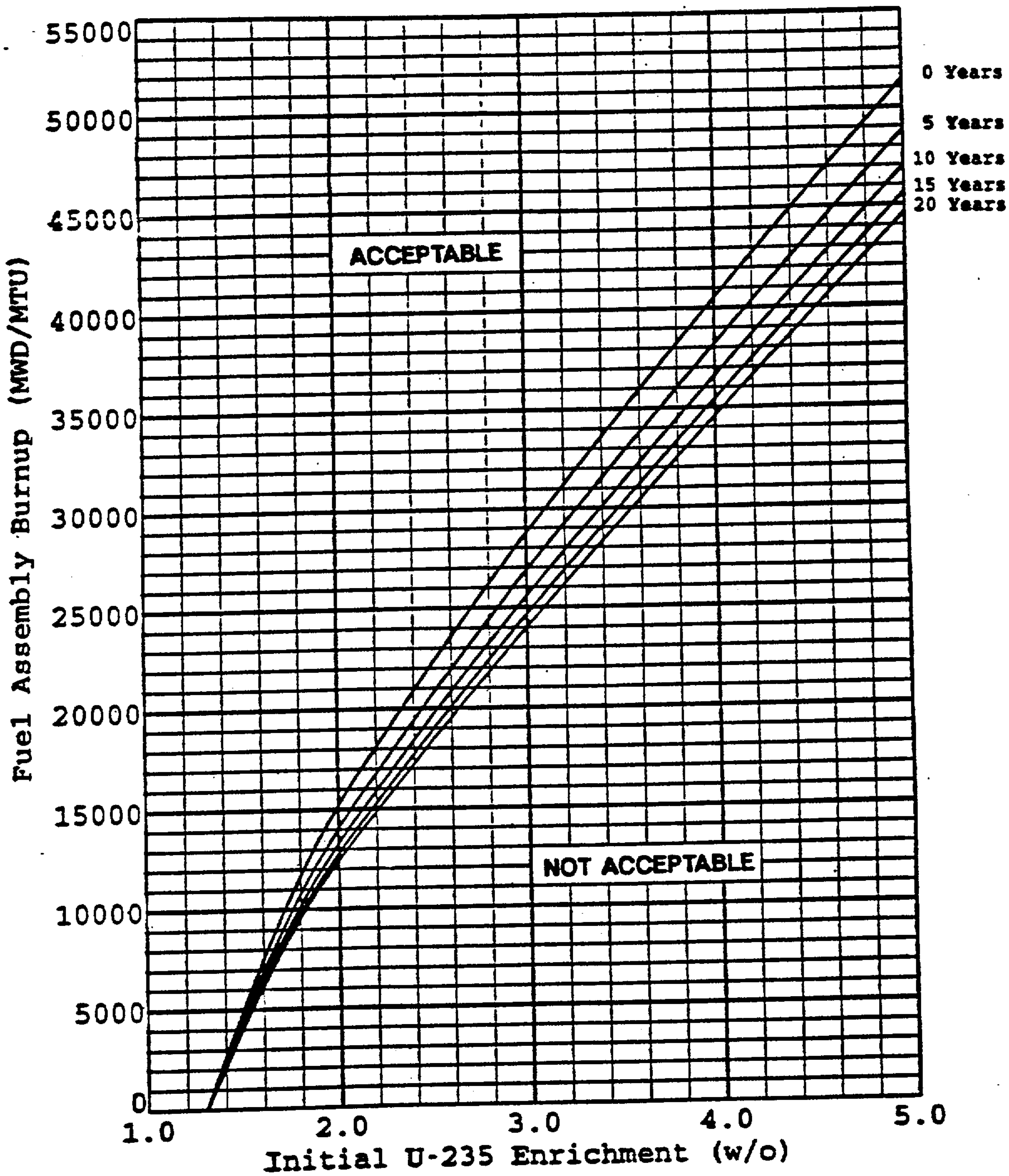
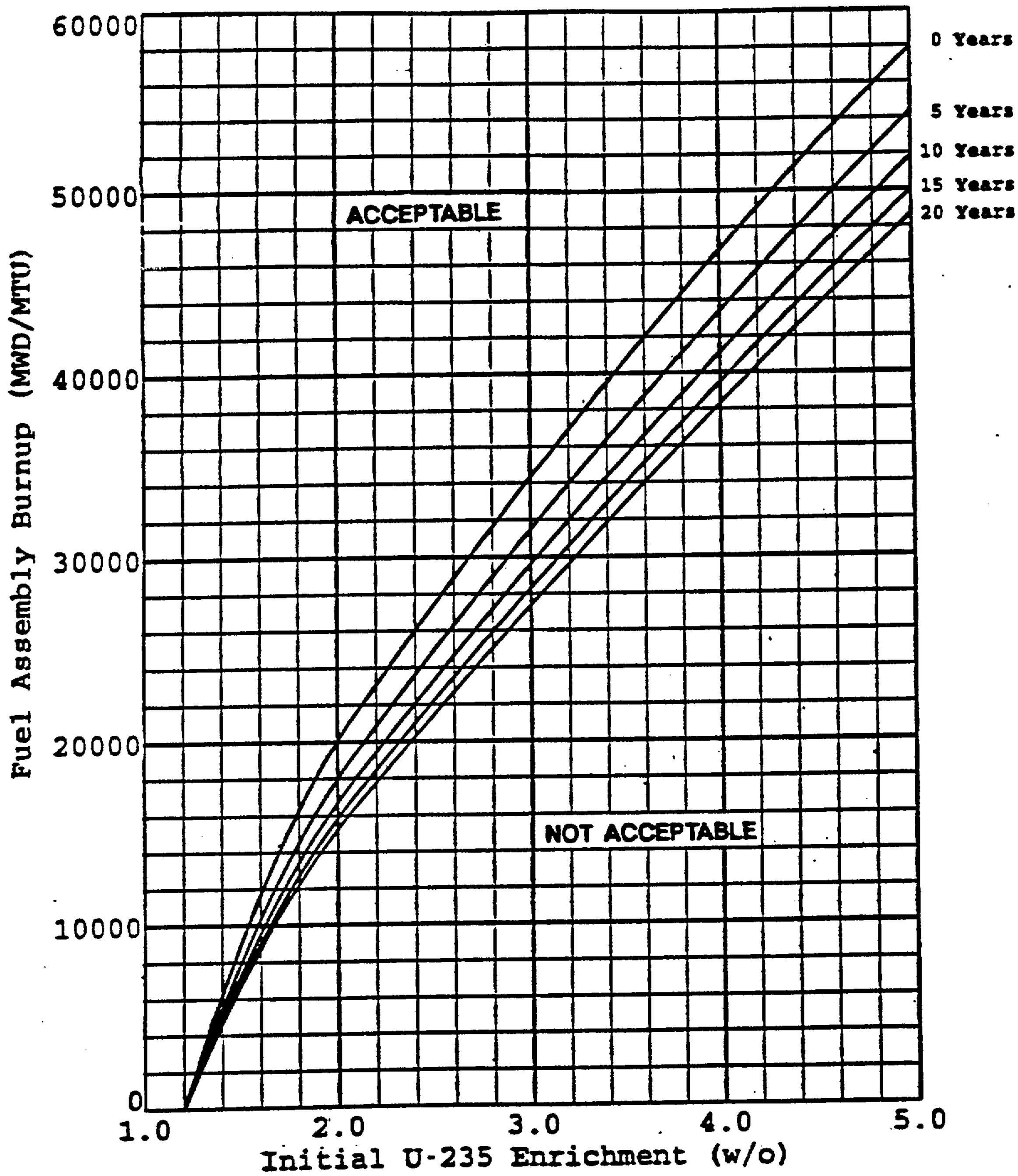
	Fresh Fuel:	Must be less than or equal to nominal 4.95 w/o ^{235}U No restrictions on burnup
	Burned Fuel:	3x3 Checkerboard Region Must satisfy minimum burnup requirements of Figures TS.5.6-3 through TS.5.6-12
	Burned Fuel:	All Cell Unrestricted Region Must satisfy minimum burnup requirements of Figures TS.3.8-1 or TS.3.8-2

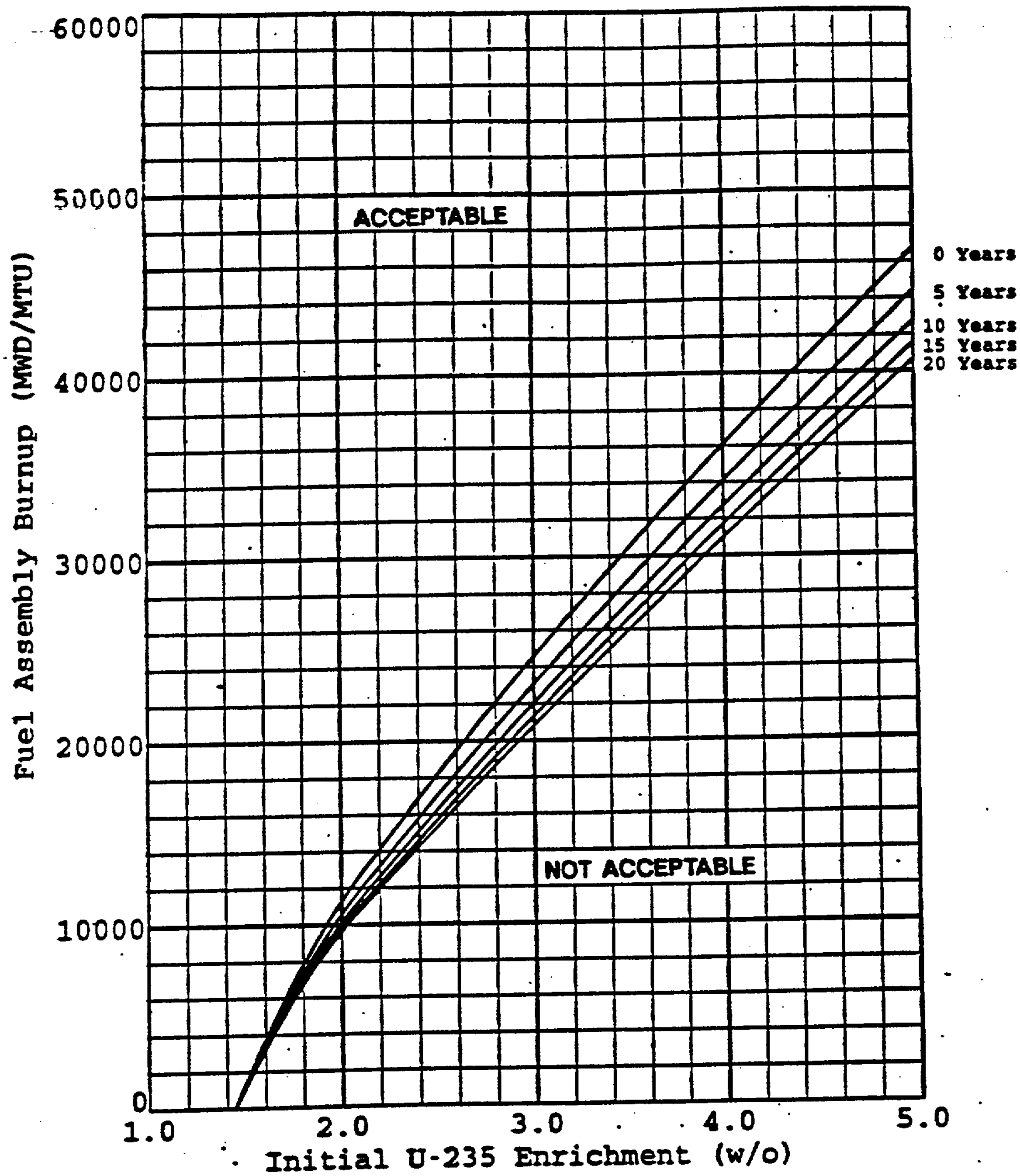
FIGURE TS.5.6-2 Spent Fuel Pool Checkerboard Interface Requirements



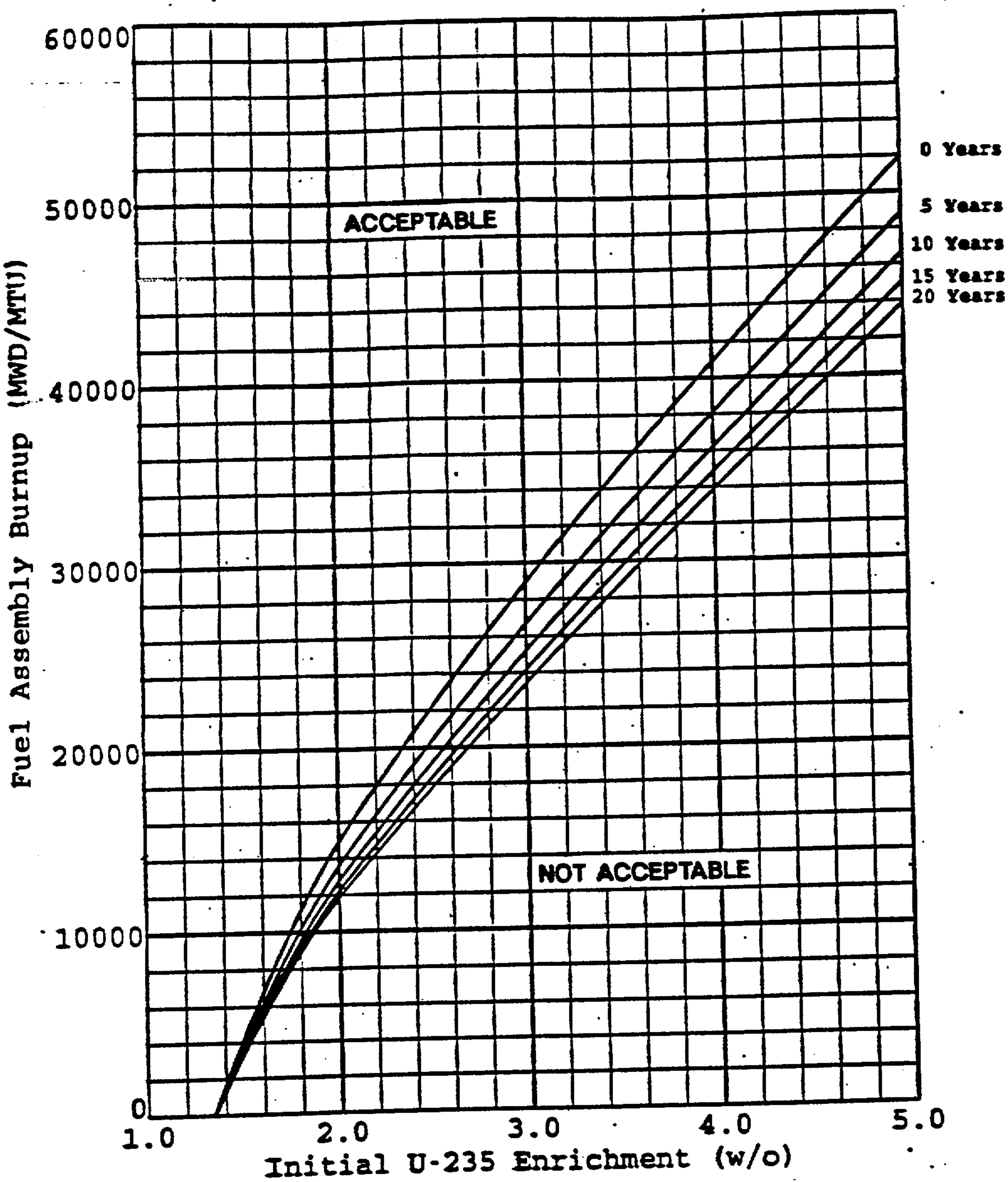
TS.5.6-3 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, No GAD



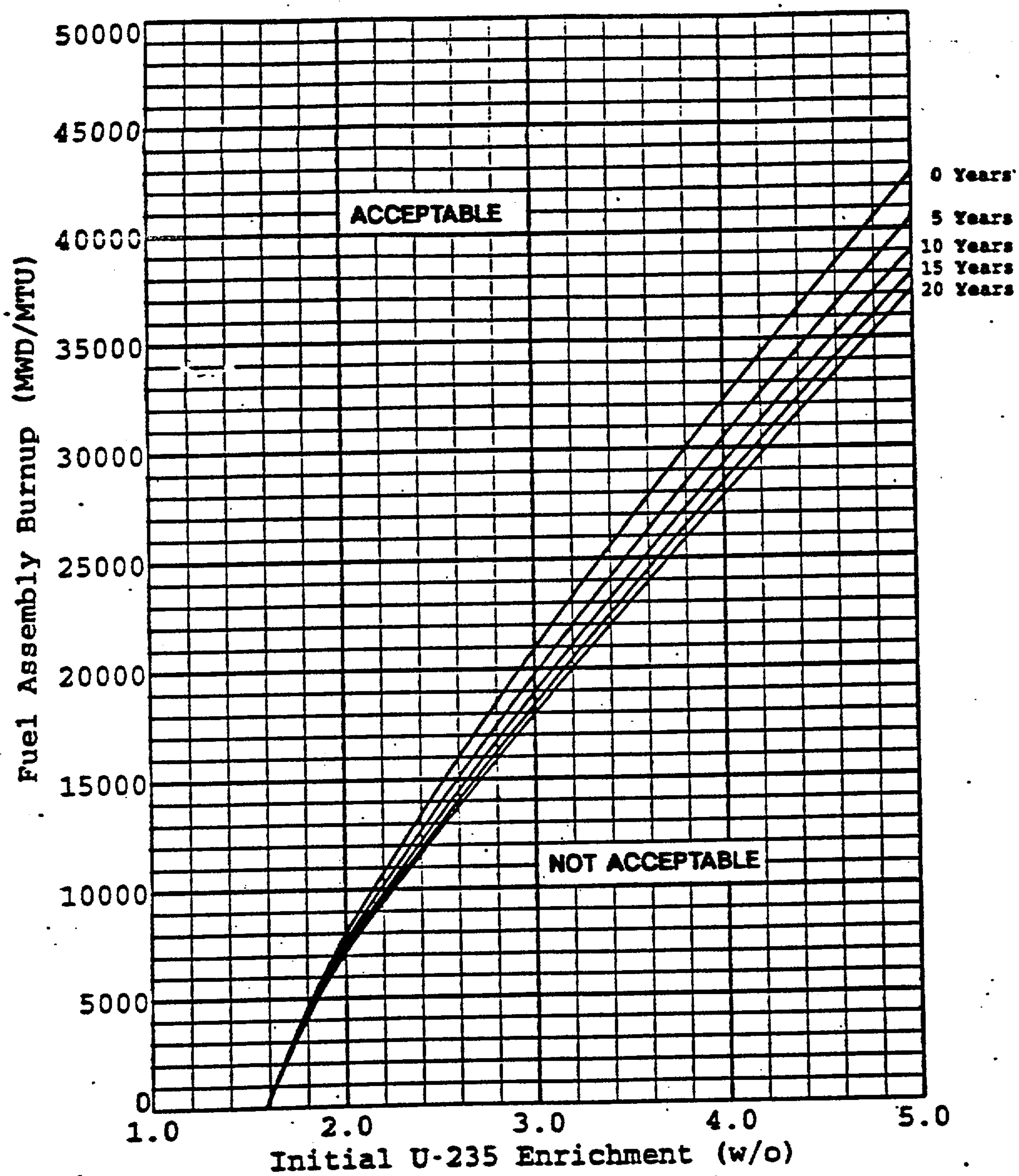
TS.5.6-4 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, No GAD



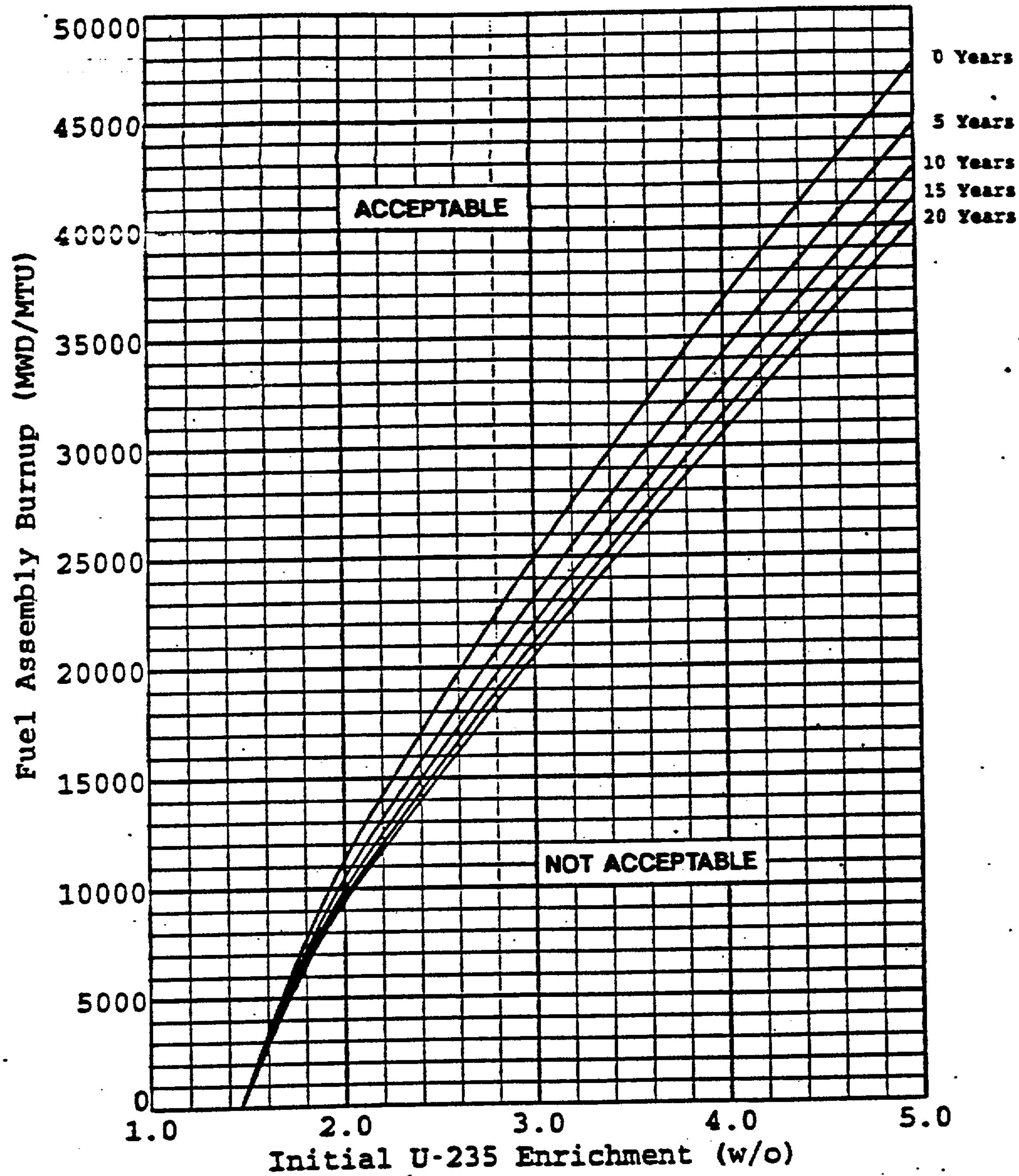
TS.5.6-5 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 4 GAD



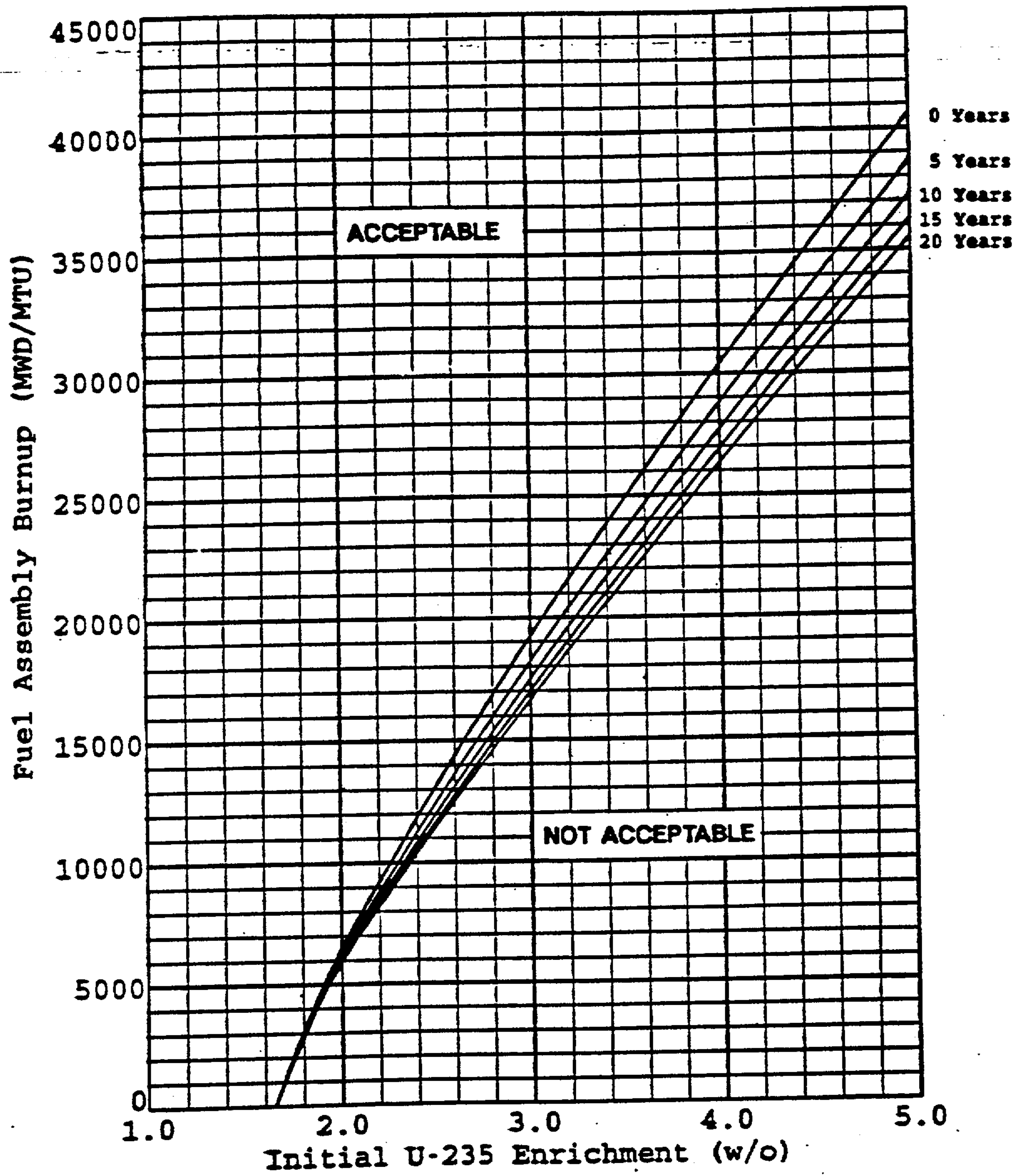
TS.5.6-6 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 4 GAD



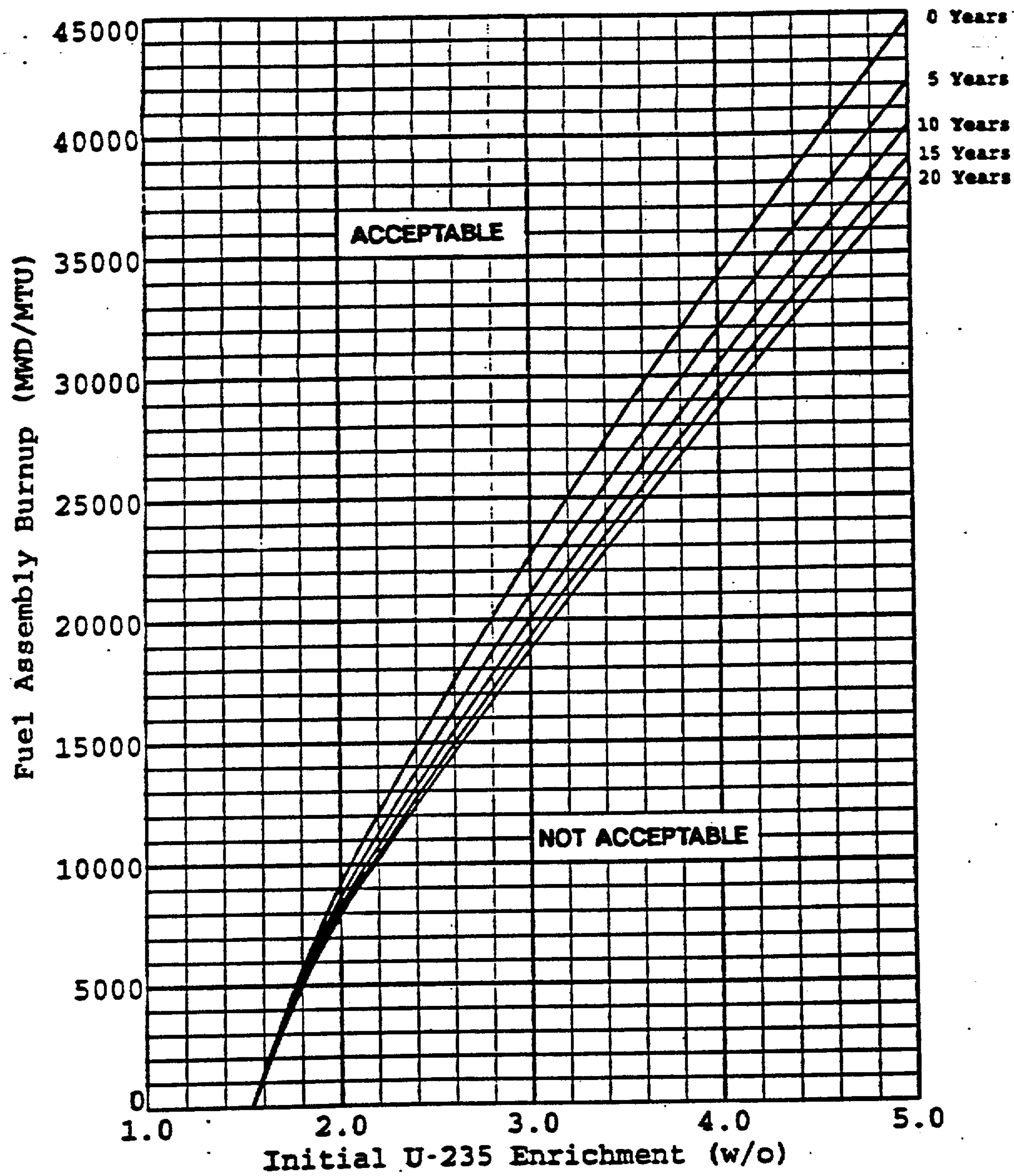
TS.5.6-7 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 8 GAD



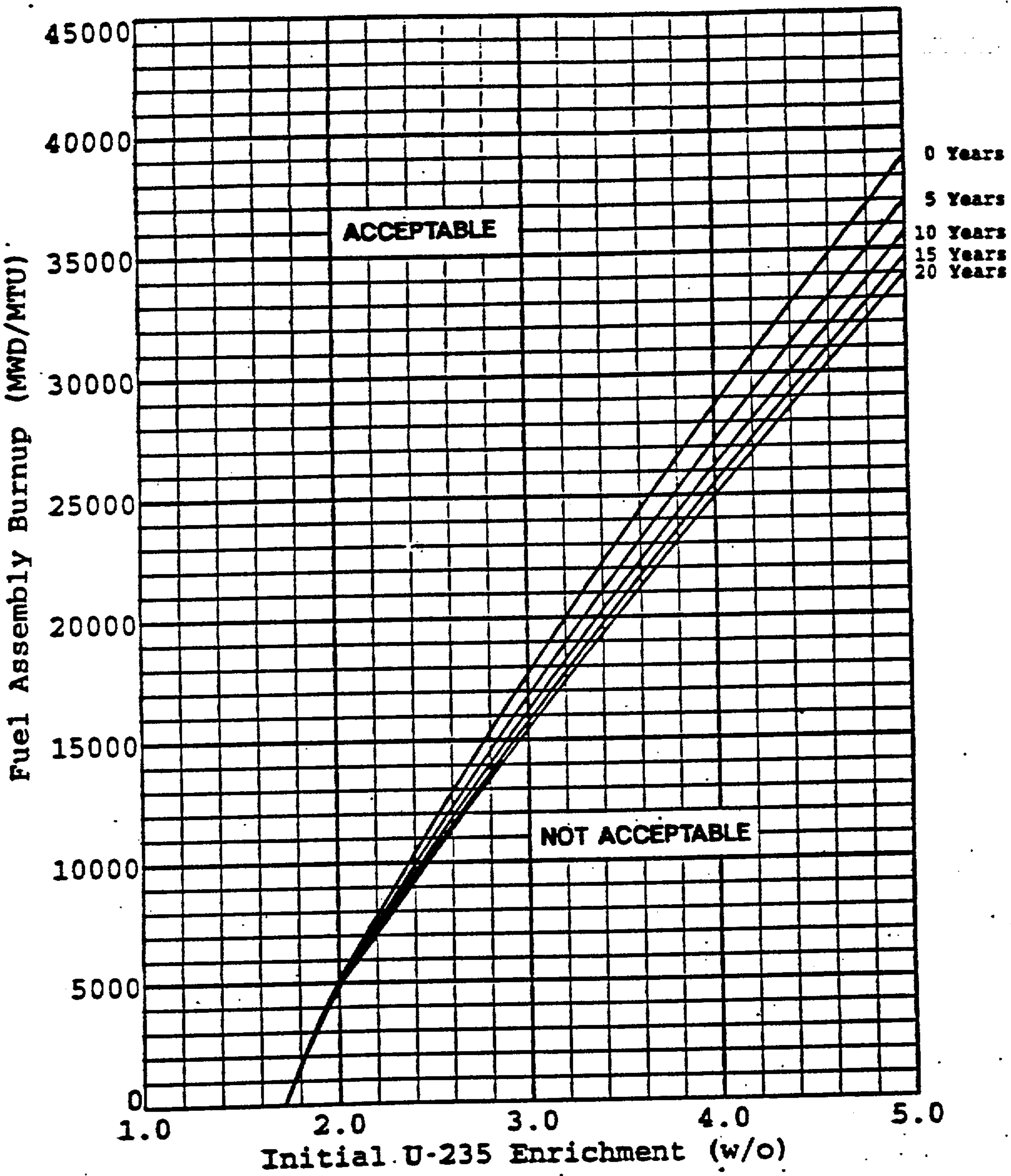
TS.5.6-8 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 8 GAD



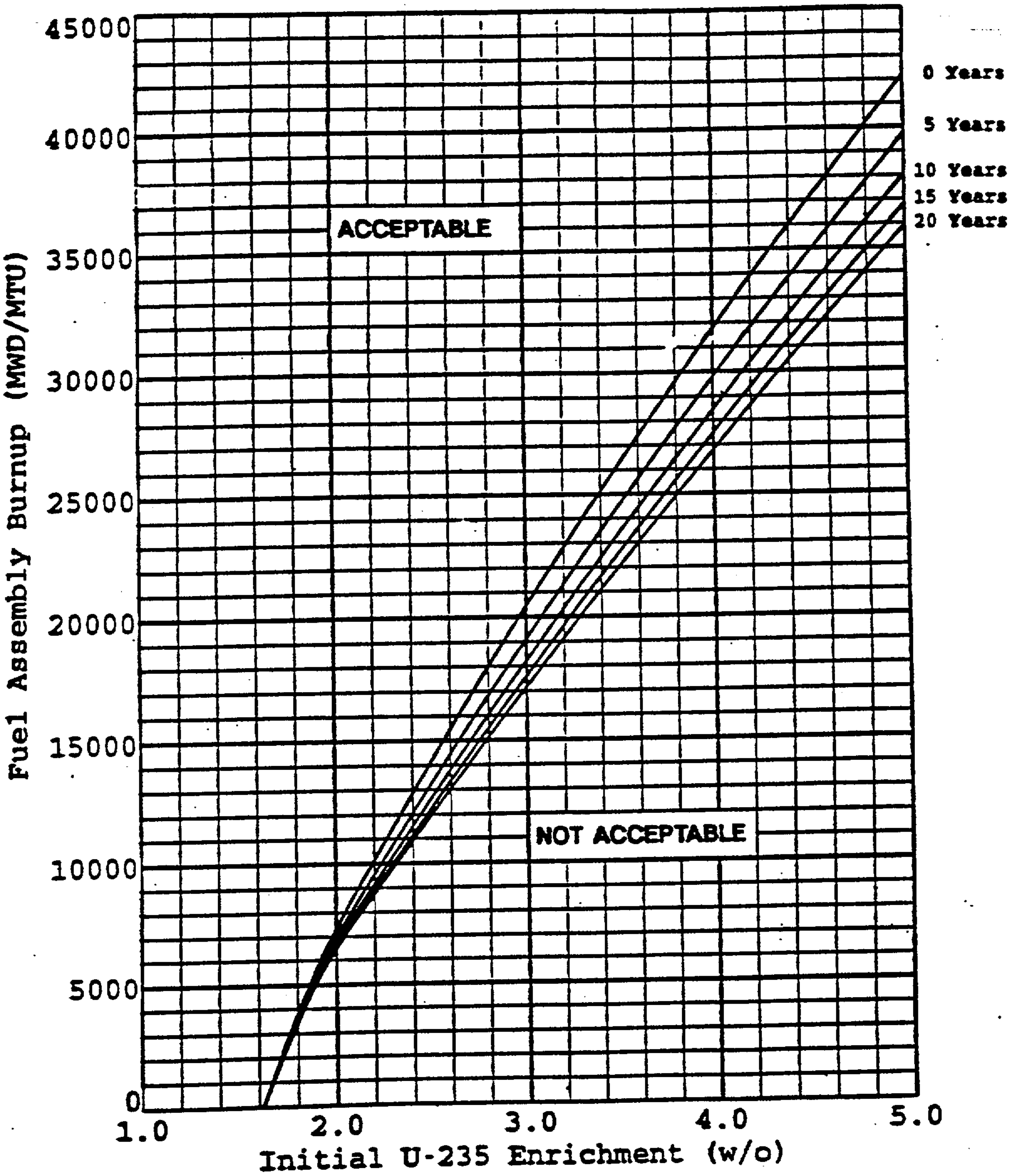
TS.5.6-9 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 12 GAD



TS.5.6-10 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 12 GAD



TS.5.6-11 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 16 or More GAD



TS.5.6-12 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 16 or More GAD

3.8 REFUELING AND FUEL HANDLING

Bases continued

The Spent Fuel Pool Special Ventilation System (Reference 3) is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation System is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

During movement of irradiated fuel assemblies or control rods, a water level of 23 feet is maintained to provide sufficient shielding.

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The water level may also be lowered below 20 feet for upper internals removal/replacement. The basis for these allowance(s) are (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching/unlatching and upper internals removal/replacement the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

The Prairie Island spent fuel storage racks have been analyzed (Reference 8) in accordance with the methodology contained in Reference 5. That methodology ensures that the spent fuel rack multiplication factor, K_{eff}, is less than 0.95 as recommended by ANSI 57.2-1983 (Reference 6) and NRC guidance (Reference 7). The codes, methods and techniques contained in the methodology are used to satisfy this criterion on K_{eff}. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum of 5.0 weight percent U-235 while maintaining $K_{eff} \leq 0.95$ including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool ($K_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing Gadolinium burnable poison.

The Prairie Island specific criticality analysis (Reference 8) utilized the following storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment:

3.8 REFUELING AND FUEL HANDLING

Bases continued

1. The first storage configuration utilizes a checkerboard loading pattern to accommodate new or low burnup fuel with a maximum enrichment of 5.0 wt% U-235. This configuration stores "burned" and "fresh" fuel assemblies in a 3x3 checkerboard pattern as shown in Figure TS.5.6-1. Fuel assemblies stored in "burned" cell locations are selected based on a combination of fuel assembly type, initial enrichment, discharge burnup and decay time (Figures TS.5.6-3 through TS.5.6-12). The criteria for the fuel stored in the "burned" locations is also dependent on the number of rods containing Gadolinium in the center "fresh" fuel assembly. The use of empty cells is also an acceptable option for the "burned" cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.

   Fuel assemblies that fall into the restricted range of Figures TS.3.8-1 or TS.3.8-2 are required to be stored in "fresh" cell locations as shown in Figure TS.5.6-1. The criteria included in Figures TS.3.8-1 and TS.3.8-2 for the selection of fuel assemblies to be stored in the "fresh" cell locations is based on a combination of fuel assembly type, initial enrichment, decay time and discharge burnup.

2. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, can be stored anywhere in the region with no special placement restrictions.

The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:

1. separated by a vacant row of cells, or

2. the interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure TS.5.6-2).

Specifications 3.8.E.1, 5.6.A.1.d and 5.6.A.1.e ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the Prairie Island spent fuel rack criticality analysis (Reference 8).

The Prairie Island spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, and the Exxon fuel assembly types in storage in the Prairie Island spent fuel pool. The OFA designation on the figures in Sections 3.8 and 5.6.A bound all of the Westinghouse OFA and Vantage Plus fuel assemblies at Prairie Island. The STD designation on the figures in Sections 3.8 and 5.6.A bound all of the Westinghouse STD and Exxon fuel assemblies at Prairie Island.

Bases continued

Most accident conditions in the spent fuel pool will not result in an increase in K_{eff} of the racks in either of the two storage configurations. Examples of those accident conditions which will not result in an increase in K_{eff} are a fuel assembly drop on the top of the racks, a fuel assembly drop between rack modules and wall (rack design precludes this condition), and a drop or placement of a fuel assembly into the cask loading area of the small pool. However, two accidents can be postulated which could increase reactivity. The first postulated accident would be a loss of the fuel pool cooling system and the second would be a misload of a fuel assembly into a cell for which the restrictions on location, enrichment, burnup, decay time or Gadolinium credit are not satisfied.

For an occurrence of these postulated accident conditions, the double contingency principle of ANSI/ANS-8.1-1983 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 750 ppm required to maintain K_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed (Reference 8) to determine the amount of soluble boron required to offset the highest reactivity increase caused by either of these postulated accidents and to maintain K_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 1300 ppm was adequate to mitigate these postulated criticality related accidents and to maintain K_{eff} less than or equal to 0.95. Specification 3.8.E.2 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.8.E.2 was chosen to be consistent with the boron concentration limit required by Specification 3.8.B.1.c for a spent fuel cask containing fuel.

Specification 5.6.A.1.c requires that the spent fuel rack K_{eff} be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 K_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

When the requirements of Specification 3.8.E.1.a are not met, immediate action must be taken to move any non complying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

3.8 REFUELING AND FUEL HANDLING

Bases continued

When the concentration of boron in the spent fuel pool is less than required by Specification 3.8.E.2.a, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. The suspension of fuel movement is not intended to preclude movement of a fuel assembly to a safe position.

References

1. USAR, Section 10.2.1.2

2. USAR, Section 14.5.1

3. USAR, Section 10.3.7

4. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

5. WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996.

6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.

7. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.

8. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.

4.20 Spent Fuel Pool Storage Configuration

Bases

This surveillance verifies that the fuel assemblies in the spent fuel storage racks are stored in accordance with the requirements of Specifications 3.8.E.1, 5.6.A.1.d and 5.6.A.1.e.

The surveillance is required to be completed within 7 days after the completion of any fuel handling campaign which involves the relocation of fuel assemblies within the spent fuel pool or the addition of fuel assemblies to the spent fuel pool. The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all of the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

It is not the intent of this surveillance to require the completion of a spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.

The 7 day allowance for completion of this surveillance provides adequate time for the completion of a spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the spent fuel pool. If a fuel assembly is misloaded during a fuel handling campaign, the minimum boron concentration required by Specification 3.8.E.2 will ensure that the spent fuel rack K_{eff} remains within limits until the spent fuel pool inventory verification is performed.

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company (the licensee) dated July 28, 1995, as revised February 21, 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. DPR-60 is hereby amended to read as follows:

Technical Specifications

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

3. This license amendment is effective as of the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel

Beth A. Wetzel, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 12, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

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

REMOVE	INSERT
TS-vi	TS-vi
TS-xi	TS-xi
TS-xiii	TS-xiii
TS.3.8-4	TS.3.8-4
TS.3.8-5	--
Figure TS.3.8-1	Figure TS.3.8-1
--	Figure TS.3.8-2
Table TS.4.1-2B (Page 1 of 2)	Table TS.4.1-2B (Page 1 of 2)
Table TS.4.1-2B (Page 2 of 2)	Table TS.4.1-2B (Page 2 of 2)
--	TS.4.20-1
TS.5.6-1	TS.5.6-1
TS.5.6-3	TS.5.6-3
Figure TS.5.6-1	Figure TS.5.6-1
Figure TS.5.6-2	Figure TS.5.6-2
--	Figure TS.5.6-3
--	Figure TS.5.6-4
--	Figure TS.5.6-5
--	Figure TS.5.6-6
--	Figure TS.5.6-7
--	Figure TS.5.6-8
--	Figure TS.5.6-9
--	Figure TS.5.6-10
--	Figure TS.5.6-11
--	Figure TS.5.6-12
B.3.8-2	B.3.8-2
B.3.8-3	B.3.8-3
B.3.8-4	B.3.8-4
--	B.3.8-5
--	B.4.20-1

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APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

3.8.C. Small Spent Fuel Pool Restrictions

No more than 45 recently discharged assemblies shall be located in the small pool (pool No. 1).

D. Spent Fuel Pool Special Ventilation System

1. Both trains of the Spent Fuel Pool Special Ventilation System shall be OPERABLE at all times (except as specified in 3.8.D.2 and 3.8.D.3 below).

2. With one train of the Spent Fuel Pool Special Ventilation System inoperable, fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure) are permissible during the following 7 days, provided the redundant train is demonstrated OPERABLE prior to proceeding with those operations.

3. With both trains of the Spent Fuel Pool Special Ventilation System inoperable, suspend all fuel handling operations and crane operations with loads over spent fuel (inside the spent fuel pool enclosure).

4. The provisions of specification 3.0.C are not applicable.

E. Spent Fuel Pool Storage

1. Fuel Assembly Storage

   a. The combination of initial enrichment, burnup and decay time of each spent fuel assembly stored in the spent fuel pool shall be within the unrestricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, or fuel assemblies shall be stored in accordance with Specification 5.6.A.1.e.

   b. If the requirements of 3.8.E.1.a are not met, immediately initiate action to move any noncomplying fuel assembly to an acceptable location.

   c. The provisions of Specification 3.0.C are not applicable.

2. Spent Fuel Pool Boron Concentration

   a. The spent fuel pool boron concentration shall be ≥ 1,800 ppm when fuel assemblies are stored in the spent fuel pool.

   b. If the spent fuel pool boron concentration is not within limit, then immediately:

      1. Suspend movement of fuel assemblies in the spent fuel pool, and

      2. Initiate action to restore spent fuel pool boron concentration to within limit.

   c. The provisions of Specification 3.0.C are not applicable.

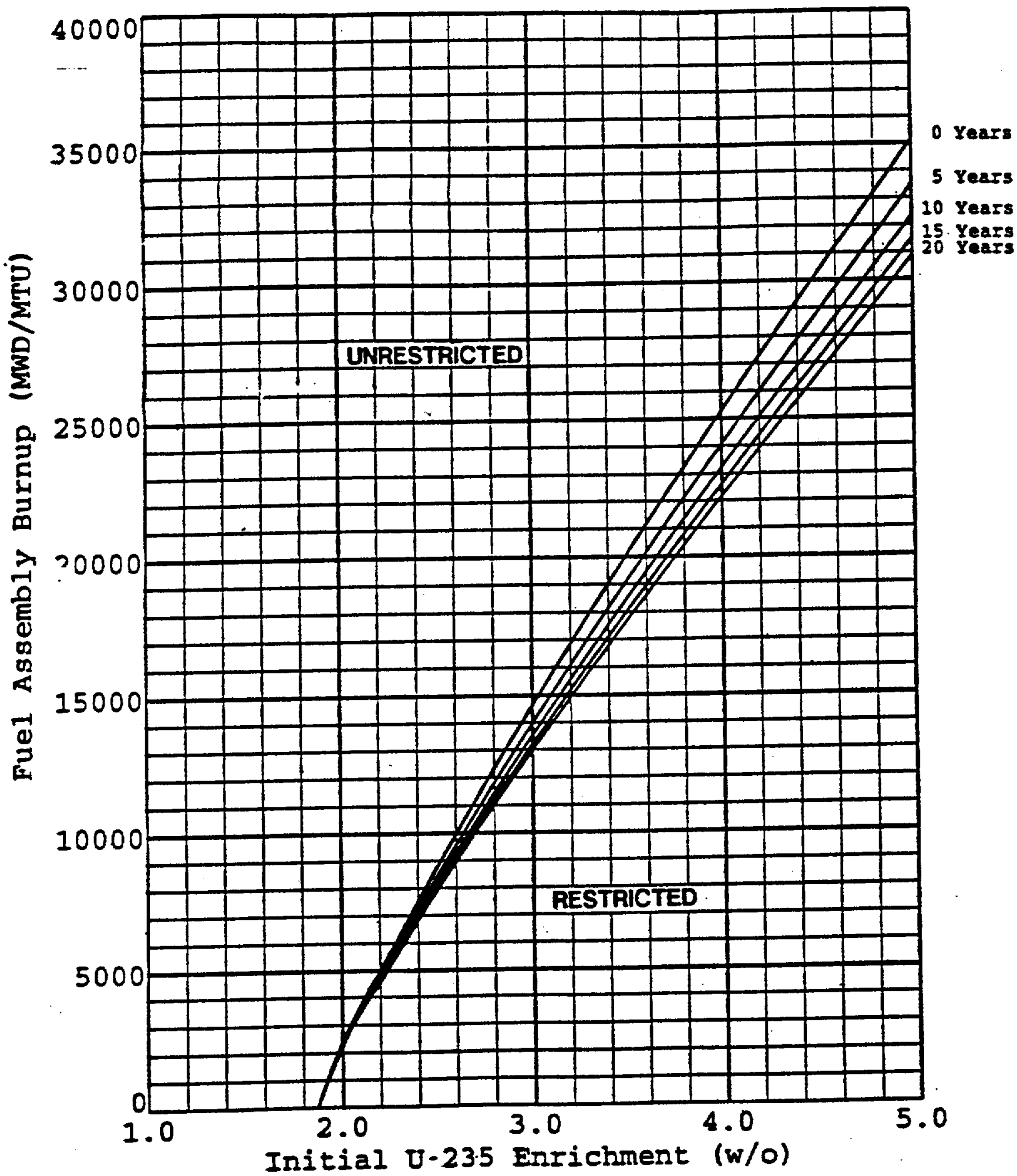


FIGURE TS.3.8-1 **Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements - OFA Fuel**

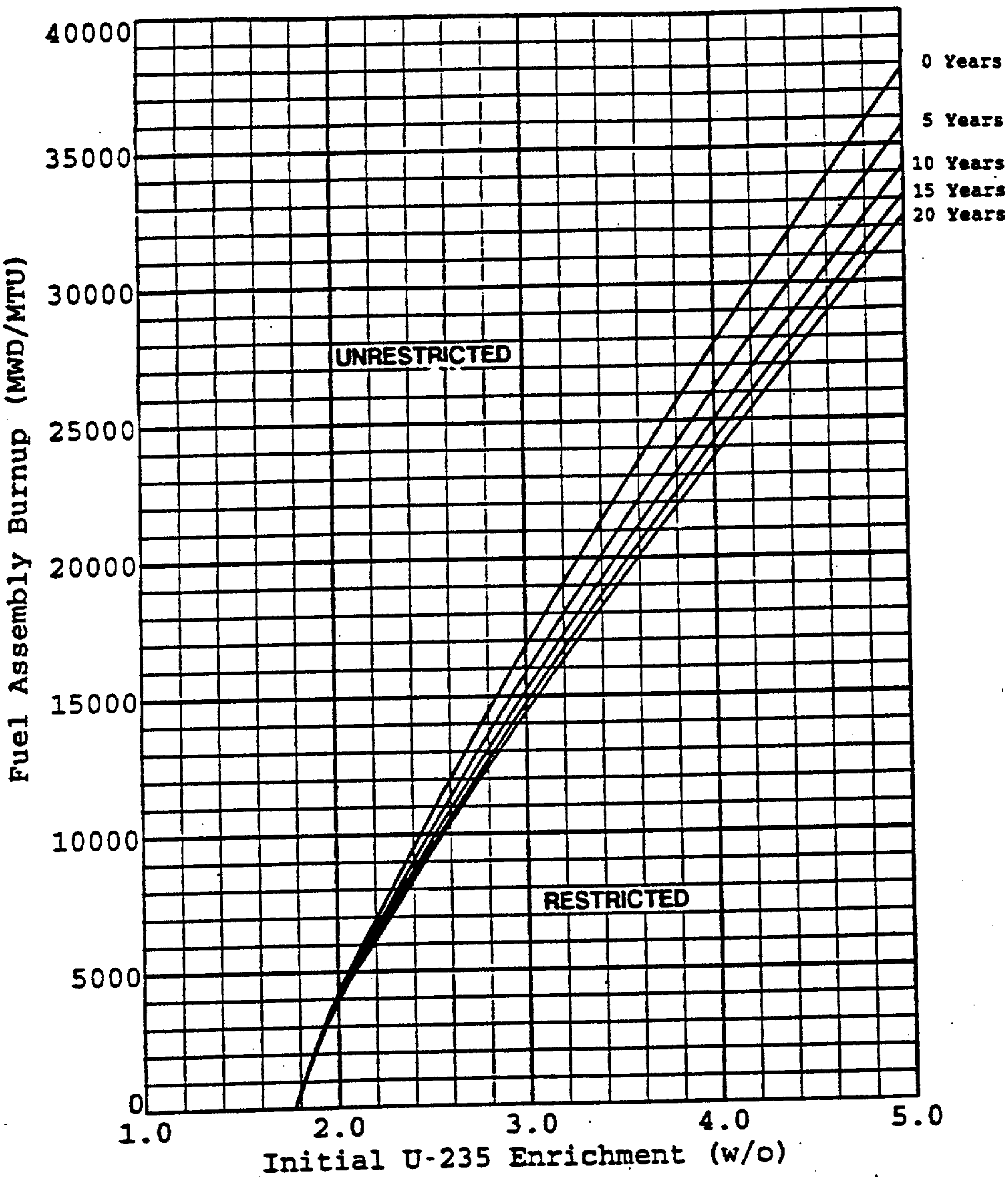


FIGURE TS.3.8-2 Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements - STD Fuel

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY
1. RCS Gross Activity Determination	5/week
2. RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3. RCS Radiochemistry $\bar{E}$ determination	1/6 months(1) (when at power)
4. RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5. RCS Radiochemistry (2)	Monthly
6. RCS Tritium Activity	Weekly
7. RCS Chemistry (Cl*, F*, O2)	5/Week
8. RCS Boron Concentration*(3)	2/Week (4)
9. RWST Boron Concentration	Weekly
10. Boric Acid Tanks Boron Concentration	2/Week
11. Caustic Standpipe NaOH Concentration	Monthly
12. Accumulator Boron Concentration	Monthly
13. Spent Fuel Pit Boron Concentration	Weekly

* Required at all times.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY
14. Secondary Coolant Gross Beta-Gamma activity	Weekly
15. Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)
16. Secondary Coolant Chemistry	
pH	5/week (6)
pH Control Additive	5/week (6)
Sodium	5/week (6)

Notes:

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

2. To determine activity of corrosion products having a half-life greater than 30 minutes.

3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.

4. The maximum interval between analyses shall not exceed 5 days.

5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.

6. The maximum interval between analyses shall not exceed 3 days.

4.20 Spent Fuel Pool Storage Configuration

Applicability

This surveillance is applicable whenever fuel is stored in the spent fuel pool.

Objective

To verify that fuel assemblies in the spent fuel pool are stored in accordance with the requirements of Specification 3.8.E.1.a.

Specification

A spent fuel pool inventory verification shall be performed within 7 days of the completion of any fuel handling campaign which involves the relocation of fuel assemblies within the spent fuel pool or the addition of fuel assemblies to the spent fuel pool.

5.6 FUEL HANDLING

A. Criticality Consideration

1. The spent fuel storage racks are designed (Reference 1) and shall be maintained with:

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

    b. K_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 3;

    c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 750 ppm, which includes an allowance for uncertainties as described in Reference 3;

    d. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the unrestricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, may be allowed unrestricted storage in the spent fuel racks; and

    e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the restricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, will be stored in compliance with Figures TS.5.6-1 through TS.5.6-12.

2. The new fuel storage racks are designed (Reference 1) and shall be maintained with:

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

    b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 2; and

    c. $K_{eff} \leq 0.98$ if accidentally filled with a low density moderator which resulted in optimum low density moderation conditions.

3. Fuel will not be inserted into a spent fuel cask in the pool, unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask. The criticality analysis for the TN-40 spent fuel storage cask was based on fresh fuel enriched to 3.85 weight percent U-235.

B. Spent Fuel Storage Structure

The spent fuel storage pool is enclosed with a reinforced concrete building having 12- to 18-inch thick walls and roof (Reference 1). The pool and pool enclosure are Class I (seismic) structures that afford protection against loss of integrity from postulated tornado missiles. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The bottoms of the slots are above the tops of the active fuel in the fuel assemblies which will be stored vertically in specially constructed racks.

D. <u>Spent Fuel Storage Capacity</u>

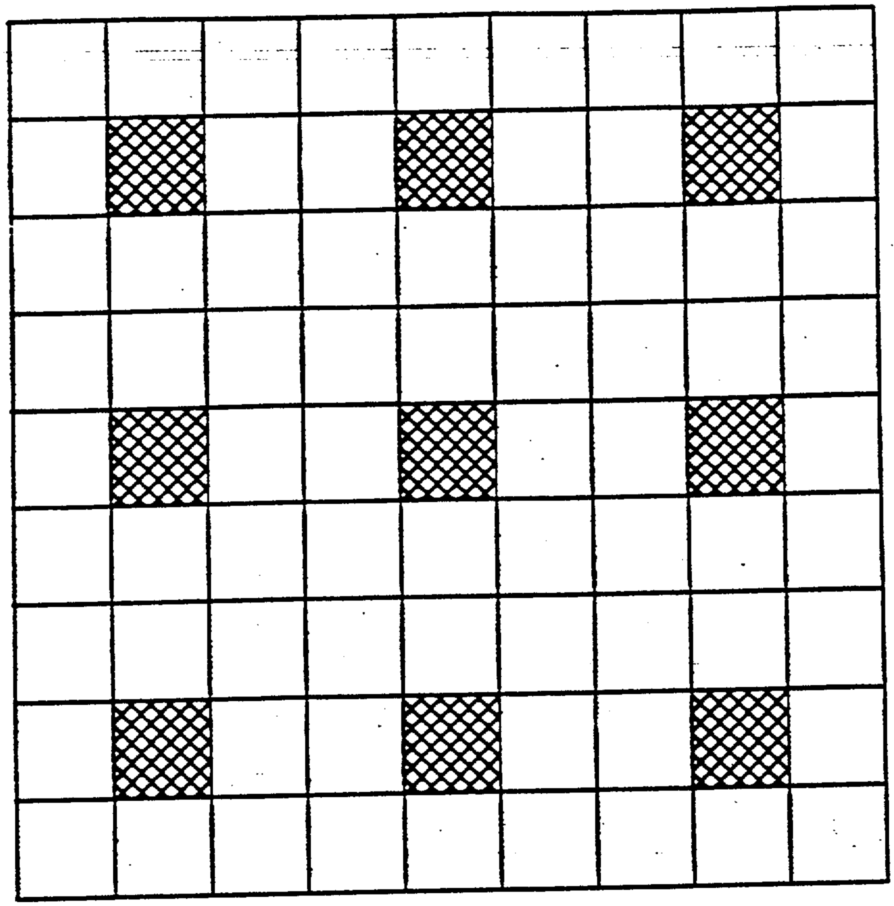
The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, provides up to 1582 storage locations. The southeast corner of the small pool (pool no. 1) also serves as the cask lay down area. During times when the cask is being used, four racks are removed from the small pool. With the four storage racks in the southeast corner of pool 1 removed, a total of 1386 storage locations are provided. To allow insertion of a spent fuel cask, total storage is limited to 1386 assemblies, not including those assemblies which can be returned to the reactor.

<u>Reference</u>

1. USAR, Section 10.2

2. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

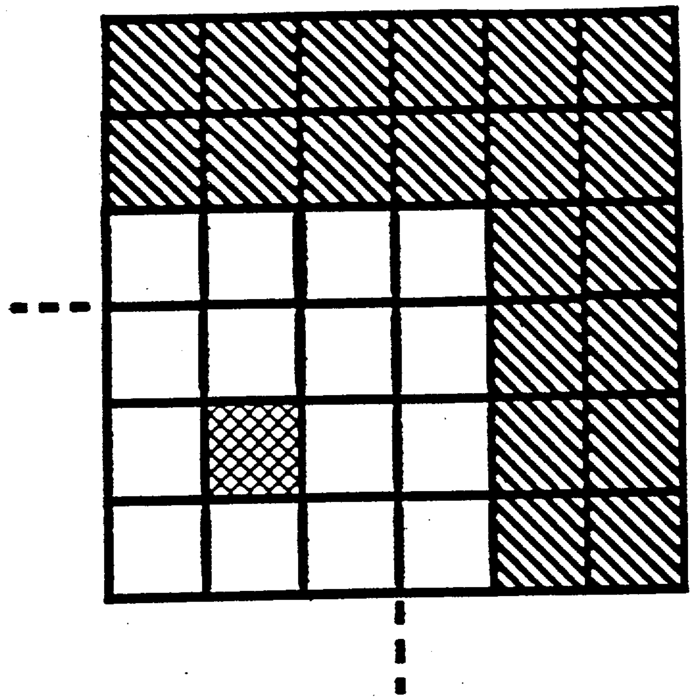
3. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.



(cross-hatched)	**Fresh Fuel:**	**Must be less than or equal to nominal 4.95 w/o ^{235}U** **No restrictions on burnup**
(blank)	**Burned Fuel:**	**Must satisfy minimum burnup requirements of Figures TS.5.6-3 through TS.5.6-12 depending on number of GAD rods in fresh fuel**

FIGURE TS.5.6-1 Spent Fuel Pool Burned/Fresh Checkerboard Cell Layout

Interface



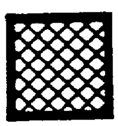
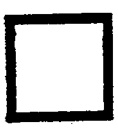
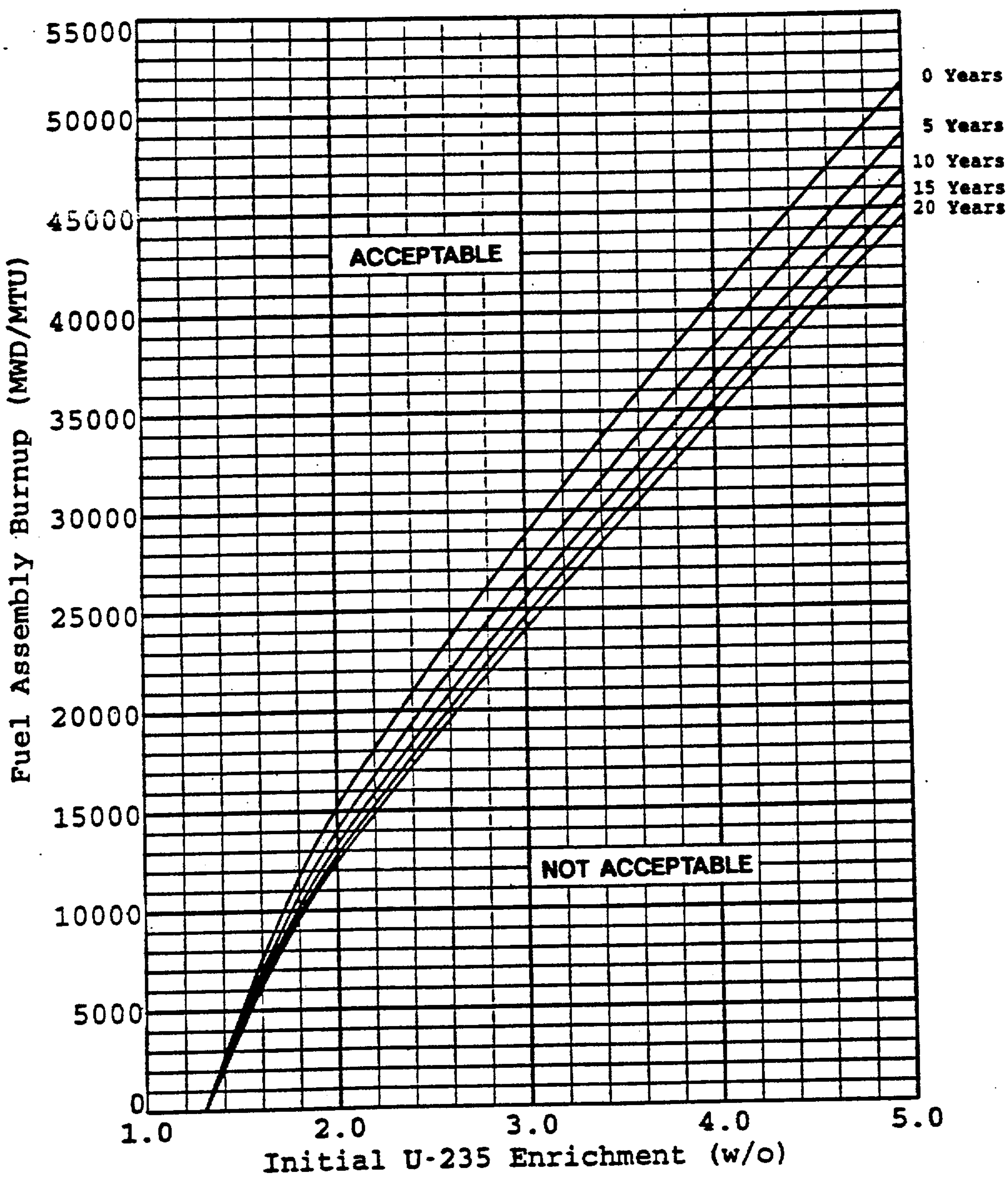
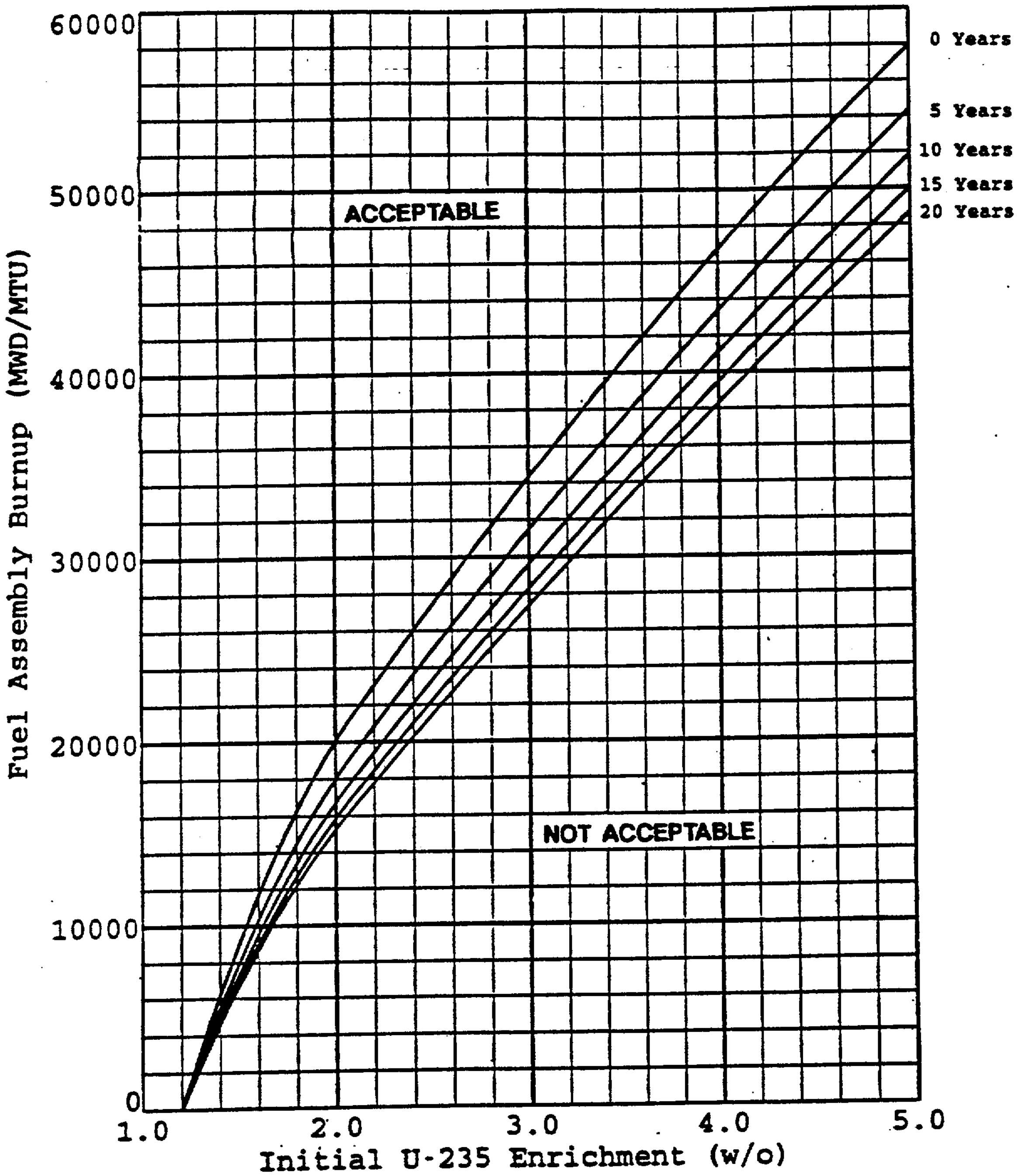
	Fresh Fuel:	Must be less than or equal to nominal 4.95 w/o ^{235}U No restrictions on burnup
	Burned Fuel:	3x3 Checkerboard Region Must satisfy minimum burnup requirements of Figures TS.5.6-3 through TS.5.6-12
	Burned Fuel:	All Cell Unrestricted Region Must satisfy minimum burnup requirements of Figures TS.3.8-1 or TS.3.8-2

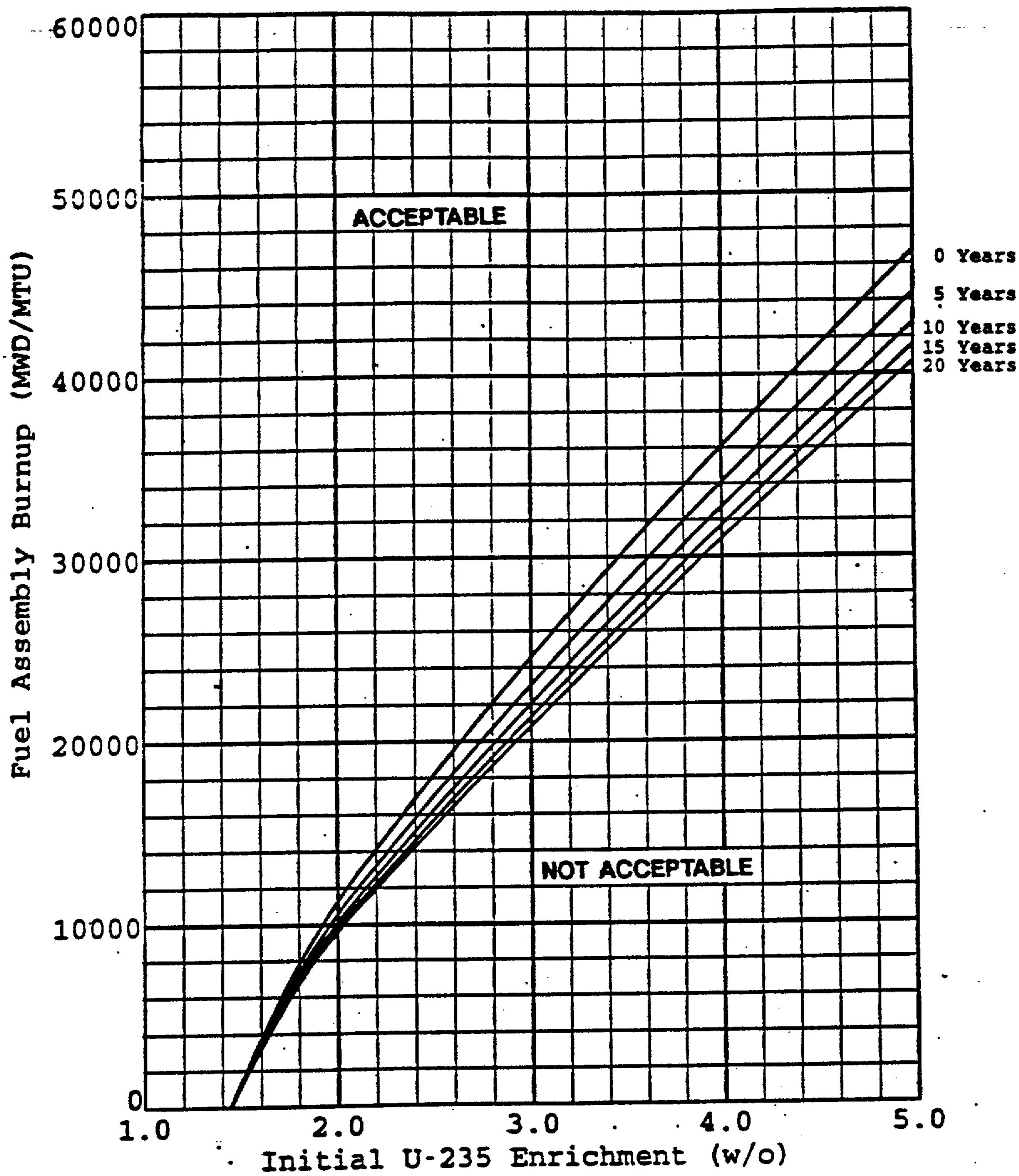
FIGURE TS.5.6-2 Spent Fuel Pool Checkerboard Interface Requirements



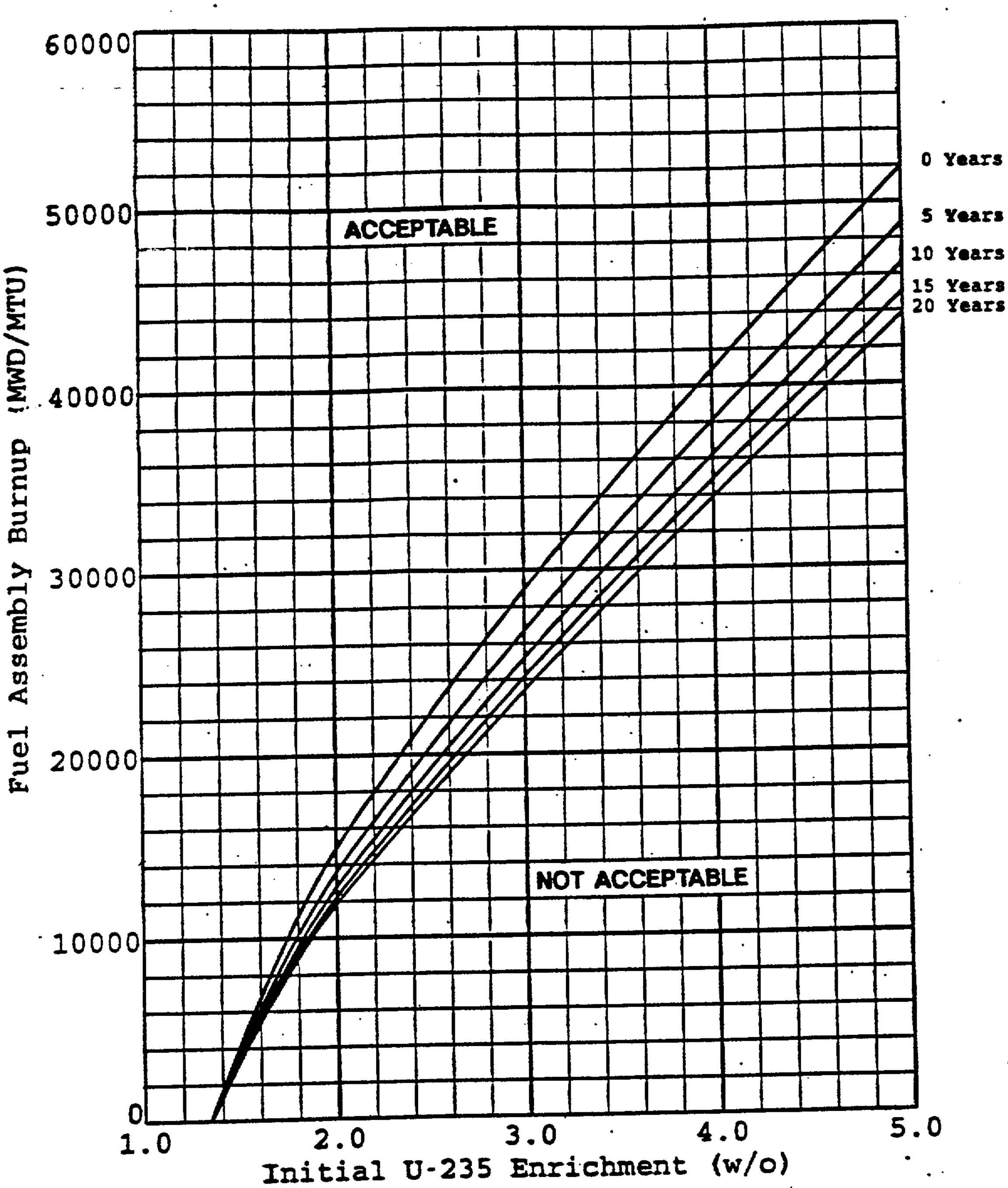
TS.5.6-3 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, No GAD



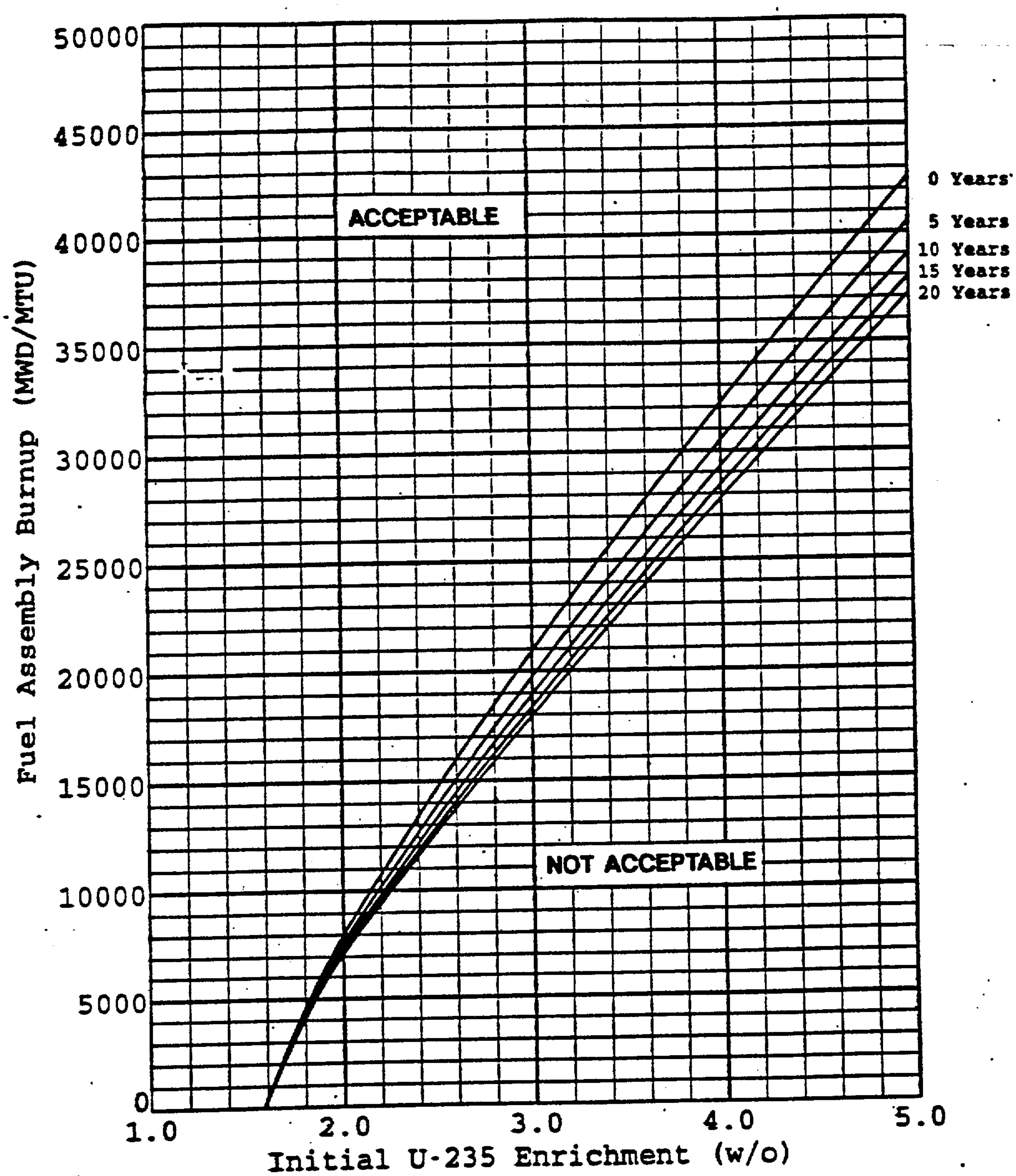
TS.5.6-4 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, No GAD



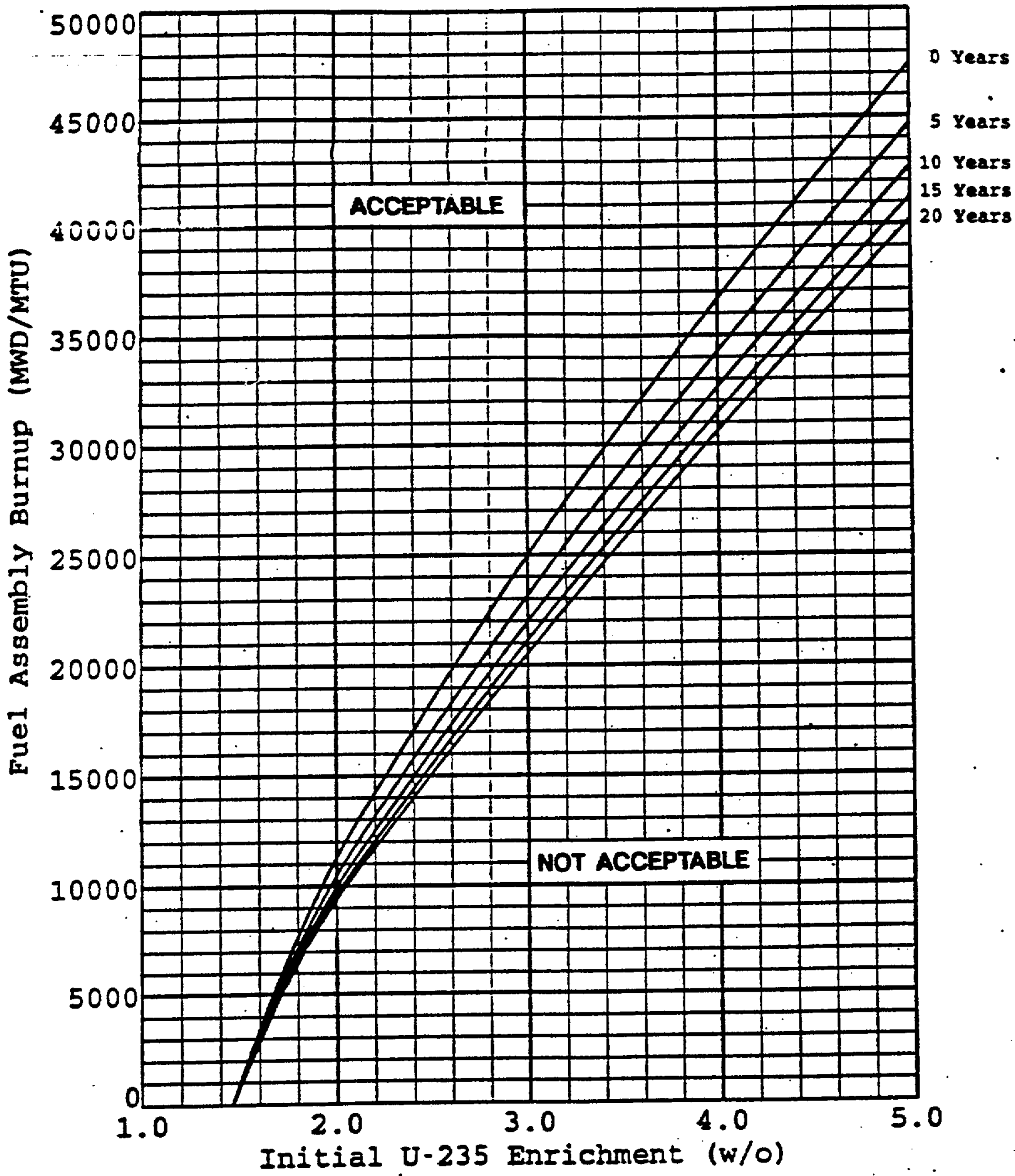
TS.5.6-5 **Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 4 GAD**



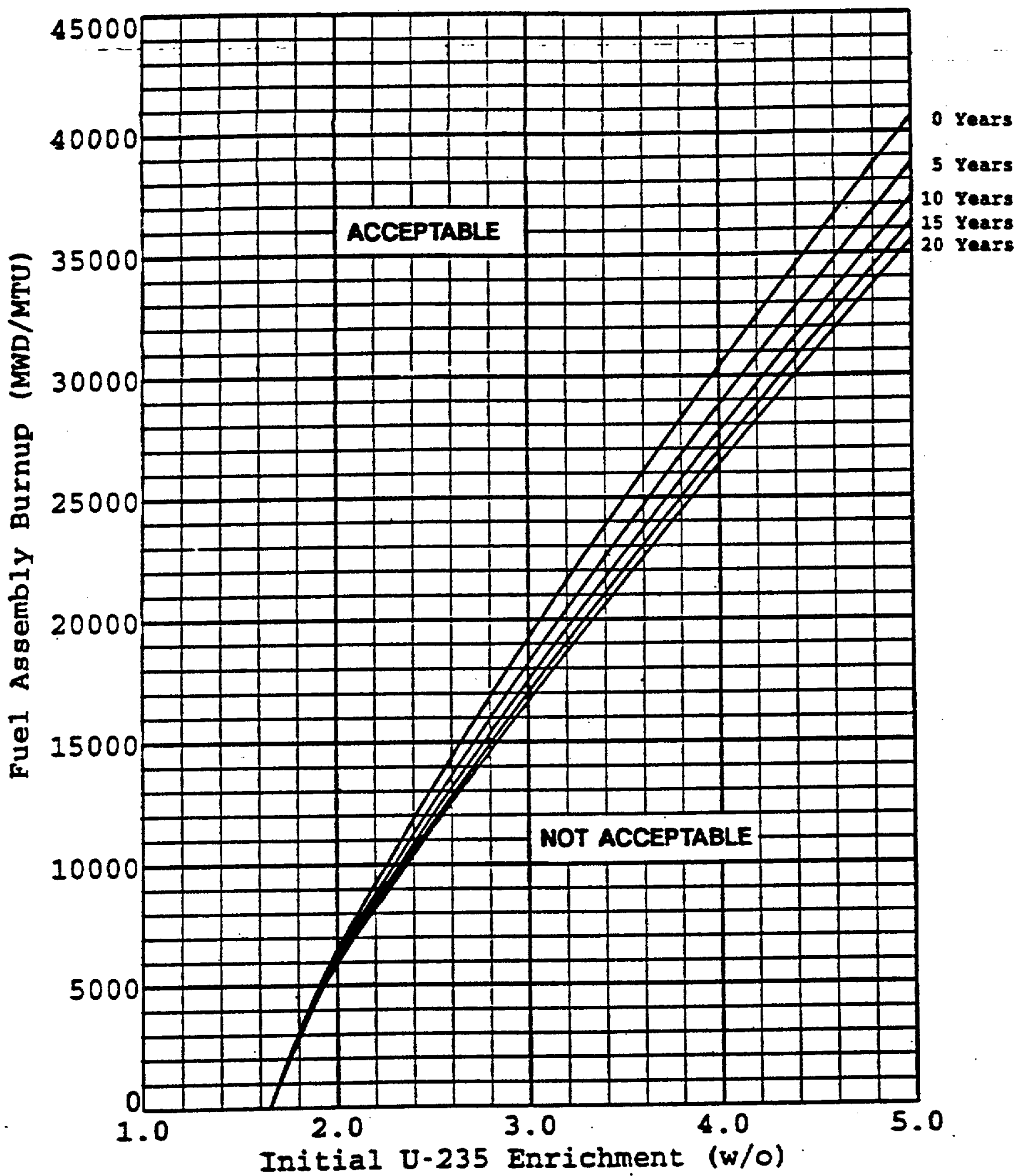
TS.5.6-6 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 4 GAD



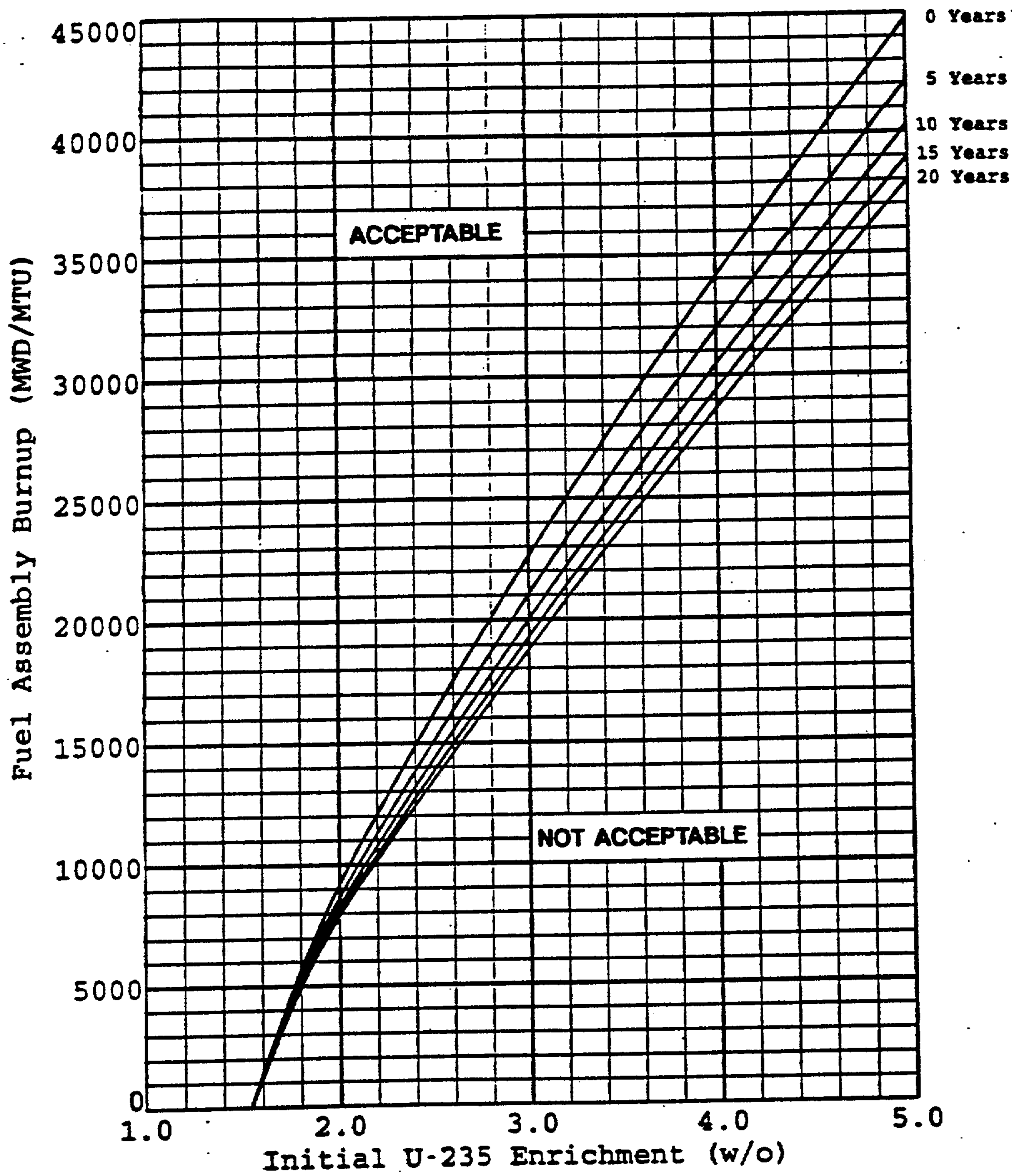
TS.5.6-7 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 8 GAD



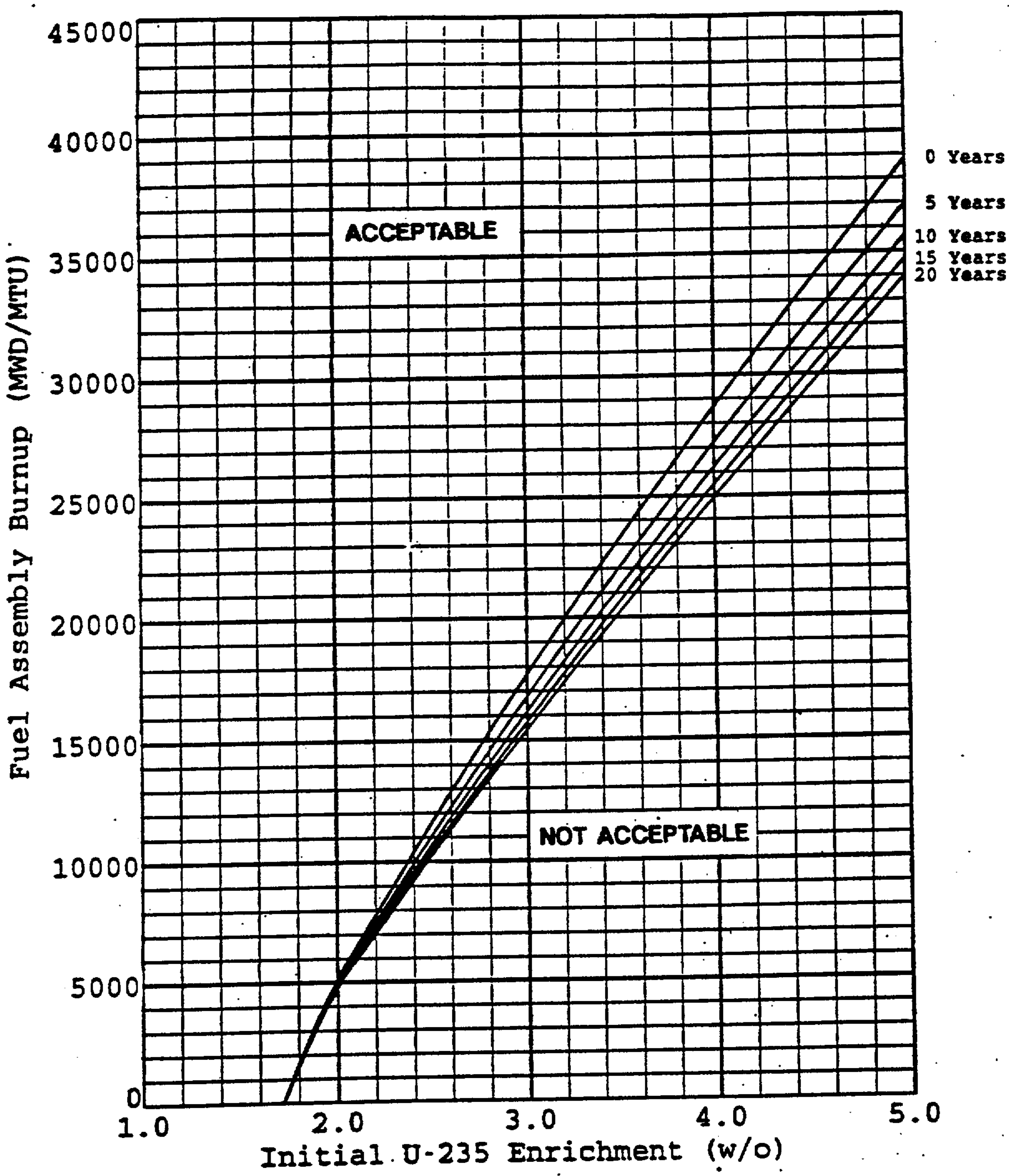
TS.5.6-8 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 8 GAD



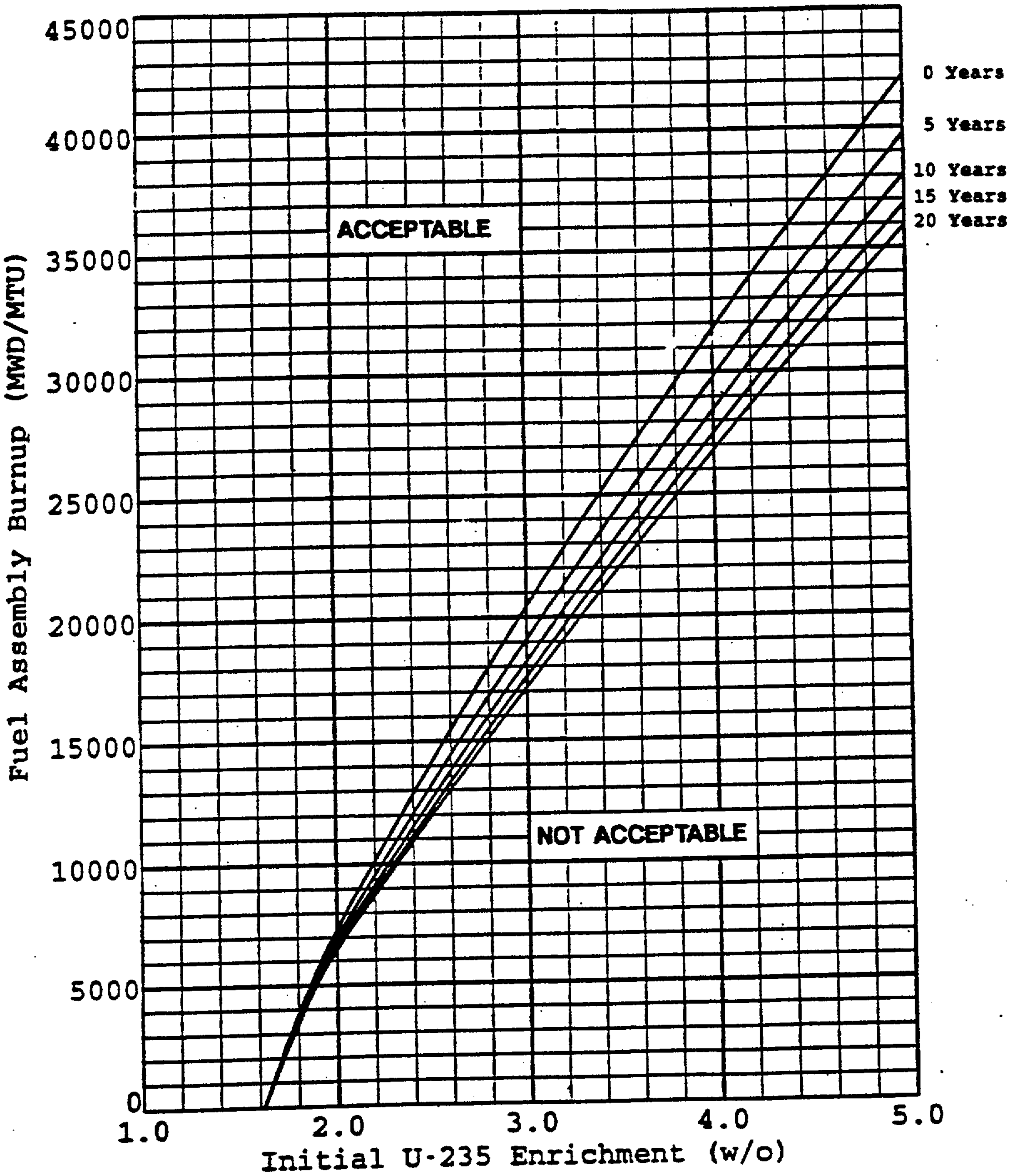
TS.5.6-9 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 12 GAD



TS.5.6-10 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 12 GAD



TS.5.6-11 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - OFA Fuel, 16 or More GAD



TS.5.6-12 Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - STD Fuel, 16 or More GAD

Bases continued

The Spent Fuel Pool Special Ventilation System (Reference 3) is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation System is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

During movement of irradiated fuel assemblies or control rods, a water level of 23 feet is maintained to provide sufficient shielding.

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The water level may also be lowered below 20 feet for upper internals removal/replacement. The basis for these allowance(s) are (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching/unlatching and upper internals removal/replacement the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

The Prairie Island spent fuel storage racks have been analyzed (Reference 8) in accordance with the methodology contained in Reference 5. That methodology ensures that the spent fuel rack multiplication factor, K_{eff}, is less than 0.95 as recommended by ANSI 57.2-1983 (Reference 6) and NRC guidance (Reference 7). The codes, methods and techniques contained in the methodology are used to satisfy this criterion on K_{eff}. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum of 5.0 weight percent U-235 while maintaining $K_{eff} \leq 0.95$ including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool ($K_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing Gadolinium burnable poison.

The Prairie Island specific criticality analysis (Reference 8) utilized the following storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment:

3.8 REFUELING AND FUEL HANDLING

Bases continued

1. The first storage configuration utilizes a checkerboard loading pattern to accommodate new or low burnup fuel with a maximum enrichment of 5.0 wt% U-235. This configuration stores "burned" and "fresh" fuel assemblies in a 3x3 checkerboard pattern as shown in Figure TS.5.6-1. Fuel assemblies stored in "burned" cell locations are selected based on a combination of fuel assembly type, initial enrichment, discharge burnup and decay time (Figures TS.5.6-3 through TS.5.6-12). The criteria for the fuel stored in the "burned" locations is also dependent on the number of rods containing Gadolinium in the center "fresh" fuel assembly. The use of empty cells is also an acceptable option for the "burned" cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.

   Fuel assemblies that fall into the restricted range of Figures TS.3.8-1 or TS.3.8-2 are required to be stored in "fresh" cell locations as shown in Figure TS.5.6-1. The criteria included in Figures TS.3.8-1 and TS.3.8-2 for the selection of fuel assemblies to be stored in the "fresh" cell locations is based on a combination of fuel assembly type, initial enrichment, decay time and discharge burnup.

2. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figures TS.3.8-1 or TS.3.8-2, as applicable, can be stored anywhere in the region with no special placement restrictions.

The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:

1. separated by a vacant row of cells, or

2. the interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure TS.5.6-2).

Specifications 3.8.E.1, 5.6.A.1.d and 5.6.A.1.e ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the Prairie Island spent fuel rack criticality analysis (Reference 8).

The Prairie Island spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, and the Exxon fuel assembly types in storage in the Prairie Island spent fuel pool. The OFA designation on the figures in Sections 3.8 and 5.6.A bound all of the Westinghouse OFA and Vantage Plus fuel assemblies at Prairie Island. The STD designation on the figures in Sections 3.8 and 5.6.A bound all of the Westinghouse STD and Exxon fuel assemblies at Prairie Island.

Bases continued

Most accident conditions in the spent fuel pool will not result in an increase in K_{eff} of the racks in either of the two storage configurations. Examples of those accident conditions which will not result in an increase in K_{eff} are a fuel assembly drop on the top of the racks, a fuel assembly drop between rack modules and wall (rack design precludes this condition), and a drop or placement of a fuel assembly into the cask loading area of the small pool. However, two accidents can be postulated which could increase reactivity. The first postulated accident would be a loss of the fuel pool cooling system and the second would be a misload of a fuel assembly into a cell for which the restrictions on location, enrichment, burnup, decay time or Gadolinium credit are not satisfied.

For an occurrence of these postulated accident conditions, the double contingency principle of ANSI/ANS-8.1-1983 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 750 ppm required to maintain K_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed (Reference 8) to determine the amount of soluble boron required to offset the highest reactivity increase caused by either of these postulated accidents and to maintain K_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 1300 ppm was adequate to mitigate these postulated criticality related accidents and to maintain K_{eff} less than or equal to 0.95. Specification 3.8.E.2 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.8.E.2 was chosen to be consistent with the boron concentration limit required by Specification 3.8.B.1.c for a spent fuel cask containing fuel.

Specification 5.6.A.1.c requires that the spent fuel rack K_{eff} be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 K_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

When the requirements of Specification 3.8.E.1.a are not met, immediate action must be taken to move any non complying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

3.8 REFUELING AND FUEL HANDLING

Bases continued

When the concentration of boron in the spent fuel pool is less than required by Specification 3.8.E.2.a, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. The suspension of fuel movement is not intended to preclude movement of a fuel assembly to a safe position.

References

1. USAR, Section 10.2.1.2
2. USAR, Section 14.5.1
3. USAR, Section 10.3.7
4. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.
5. WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996.
6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.
7. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.
8. "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, February 1997.

4.20 Spent Fuel Pool Storage Configuration

Bases

This surveillance verifies that the fuel assemblies in the spent fuel storage racks are stored in accordance with the requirements of Specifications 3.8.E.1, 5.6.A.1.d and 5.6.A.1.e.

The surveillance is required to be completed within 7 days after the completion of any fuel handling campaign which involves the relocation of fuel assemblies within the spent fuel pool or the addition of fuel assemblies to the spent fuel pool. The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all of the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

It is not the intent of this surveillance to require the completion of a spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.

The 7 day allowance for completion of this surveillance provides adequate time for the completion of a spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the spent fuel pool. If a fuel assembly is misloaded during a fuel handling campaign, the minimum boron concentration required by Specification 3.8.E.2 will ensure that the spent fuel rack K_{eff} remains within limits until the spent fuel pool inventory verification is performed.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 129 AND 121 TO

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated July 28, 1995, as revised February 21, 1997, the Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant (PI), Unit Nos. 1 and 2. The proposed amendments would allow credit for soluble boron in spent fuel criticality analyses. The request is based on the Nuclear Regulatory Commission (NRC) approval of the Westinghouse Owners Group generic methodology for crediting soluble boron given in Topical Report WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Revision 1, November 1996.

2.0 EVALUATION

The PI spent fuel storage racks were analyzed using the Westinghouse methodology (Ref. 1) which has been reviewed and approved by the NRC (Ref. 2). 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% probability, 95% confidence (95/95) level as described in WCAP-14416-NP-A; and

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 PI 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 PI 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 PI storage racks with a high degree of confidence.

The PI spent fuel storage rack design has previously been qualified for storage of various 14 x 14 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. The PI spent fuel racks have been reanalyzed to allow storage of all 14 x 14 fuel assemblies with nominal enrichments up to 4.95 w/o U-235 using credit for checkerboarding, burnup, burnable absorbers, and soluble boron.

For the nominal storage cell design, no credit was taken for the presence of Boraflex panels in the storage racks. In addition, the moderator was assumed to be pure water at a temperature of 68 °F 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. 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 °F to 150 °F) were included. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

In order to determine the enrichment required to maintain k_{eff} less than 1.0 with no credit for soluble boron or Boraflex, two 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 checkerboarding in which assemblies with nominal enrichments up to 4.95 weight percent (w/o) U-235 were stored in the center of a 3x3 checkerboard configuration. The criticality analyses considered all the fuel types currently stored in the spent fuel pool and in use at PI. These include the Optimized Fuel Assembly (OFA) and Vantage Plus designs currently being used, and the Westinghouse Standard (STD) and Exxon (currently Siemens Power Corporation) fuel types used in the past at PI and currently stored in the spent fuel pool. The reactivity of the Westinghouse OFA design was found to be equivalent to the Westinghouse Vantage Plus fuel assemblies and the Westinghouse STD design was found to bound the reactivity of the Exxon fuel assemblies.

The nominal k_{eff} for the all cell storage configuration was determined to be 0.96914 for Westinghouse OFA fuel enriched to 1.87 w/o U-235 and 0.96799 for Westinghouse STD fuel enriched to 1.77 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 1. This resulted in a 95/95 k_{eff} of 0.99947 and 0.99893 for the OFA and STD fuel types, respectively. Since these values are less than 1.0 and were determined at a 95/95 probability/confidence level, they meet the NRC criteria for precluding criticality and are acceptable.

The soluble boron credit calculations assumed the all cell storage configuration moderated by water borated to 200 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 0.93505 and 0.94070 for OFA and STD fuel assembly types, respectively. Since k_{eff} is less than 0.95 with 200 ppm of boron and uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion for precluding criticality is satisfied. Therefore, storage of fuel assemblies with nominal enrichments up to 1.87 w/o U-235 and 1.77 w/o U-235 is acceptable for Westinghouse OFA or STD fuel assembly types, respectively, in all cells of the PI spent fuel storage racks with credit for the presence of 200 ppm boron in the water. This is well below the minimum spent fuel pool boron concentration value of 1800 ppm required by TS 3.8.E.2 and is, therefore, acceptable.

The concept of reactivity equivalencing due to fuel burnup was used to achieve the storage of OFA assemblies with enrichments higher than 1.87 w/o U-235 and STD fuel assemblies with enrichments higher than 1.77 w/o U-235. 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 NRC 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 with enrichments higher than 1.87 w/o U-235 and 1.77 w/o U-235, a series of reactivity calculations were performed to generate a set of enrichment versus fuel assembly discharge burnup (for different decay times) ordered pairs which all yield an equivalent k_{eff} when stored in the PI spent fuel storage racks. These are shown in TS Figures 3.8-1 and 3.8-2 and represent combinations of fuel enrichment and discharge burnup which yield the same rack k_{eff} as the rack loaded with 1.87 w/o and 1.77 w/o fuel (at zero burnup) for OFA and STD assemblies in all cell locations. 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 4% 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 200 ppm required above, that is needed to account for these uncertainties is 200 ppm and 250 ppm for the OFA and STD fuel types, respectively. This results in a total soluble boron credit of 400 ppm and 450 ppm for OFA and STD fuel, respectively. This is well below the minimum spent fuel pool boron concentration value of 1800 ppm required by TS 3.8.E.2 and is, therefore, acceptable.

The nominal k_{eff} for the 3x3 checkerboard storage configuration with a fresh 4.95 w/o U-235 OFA fuel assembly (current PI fuel type) at the center surrounded by OFA fuel enriched to 1.30 w/o U-235 was determined to be 0.96157, and 0.95918, if surrounded by STD fuel enriched to 1.20 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 Ref. 1. This resulted in a 95/95 k_{eff} of 0.99983 and 0.99944 for a fresh OFA assembly surrounded by the OFA and STD fuel types, respectively. Since these values are less than 1.0 and were determined at a 95/95 probability/confidence level, they are acceptable. These results indicate that the PI spent fuel racks will remain subcritical when cells are loaded in a 3x3 checkerboard configuration with a 4.95 w/o U-235 OFA fuel assembly at the center surrounded by any combination of 1.30 w/o U-235 OFA assemblies or 1.20 w/o U-235 STD assemblies.

The soluble boron credit calculations for the 3x3 checkerboard storage configuration assumed the assemblies were moderated by water borated to 250 ppm for surrounding OFA fuel and 300 ppm for surrounding STD fuel. 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 0.94134 and 0.93466 for surrounding OFA and STD fuel assembly types, respectively. Since k_{eff} is less than 0.95 with credit for boron and uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion for precluding criticality is satisfied. Therefore, storage of fresh OFA fuel assemblies with nominal enrichments up to 4.95 w/o U-235 in the center of a 3x3 checkerboard surrounded by 1.30 w/o U-235 OFA or 1.20 w/o U-235 STD fuel assemblies is acceptable in the fuel storage racks with credit for the presence of 250 ppm and 300 ppm boron in the water, respectively.

Storage of surrounding fuel assemblies with enrichments higher than 1.30 w/o U-235 and 1.20 w/o U-235 for the OFA and STD fuel types in the 3x3 checkerboard configuration is achievable by means of reactivity equivalencing, as described above. TS Figures 5.6-3 and 5.6-4 show the constant k_{eff} contours as a function of assembly average burnup, for different decay times, generated for the 3x3 checkerboard storage configuration. The amount of additional boron needed to account for the uncertainties associated with burnup credit was determined to be 350 ppm for surrounding OFA assemblies and 450 ppm for surrounding STD assemblies. This results in a total soluble boron credit of 600 ppm (OFA) and 750 ppm (STD).

The reactivity decrease associated with the presence of gadolinium (GAD) burnable absorbers imbedded in the UO_2 fuel pellet was also analyzed to determine the allowable storage of fuel assemblies with enrichments higher than 1.30 w/o U-235 and 1.20 w/o U-235 for the surrounding OFA and STD fuel types in the 3x3 checkerboard configuration. The credit for the presence of GAD is based on matching the reactivity of these assemblies to an equivalent enrichment of fresh assemblies without GAD. The assemblies with equivalent enrichment are put in a 3x3 checkerboard configuration and the enrichment for the assemblies surrounding the center location (fresh 4.95 w/o U-235 OFA assembly with varying number of GAD rods) is determined so that the new 3x3 checkerboard configuration meets the $0.95k_{eff}$ limit. TS Figures 5.6-5 through 5.6-12 represent combinations of allowable fuel enrichment and discharge burnup of the surrounding assemblies. The uncertainties associated with GAD credit include 3% for manufacturing and 10% for calculational uncertainties. These are acceptable based on current calculational methods. The amount of additional soluble boron needed to account for these uncertainties was determined to be 150 ppm for the OFA fuel assembly type since GAD is only in the center assembly location which is an OFA assembly. This results in a soluble boron credit of 400 ppm and 450 ppm for the surrounding OFA and STD fuel assembly types, respectively, and a maximum total credit of 750 ppm based on STD fuel.

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 150 °F to 240 °F. The second would be a misload of an assembly into a cell for which the restrictions on location, enrichment, burnup, decay time, or GAD credit are not satisfied. Calculations have shown that the misload assembly accident for a 3x3 checkerboard configuration in which a fresh 4.95 w/o U-235 OFA fuel assembly is placed into an incorrect cell results in the highest reactivity increase. The reactivity increase is 0.05891 Δk, which is equivalent to an additional 550 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 3.8.E.2.a (1800 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 (750 ppm maximum) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The NRC Safety Evaluation (SE) for crediting soluble boron (Ref. 2) states that potential events that could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit should be identified. In addition, the available time span of these dilution events should be quantified to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

Deterministic dilution event calculations were performed for PI in order to define the dilution times and volumes necessary to dilute the spent fuel pool from the minimum TS boron concentration of 1800 ppm to a soluble boron concentration where a k_{eff} of 0.95 would be approached (750 ppm). The various initiating events considered included dilution from chemical and volume control system (CVCS) holdup tanks, CVCS monitor tanks, reactor water makeup tanks, CVCS blender, demineralized water system, component cooling water, aerated water, resin flush/fill system, fire protection system, and the reverse osmosis system, and other events that may affect the boron concentration of the pool, such as seismic events, random pipe breaks, loss of offsite power, and the effects of the spent fuel pool demineralizer. Both the small and the large PI spent fuel pools were considered.

An evaluation of these sources has shown that the only credible scenarios involve the transfer of unborated water from the reactor water makeup system to the spent fuel pool cooling or cleanup systems. The reactor water makeup system is capable of supplying the approximately 345,000 gallons of water necessary to dilute the pool from 1800 ppm to 750 ppm at a rate of approximately 80 gpm if the inservice reactor water makeup tank is repeatedly replenished from the water treatment system. However, the worst scenario would require continued manual actions on the part of plant personnel for at least 72 hours for the dilution to occur. Therefore, a dilution event large enough to result in a significant reduction in the spent fuel pool boron concentration would involve such a large water volume turnover and would occur over such a long time period that it should be readily detected via level alarms and/or visual inspections and terminated by plant personnel before a dilution sufficient to approach the 0.95 k_{eff} limit could occur. The weekly spent fuel pool boron concentration sampling requirement specified in TS Table 4.1-2B will provide assurance that smaller and less readily identifiable boron concentration reductions are not taking place.

Additionally, the criticality analysis for the PI spent fuel racks also showed that k_{eff} would remain less than 1.0 at a 95/95 probability/confidence level even if the pool were completely filled with unborated water. Thus, even if the pool were diluted to zero ppm, which would take significantly more water than evaluated above, the racks would be expected to remain subcritical.

The TS changes proposed as a result of the revised criticality analysis are consistent with the changes stated in the NRC SE for WCAP-14416 (Ref. 2). Westinghouse submitted a revised Topical Report, WCAP-14416-NP-A, which incorporated the changes stated in the NRC SE (Ref 2). Also, since the staff disagreed with the proprietary finding of the original WCAP-14416, 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.

2.1 Summary

Based on the review described above, the staff finds the criticality aspects of the proposed PI 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 following storage configurations and U-235 enrichment limits were determined to be acceptable:

1) Westinghouse 14x14 OFA fuel assemblies with nominal enrichments no greater than 1.87 w/o U-235 and Westinghouse 14x14 STD and Exxon 14x14 fuel assemblies with nominal enrichments no greater than 1.77 w/o U-235 can be stored in any cell location. Fuel assemblies with initial nominal enrichments greater than these must satisfy a minimum burnup and decay time requirement as shown in TS Figures 3.8-1 and 3.8-2.

2) Westinghouse 14x14 OFA assemblies with nominal enrichments no greater than 4.95 w/o U-235 can be stored in the center of a 3x3 checkerboard configuration. The surrounding fuel assemblies must have an initial nominal enrichment no greater than 1.30 w/o U-235 for Westinghouse 14x14 OFA fuel assemblies and 1.20 w/o U-235 for Westinghouse 14x14 STD and Exxon 14x14 fuel assemblies. Fuel assemblies with initial nominal enrichments greater than these must satisfy a minimum burnup and decay time requirement as shown in TS Figures 5.6-3 and 5.6-4. The enrichment limits of the surrounding fuel assemblies may be increased by crediting GAD burnable absorber in the center assembly as shown in TS Figures 5.6-4 through 5.6-12.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* on June 11, 1997 (62 FR 31852).

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

5.0 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.

6.0 REFERENCES

1. Newmyer, W. D., Westinghouse Electric Corp., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, Rev. 1, November 1996.

2. Collins, T. E., NRC letter to T. Greene, Westinghouse Owners Group, Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," (TAC No. M93254), October 25, 1996.

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