September 3, 1993

Docket Nos. 50-282
and 50-306

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

Dear Mr. Anderson::

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: INCREASED FUEL ENRICHMENT LIMIT CHANGES (TAC NOS. M86727 AND M86728)

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-42 and Amendment No. 101 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 11, 1993, as revised June 30, 1993.

The amendments revise the TS to increase fuel enrichment from 4.25 weight percent to 5.0 weight percent. This includes a revision to the TS to allow 5.0 weight percent U-235 fuel to be stored in the new fuel vault and the spent fuel pool and used in the core. In addition, the amendments revise the TS to increase the minimum RWST boron concentration and incorporate references to natural uranium and ZIRLO clad material into the reactor core design description.

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

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

Enclosures:
1. Amendment No. 108 to DPR-42
2. Amendment No. 101 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PM:PD31	SRXB	OGC	(A)D:PD31
NAME	CJamerson	MGamberoni	RJones	MYoung	WDean
DATE	8/31/93	8/31/93	8/31/93	9/1/93	9/3/93

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P PDR

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

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cc w/enclosures:
See next page

Mr. Roger O. Anderson, Director
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Prairie Island Nuclear Generating Plant

cc:

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DATED: September 3, 1993

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

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130013

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. 108
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 June 11, 1993, as revised June 30, 1993, 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:

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Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 108 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 3, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 108

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-iii	TS-iii
TS-vii	TS-vii
TS-xiii	TS-xiii
TS.3.3-1	TS.3.3-1
TS.3.8-4	TS.3.8-4
----	TS.3.8-5
----	Figure TS.3.8-1
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.5.3-1	TS.5.3-1
TS.5.6-1	TS.5.6-1
TS.5.6-2	TS.5.6-2
----	TS.5.6-3
----	Figure TS.5.6-1
----	Figure TS.5.6-2
B.3.8-2	B.3.8-2
----	B.3.8-3
----	B.3.8-4

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	B. Vacuum Breaker System	TS.3.6-1
	C. Containment Isolation Valves	TS.3.6-1
	D. Containment Purge System	TS.3.6-2
	E. Auxiliary Building Special Ventilation Zone Integrity	TS.3.6-2
	F. Auxiliary Building Special Ventilation System	TS.3.6-3
	G. Shield Building Integrity	TS.3.6-3
	H. Shield Building Ventilation System	TS.3.6-3
	I. Containment Internal Pressure	TS.3.6-3
	J. Containment and Shield Building Air Temperature	TS.3.6-4
	K. Containment Shell Temperature	TS.3.6-4
	L. Electric Hydrogen Recombiners	TS.3.6-4
	M. Containment Air Locks	TS.3.6-4
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	A. Core Alterations	TS.3.8-1
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	D. Spent Fuel Pool Special Ventilation System	TS.3.8-4
	E. Spent Fuel Pool Storage	TS.3.8-4
3.9	Radioactive Effluents	TS.3.9-1
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	1. Dose Rate	TS.3.9-3
	2. Dose from Noble Gases	TS.3.9-3
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	4. Gaseous Radwaste Treatment System and Ventilation Exhaust Treatment Systems	TS.3.9-4
	5. Containment Purging	TS.3.9-5
	C. Solid Radioactive Waste	TS.3.9-6
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	E. Radioactive Liquid Effluent Monitoring Instrumentation	TS.3.9-7
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	2. Shield Building	TS.5.2-2
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	C. Containment System Functional Design	TS.5.2-3
5.3	Reactor	TS.5.3-1
	A. Reactor Core	TS.5.3-1
	B. Reactor Coolant System	TS.5.3-1
	C. Protection Systems	TS.5.3-1
5.4	Engineered Safety Features	TS.5.4-1
5.5	Radioactive Waste Systems	TS.5.5-1
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5.6	Fuel Handling	TS.5.6-1
	A. Criticality Consideration	TS.5.6-1
	B. Spent Fuel Storage Structure	TS.5.6-1
	C. Fuel Handling	TS.5.6-2
	D. Spent Fuel Storage Capacity	TS.5.6-3

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

TS FIGURE	TITLE
2.1-1	Safety Limits, Reactor Core, Thermal and Hydraulic Two Loop Operation
3.1-1	Unit 1 and Unit 2 Reactor Coolant System Heatup Limitations
3.1-2	Unit 1 and Unit 2 Reactor Coolant System Cooldown Limitations
3.1-3	DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity >1.0 uCi/gram DOSE EQUIVALENT I-131
3.8-1	Spent Fuel Pool Unrestricted Region Minimum Burnup Requirements
3.9-1	Prairie Island Nuclear Generating Plant Site Boundary for Liquid Effluents
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5.6-1	Spent Fuel Pool Burned/Fresh Checkerboard Cell Layout
5.6-2	Spent Fuel Pool Checkerboard Region Minimum Burnup Requirements
6.1-1	NSP Corporate Organizational Relationship to On-Site Operating Organizations
6.1-2	Prairie Island Nuclear Generating Plant Functional Organization for On-Site Operating Group

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.3.A.2 below):

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 2500 ppm.

   b. Each reactor coolant system accumulator shall be OPERABLE when reactor coolant system pressure is greater than 1000 psig.

      OPERABILITY requires:
      (1) The isolation valve is open
      (2) Volume is 1270 ± 20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of 740 ± 30 psig

   c. Two safety injection pumps are OPERABLE except that pump control switches in the control room shall meet the requirements of Section 3.3.A.3, 3.3.A.4 and 3.1.A.1.d.(2) whenever the reactor coolant system temperature is less than 310°F*.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers are OPERABLE.

*Valid until 20 EFPY

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. To be stored without restriction in the spent fuel pool, the burnup and initial enrichment of a fuel assembly shall be within the unrestricted range of Figure TS.3.8-1.

b. Fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 shall be stored in accordance with Specification 5.6.A.1.d.

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

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

3.8.E.2. Spent Fuel Pool Boron Concentration

a. The spent fuel pool boron concentration shall be ≥ 1,800 ppm when fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool.

b. If the requirements of specification 3.8.E.2.a are applicable and the spent fuel pool boron concentration is not within its limit, then immediately:

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

   2. Either initiate action to restore spent fuel pool boron concentration to within its limit or perform a spent fuel pool verification.

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

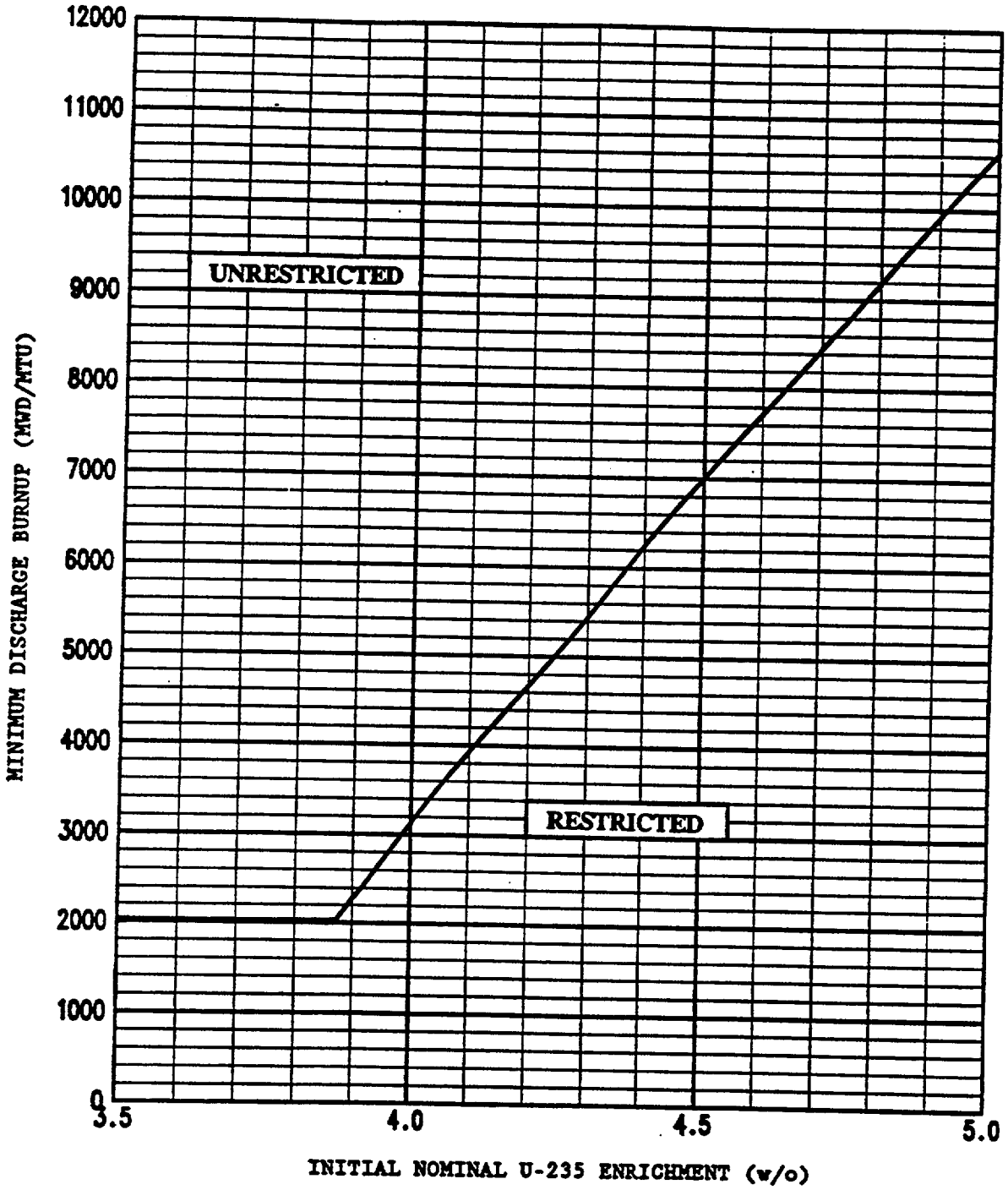


FIGURE TS.3.8-1 Spent Fuel Pool Unrestricted Region Minimum Burnup Requirements

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
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)	9.2
9. RWST Boron Concentration	Weekly	
10. Boric Acid Tanks Boron Concentration	2/Week	
11. Caustic Standpipe NaOH Concentration	Monthly	6.4
12. Accumulator Boron Concentration	Monthly	6
13. Spent Fuel Pit Boron Concentration	Monthly/Weekly(7)(8)	9.5.5

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
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.

7. The minimum spent fuel pool boron concentration from Specification 3.8.B.1.b shall be verified by chemical analysis weekly while a spent fuel cask containing fuel is located in the spent fuel pool.

8. The spent fuel pool boron concentration shall be verified weekly, by chemical analysis, to be within the limits of Specification 3.8.E.2.a when fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool.

5.3 REACTOR

A. Reactor Core

1. The reactor core contains uranium in the form of natural or slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods (Reference 1).

2. The maximum enrichment will be 5.0 weight percent U-235.

3. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel (Reference 2).

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements (Reference 3).

2. All high pressure piping, components of the reactor coolant system and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

   b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, is 6100 cubic feet.

C. Protection Systems

The protection systems for the reactor and engineered safety features are designed to applicable codes, including IEEE-279, dated 1968. The design includes a reactor trip for a high negative rate of change of neutron flux as measured by the excore nuclear instruments (Reference 4). The system is intended to trip the reactor upon the abnormal dropping of more than one control rod (Reference 4). If only one control rod is dropped, the core can be operated at full power for a short time, as permitted by Specification 3.10.

References

1. USAR, Section 3.4.2
2. USAR, Section 3.5.2
3. USAR, Table 4.1-11
4. USAR, Section 7.1

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

   c. New or spent fuel assemblies with a combination of burnup and initial enrichment in the unrestricted range of Figure TS.3.8-1 allowed unrestricted storage in the spent fuel racks; and

   d. New or spent fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 stored in compliance with Figures TS.5.6-1 and TS.5.6-2.

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.

The spent fuel pool has a reinforced concrete bottom slab nearly 6 feet thick and has been designed to minimize loss of water due to a dropped cask accident. Piping to the pool is arranged so that failure of any pipe cannot drain the pool below the tops of the stored fuel assemblies.

The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I (seismic) structures. The spent fuel pit has a stainless steel liner to ensure against loss of water (Reference 1).

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations.

C. Fuel Handling

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after post-irradiation cooling. The system consists of the refueling cavity, the fuel transfer system, the spent fuel storage pit, and the spent fuel cask transfer system.

Major components of the fuel handling system are the manipulation crane, the spent fuel pool bridge, the auxiliary building crane, the fuel transfer system, the spent fuel storage racks, the spent fuel cask, and the rod cluster control changing fixture. The reactor vessel stud tensioner, the reactor vessel head lifting device, and the reactor internals lifting device are used for preparing the reactor for refueling and for assembling the reactor after refueling.

Upon arrival in the storage pit, spent fuel will be removed from the transfer system and placed, one assembly at a time, in storage racks using a long-handled manual tool suspended from the spent fuel pit bridge crane. After sufficient decay, the fuel will be loaded into storage casks for storage in the Independent Spent Fuel Storage Installation or into shipping casks for removal from the site. The casks will be handled by the auxiliary building crane.

Spent fuel casks will be handled by a single failure proof handling system meeting the requirements of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980. The auxiliary building crane has been upgraded to conform with the single failure proof requirements of Section 5.1.6 of NUREG-0612. The auxiliary building crane is designed to not allow a load drop as a result of any single failure. The improved reliability of the auxiliary building crane is achieved through increased factors of safety and through redundancy or duality in certain active components.

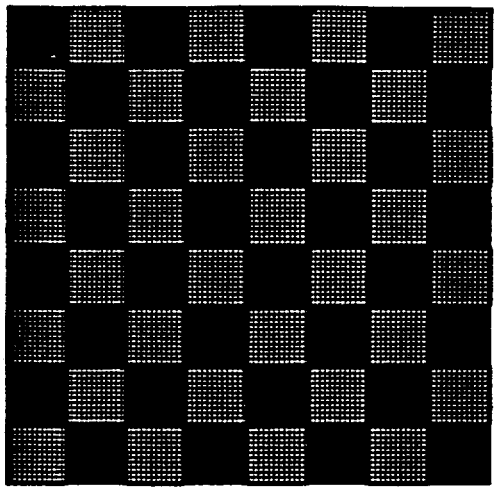
D. Spent Fuel Storage Capacity

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.

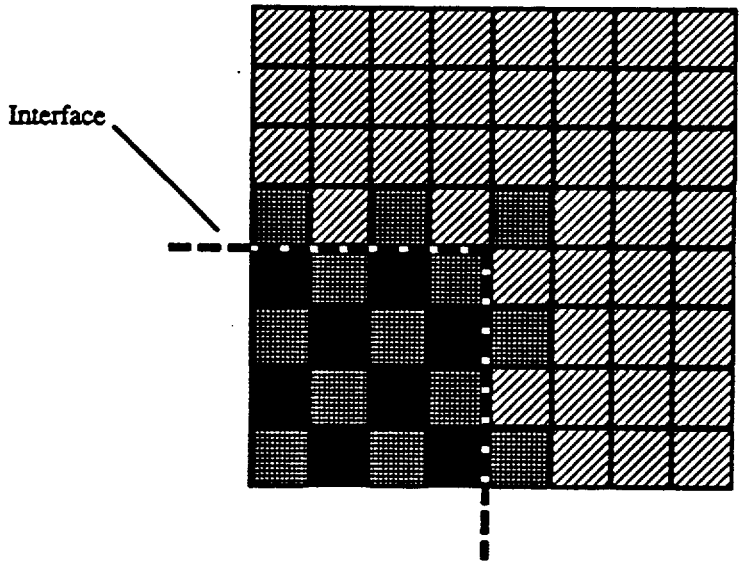
References

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.



PATTERN FOR CHECKERBOARD REGION



BOUNDARY BETWEEN CHECKERBOARD AND UNRESTRICTED REGIONS

	Fresh Fuel:	Enrichments up to 5.0 w/o U-235, no restrictions on burnup
	Checkerboard Region Burned Fuel:	Must satisfy minimum burnup requirements of Figure TS.5.6-2.
	Unrestrcited Region Burned Fuel:	Must satisfy minimum burnup requirements of Figure TS.3.8-1.

Note: The Checkerboard and unrestricted regions can alternatively be separated by a single row of vacant cells on each adjacent face.

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

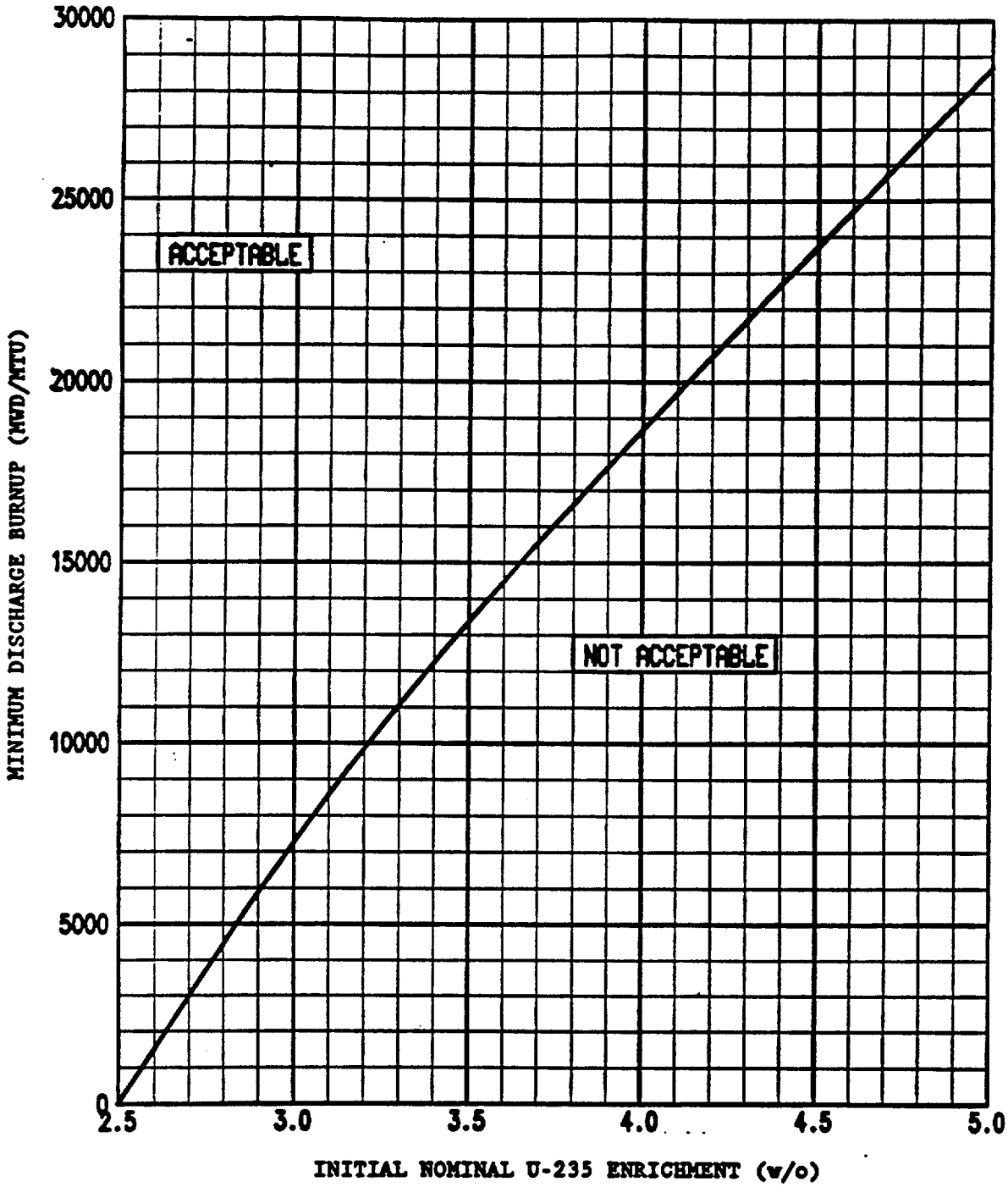


FIGURE TS.5.6-2 Spent Fuel Pool Checkerboard Region Minimum Burnup Requirements

3.8 REFUELING AND FUEL HANDLING

Bases continued

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 4) to allow for the storage of fuel assemblies with enrichments up to 5.0 weight percent U-235 while maintaining $K_{eff} \leq 0.95$ including uncertainties. This criticality analysis utilized the following storage configurations or regions 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:

1. The first region 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 2x2 checkerboard pattern. Fuel assemblies stored in "burned" cell locations must have an initial enrichment less than 2.5 wt% U-235 (nominal) or satisfy a minimum burnup requirement. The use of empty cells is also an acceptable option for the "burned" cell locations. Fuel assemblies stored in the "fresh" cell locations can have enrichments up to 5.0 wt% U-235 with no requirements for burnup or burnable absorbers.

2. The second region does not utilize any special loading pattern. Fuel assemblies with burnup and initial enrichments which fall into the unrestricted range of Figure TS.3.8-1 can be stored anywhere in the region with no special placement restrictions. Fuel assemblies which fall into the restricted range of Figure TS.3.8-1 must be stored in the checkerboard region in accordance with Specification 5.6.A.1.d.

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.5.6-1).

3.8 REFUELING AND FUEL HANDLING

Bases continued

Figure TS.3.8-1, which specifies the minimum burnup requirements for unrestricted storage in the spent fuel pool, is based on enrichments from 3.87 to 5.0 weight percent U-235. Enrichments lower than 3.87 weight percent are conservatively bounded by the minimum burnup requirement for 3.87 weight percent U-235 which is 2000 MWD/MTU. Therefore, Figure TS.3.8-1 has been drawn to require that fuel with an initial enrichment of less than 3.87 weight percent U-235 have 2000 MWD/MTU burnup or greater before unrestricted storage in the spent fuel pool will be allowed.

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which ensures that each region is maintained in a subcritical condition during normal operation with the regions fully loaded.

Most accident conditions do not result in a significant increase in the activity of either of the two regions. Examples of these accident conditions are the loss of cooling, the dropping of a fuel assembly on the top of the rack, and the dropping of a fuel assembly between rack modules and wall (rack design precludes this condition). However, accidents can be postulated that could increase the reactivity. For these accident conditions, the double contingency principle of ANSI N16.1-1975 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident.

The double contingency principle allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is the accidental misloading of a fuel assembly into a rack location for which the restrictions on location, enrichment or burnup are not satisfied. This could potentially increase the reactivity in spent fuel racks. To mitigate these postulated criticality related accidents, Specification 3.8.E.2 ensures the spent fuel pool contains adequate dissolved boron anytime fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool. The negative reactivity effect of the soluble boron would compensate for the increased reactivity caused by a mispositioned fuel assembly.

The boron concentration requirements of Specification 3.8.E.2 are no longer imposed when no fuel movements are occurring and a spent fuel pool verification has been completed, because the storage requirements of Specifications 3.8.E.1 and 5.6.A.1.d are then adequate to prevent criticality.

Specification 3.8.E.2.a is not imposed when only fuel assemblies with a combination of burnup and initial enrichment in the unrestricted range of Figure TS.3.8-1 are stored in the spent fuel pool. The requirements of Specification 3.8.E.2.a are not required in that case because with only fuel assemblies that have burnup and initial enrichment in the unrestricted range of Figure TS.3.8-1 it is not possible to cause an inadvertent criticality by mispositioning a fuel assembly in the spent fuel pool.

3.8 REFUELING AND FUEL HANDLING

Bases continued

When the requirements of Specification 3.8.E.2.a are applicable, and the concentration of boron in the spent fuel pool is less than required, 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. An acceptable alternative is to complete a spent fuel pool verification. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

A spent fuel verification is required following the last movement of fuel assemblies in the spent fuel pool, if fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool. This verification will confirm that any fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in accordance with the requirements of Specification 5.6.A.1.d.

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.

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. 101
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 June 11, 1993, as revised June 30, 1993, 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. 101, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 3, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 101

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-iii	TS-iii
TS-vii	TS-vii
TS-xiii	TS-xiii
TS.3.3-1	TS.3.3-1
TS.3.8-4	TS.3.8-4
----	TS.3.8-5
----	Figure TS.3.8-1
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.5.3-1	TS.5.3-1
TS.5.6-1	TS.5.6-1
TS.5.6-2	TS.5.6-2
----	TS.5.6-3
----	Figure TS.5.6-1
----	Figure TS.5.6-2
B.3.8-2	B.3.8-2
----	B.3.8-3
----	B.3.8-4

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	C. Containment Isolation Valves	TS.3.6-1
	D. Containment Purge System	TS.3.6-2
	E. Auxiliary Building Special Ventilation Zone Integrity	TS.3.6-2
	F. Auxiliary Building Special Ventilation System	TS.3.6-3
	G. Shield Building Integrity	TS.3.6-3
	H. Shield Building Ventilation System	TS.3.6-3
	I. Containment Internal Pressure	TS.3.6-3
	J. Containment and Shield Building Air Temperature	TS.3.6-4
	K. Containment Shell Temperature	TS.3.6-4
	L. Electric Hydrogen Recombiners	TS.3.6-4
	M. Containment Air Locks	TS.3.6-4
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	D. Spent Fuel Pool Special Ventilation System	TS.3.8-4
	E. Spent Fuel Pool Storage	TS.3.8-4
3.9	Radioactive Effluents	TS.3.9-1
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APPENDIX A TECHNICAL SPECIFICATIONS

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3.1-3	DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity >1.0 uCi/gram DOSE EQUIVALENT I-131
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5.6-2	Spent Fuel Pool Checkerboard Region Minimum Burnup Requirements
6.1-1	NSP Corporate Organizational Relationship to On-Site Operating Organizations
6.1-2	Prairie Island Nuclear Generating Plant Functional Organization for On-Site Operating Group

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.3.A.2 below):

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 2500 ppm.

   b. Each reactor coolant system accumulator shall be OPERABLE when reactor coolant system pressure is greater than 1000 psig.

      OPERABILITY requires:
      (1) The isolation valve is open
      (2) Volume is 1270 ± 20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of 740 ± 30 psig

   c. Two safety injection pumps are OPERABLE except that pump control switches in the control room shall meet the requirements of Section 3.3.A.3, 3.3.A.4 and 3.1.A.1.d.(2) whenever the reactor coolant system temperature is less than 310°F*.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers are OPERABLE.

*Valid until 20 EFPY

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. To be stored without restriction in the spent fuel pool, the burnup and initial enrichment of a fuel assembly shall be within the unrestricted range of Figure TS.3.8-1.

b. Fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 shall be stored in accordance with Specification 5.6.A.1.d.

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

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

3.8.E.2. Spent Fuel Pool Boron Concentration

a. The spent fuel pool boron concentration shall be ≥ 1,800 ppm when fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool.

b. If the requirements of specification 3.8.E.2.a are applicable and the spent fuel pool boron concentration is not within its limit, then immediately:

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

   2. Either initiate action to restore spent fuel pool boron concentration to within its limit or perform a spent fuel pool verification.

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

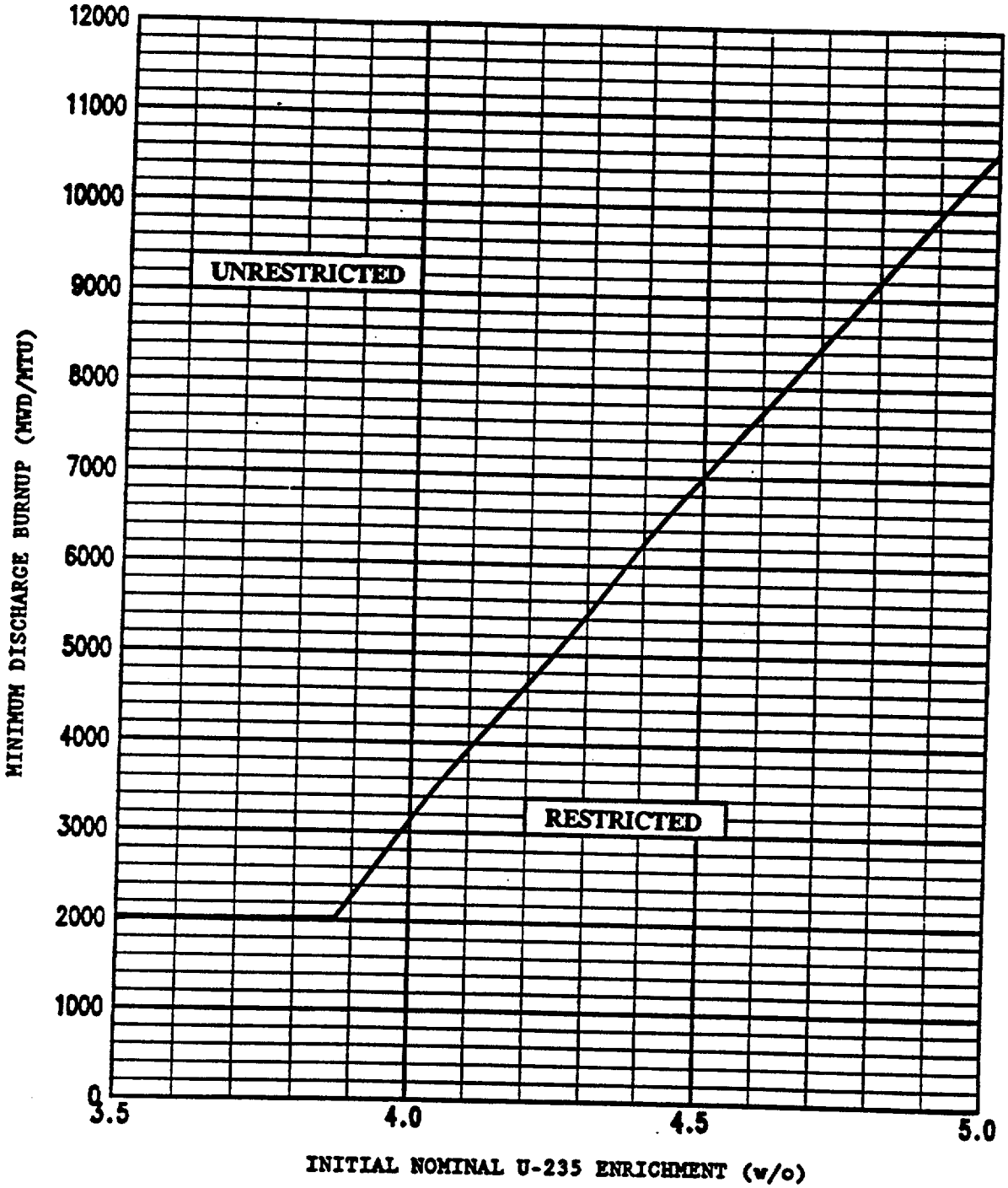


FIGURE TS.3.8-1 Spent Fuel Pool Unrestricted Region Minimum Burnup Requirements

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
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)	9.2
9. RWST Boron Concentration	Weekly	
10. Boric Acid Tanks Boron Concentration	2/Week	
11. Caustic Standpipe NaOH Concentration	Monthly	6.4
12. Accumulator Boron Concentration	Monthly	6
13. Spent Fuel Pit Boron Concentration	Monthly/Weekly(7)(8)	9.5.5

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
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.

7. The minimum spent fuel pool boron concentration from Specification 3.8.B.1.b shall be verified by chemical analysis weekly while a spent fuel cask containing fuel is located in the spent fuel pool.

8. The spent fuel pool boron concentration shall be verified weekly, by chemical analysis, to be within the limits of Specification 3.8.E.2.a when fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool.

5.3 REACTOR

A. Reactor Core

1. The reactor core contains uranium in the form of natural or slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods (Reference 1).

2. The maximum enrichment will be 5.0 weight percent U-235.

3. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel (Reference 2).

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements (Reference 3).

2. All high pressure piping, components of the reactor coolant system and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

   b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, is 6100 cubic feet.

C. Protection Systems

The protection systems for the reactor and engineered safety features are designed to applicable codes, including IEEE-279, dated 1968. The design includes a reactor trip for a high negative rate of change of neutron flux as measured by the excore nuclear instruments (Reference 4). The system is intended to trip the reactor upon the abnormal dropping of more than one control rod (Reference 4). If only one control rod is dropped, the core can be operated at full power for a short time, as permitted by Specification 3.10.

References

1. USAR, Section 3.4.2
2. USAR, Section 3.5.2
3. USAR, Table 4.1-11
4. USAR, Section 7.1

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

   c. New or spent fuel assemblies with a combination of burnup and initial enrichment in the unrestricted range of Figure TS.3.8-1 allowed unrestricted storage in the spent fuel racks; and

   d. New or spent fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 stored in compliance with Figures TS.5.6-1 and TS.5.6-2.

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.

The spent fuel pool has a reinforced concrete bottom slab nearly 6 feet thick and has been designed to minimize loss of water due to a dropped cask accident. Piping to the pool is arranged so that failure of any pipe cannot drain the pool below the tops of the stored fuel assemblies.

The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I (seismic) structures. The spent fuel pit has a stainless steel liner to ensure against loss of water (Reference 1).

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations.

C. Fuel Handling

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after post-irradiation cooling. The system consists of the refueling cavity, the fuel transfer system, the spent fuel storage pit, and the spent fuel cask transfer system.

Major components of the fuel handling system are the manipulation crane, the spent fuel pool bridge, the auxiliary building crane, the fuel transfer system, the spent fuel storage racks, the spent fuel cask, and the rod cluster control changing fixture. The reactor vessel stud tensioner, the reactor vessel head lifting device, and the reactor internals lifting device are used for preparing the reactor for refueling and for assembling the reactor after refueling.

Upon arrival in the storage pit, spent fuel will be removed from the transfer system and placed, one assembly at a time, in storage racks using a long-handled manual tool suspended from the spent fuel pit bridge crane. After sufficient decay, the fuel will be loaded into storage casks for storage in the Independent Spent Fuel Storage Installation or into shipping casks for removal from the site. The casks will be handled by the auxiliary building crane.

Spent fuel casks will be handled by a single failure proof handling system meeting the requirements of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980. The auxiliary building crane has been upgraded to conform with the single failure proof requirements of Section 5.1.6 of NUREG-0612. The auxiliary building crane is designed to not allow a load drop as a result of any single failure. The improved reliability of the auxiliary building crane is achieved through increased factors of safety and through redundancy or duality in certain active components.

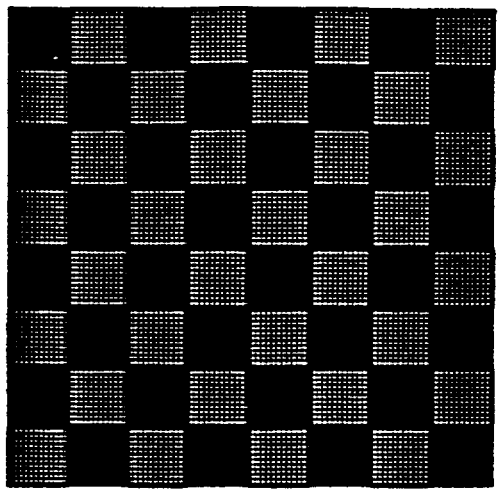
D. Spent Fuel Storage Capacity

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.

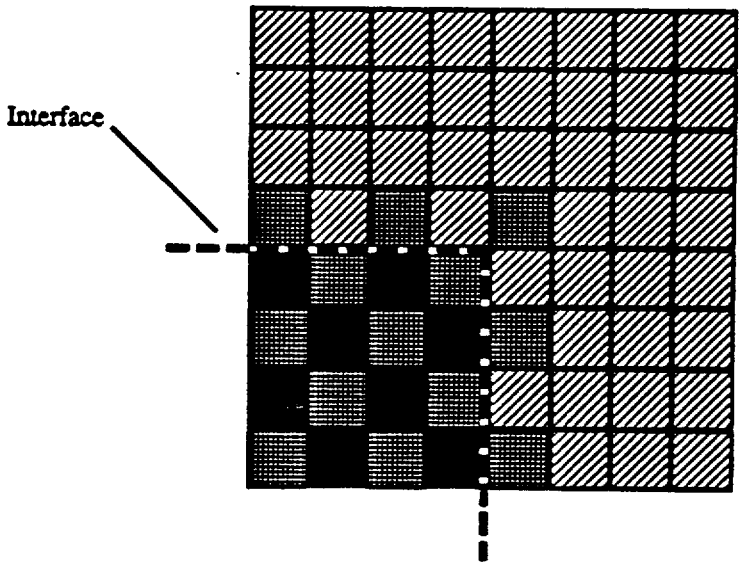
References

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.



PATTERN FOR CHECKERBOARD REGION



BOUNDARY BETWEEN CHECKERBOARD AND UNRESTRICTED REGIONS

Fresh Fuel:	Enrichments up to 5.0 w/o U-235, no restrictions on burnup
Checkerboard Region Burned Fuel:	Must satisfy minimum burnup requirements of Figure TS.5.6-2.
Unrestrcited Region Burned Fuel:	Must satisfy minimum burnup requirements of Figure TS.3.8-1.

Note: The Checkerboard and unrestricted regions can alternatively be separated by a single row of vacant cells on each adjacent face.

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

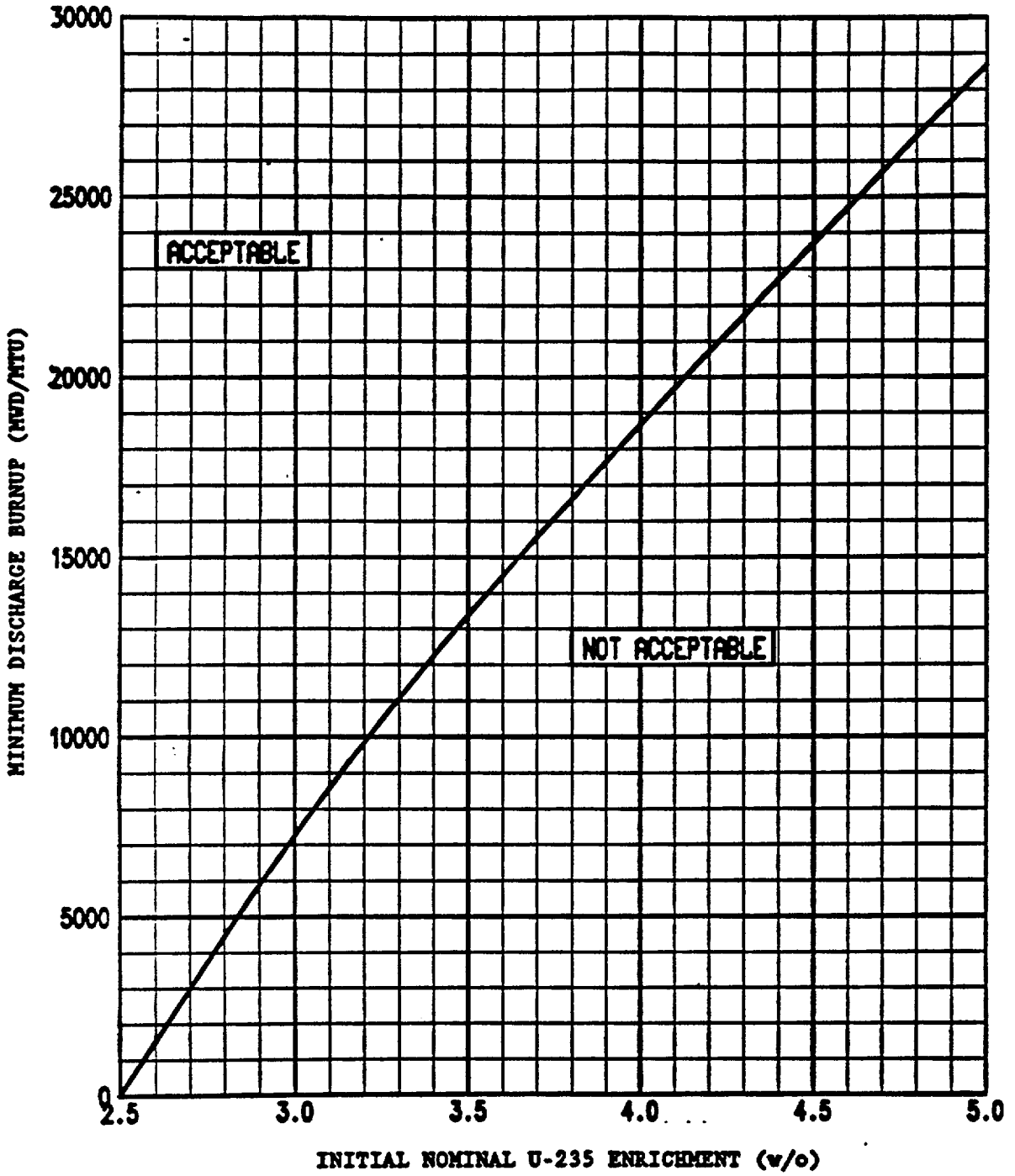


FIGURE TS.5.6-2 Spent Fuel Pool Checkerboard Region Minimum Burnup Requirements

3.8 REFUELING AND FUEL HANDLING

Bases continued

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 4) to allow for the storage of fuel assemblies with enrichments up to 5.0 weight percent U-235 while maintaining $K_{eff} \leq 0.95$ including uncertainties. This criticality analysis utilized the following storage configurations or regions 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:

1. The first region 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 2x2 checkerboard pattern. Fuel assemblies stored in "burned" cell locations must have an initial enrichment less than 2.5 wt% U-235 (nominal) or satisfy a minimum burnup requirement. The use of empty cells is also an acceptable option for the "burned" cell locations. Fuel assemblies stored in the "fresh" cell locations can have enrichments up to 5.0 wt% U-235 with no requirements for burnup or burnable absorbers.

2. The second region does not utilize any special loading pattern. Fuel assemblies with burnup and initial enrichments which fall into the unrestricted range of Figure TS.3.8-1 can be stored anywhere in the region with no special placement restrictions. Fuel assemblies which fall into the restricted range of Figure TS.3.8-1 must be stored in the checkerboard region in accordance with Specification 5.6.A.1.d.

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.5.6-1).

3.8 REFUELING AND FUEL HANDLING

Bases continued

Figure TS.3.8-1, which specifies the minimum burnup requirements for unrestricted storage in the spent fuel pool, is based on enrichments from 3.87 to 5.0 weight percent U-235. Enrichments lower than 3.87 weight percent are conservatively bounded by the minimum burnup requirement for 3.87 weight percent U-235 which is 2000 MWD/MTU. Therefore, Figure TS.3.8-1 has been drawn to require that fuel with an initial enrichment of less than 3.87 weight percent U-235 have 2000 MWD/MTU burnup or greater before unrestricted storage in the spent fuel pool will be allowed.

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which ensures that each region is maintained in a subcritical condition during normal operation with the regions fully loaded.

Most accident conditions do not result in a significant increase in the activity of either of the two regions. Examples of these accident conditions are the loss of cooling, the dropping of a fuel assembly on the top of the rack, and the dropping of a fuel assembly between rack modules and wall (rack design precludes this condition). However, accidents can be postulated that could increase the reactivity. For these accident conditions, the double contingency principle of ANSI N16.1-1975 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident.

The double contingency principle allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is the accidental misloading of a fuel assembly into a rack location for which the restrictions on location, enrichment or burnup are not satisfied. This could potentially increase the reactivity in spent fuel racks. To mitigate these postulated criticality related accidents, Specification 3.8.E.2 ensures the spent fuel pool contains adequate dissolved boron anytime fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the fuel pool and a spent fuel pool verification has not been performed since the last movement of any fuel assembly in the spent fuel pool. The negative reactivity effect of the soluble boron would compensate for the increased reactivity caused by a mispositioned fuel assembly.

The boron concentration requirements of Specification 3.8.E.2 are no longer imposed when no fuel movements are occurring and a spent fuel pool verification has been completed, because the storage requirements of Specifications 3.8.E.1 and 5.6.A.1.d are then adequate to prevent criticality.

Specification 3.8.E.2.a is not imposed when only fuel assemblies with a combination of burnup and initial enrichment in the unrestricted range of Figure TS.3.8-1 are stored in the spent fuel pool. The requirements of Specification 3.8.E.2.a are not required in that case because with only fuel assemblies that have burnup and initial enrichment in the unrestricted range of Figure TS.3.8-1 it is not possible to cause an inadvertent criticality by mispositioning a fuel assembly in the spent fuel pool.

3.8 REFUELING AND FUEL HANDLING

Bases continued

When the requirements of Specification 3.8.E.2.a are applicable, and the concentration of boron in the spent fuel pool is less than required, 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. An acceptable alternative is to complete a spent fuel pool verification. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

A spent fuel verification is required following the last movement of fuel assemblies in the spent fuel pool, if fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in the spent fuel pool. This verification will confirm that any fuel assemblies with a combination of burnup and initial enrichment in the restricted range of Figure TS.3.8-1 are stored in accordance with the requirements of Specification 5.6.A.1.d.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 108 AND 101 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 June 11, 1993, as revised June 30, 1993, 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, Unit Nos. 1 and 2. The proposed changes would increase the maximum fuel enrichment specified in TS 5.3.A.2 from 4.25 weight percent (w/o) U^{235} to 5.0 w/o U^{235}. Due to this proposed change, TS 5.6.A and 3.8.E regarding new and spent fuel storage have been modified to accommodate the higher enriched fuel. Technical Specification table TS 4.1-2B and figures TS 3.8-1, TS 5.6-1 and TS 5.6-2 are being added or modified, as necessary, to update the surveillance and storage requirements. Additionally, proposed TS 3.3.A.1.a and TS 5.3.A.1 would increase the minimum refueling water storage tank (RWST) boron concentration from 1950 ppm to 2500 ppm and allow the use of ZIRLO clad and natural uranium. The staff's safety evaluation of the proposed changes follows. Bases for proposed TS 3.8.E.1 and 3.8.E.2 are being incorporated into the fuel handling specification bases.

2.0 EVALUATION

2.1 Reactor Core Changes

Prairie Island was previously licensed for the use of Zircaloy-4 clad material and slightly enriched uranium. TS 5.3.A.1 has been modified to allow the use of ZIRLO cladding material and natural uranium. The licensee states that the use of ZIRLO and natural uranium will be consistent with paragraph 4.2.1 of the Westinghouse Standard Technical Specifications, NUREG-1431. Furthermore, the licensee states that the requested change allowing the use of natural uranium is merely a clarification because natural uranium has been previously used as an axial blanket and replacement fuel pins.

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The ZIRLO clad material has evolved from Zircaloy-4 as a material that has better corrosion resistance and less radiation induced growth than its predecessor. ZIRLO has very similar neutronic properties to Zircaloy-4 and, therefore, should not significantly perturb the spectrum from the previously analyzed state. Furthermore, the licensee states that "...(the) use of ZIRLO clad fuel in the reactor cores will be evaluated with NRC approved methodologies prior to use." The staff considers the use of ZIRLO acceptable up to burnups of 60 GWD/MTU. This restriction will remain valid pending approval of extended burnup fuel cycles using Westinghouse VANTAGE+ fuel (ref. 1).

The proposed use of natural uranium is acceptable because reload core designs are evaluated for each cycle. The results of these analyses will be compared to the limits in the safety analyses and the TS to ensure that operation in the proposed fashion is acceptable.

2.2 RWST Boron Concentration Increase

The licensee has proposed, by modifying TS 3.3.A.1.a, increasing the minimum RWST boron concentration from 1950 ppm to 2500 ppm in anticipation that this change will be required due to an increase in fuel enrichment. Increasing the minimum RWST minimum boron concentration to 2500 ppm will provide adequate negative reactivity to ensure that the reactor will remain subcritical following a loss of coolant accident (LOCA) for reload cores utilizing fuel enriched to 5.0 w/o U^{235}. The licensee states that this will not cause any change to the current plant procedures because the RWST concentration is already administratively controlled at concentrations higher than 2500 ppm. The licensee states in its submittal that it will have to confirm that the 2500 ppm value is acceptable during the appropriate reload analysis. If the reload analysis concludes that the results are not consistent with the increase to 2500 ppm, then the licensee will need to submit a separate license amendment request. The staff finds this change to be acceptable.

2.3 Fuel Enrichment Increase

The licensee has modified TSs 5.6.A and 3.8.E in order to allow for the storage of fuel up to enrichments of 5.0 w/o U^{235}. To accommodate the higher enriched fuel, two fuel storage configurations have been proposed; one for new fuel and one for spent fuel. Westinghouse analyzed the proposed fuel storage configurations using the AMPX (using ENDF/B-V) code package (ref. 2) for cross section development and the KENO-Va Monte Carlo Transport Theory code (ref. 2) for reactivity calculations. The NITAWL (part of the AMPX package) program (ref. 2) began with 227 group cross sections and developed the geometrically specific energy self-shielded cross sections. Energy and spatial weighing was performed using the XSDRNPM (part of the AMPX package) S_n transport theory code (ref. 2). These codes are widely used for criticality analyses and have been bench-marked against the results from numerous critical experiments which closely simulate the Prairie Island spent fuel racks. All analyses were performed at the required 95/95 probability/confidence level.

TSs 5.6.A and 3.8.E have been modified to update the fuel storage requirements to accommodate the 5.0 w/o fuel. The proposed changes will require the 14 central locations in the new fuel storage area to be blocked and mandate a two-region, checkerboard loading spent fuel storage configuration. Spent fuel pool region 1 refers to the restricted, checkerboard loading area and spent fuel pool region 2 refers to the unrestricted area. Unrestricted when used in this context refers to fuel which can be placed in any configuration. Regional requirements will be prescribed by a combination of original enrichment and burnup. Figures TS 3.8-1 and TS 5.6-2 will be used to determine whether a fuel assembly will be considered restricted and, if so determined, whether it must be stored in a high burnup or low burnup location. Figure TS 5.6-1 shows the checkerboard loading pattern proposed and used in the Westinghouse analyses. The rows between the restricted and unrestricted regions must either be filled with alternating unrestricted and restricted burned fuel or be left empty.

Analyses of the proposed new fuel storage rack were performed with a KENO-Va model using the following assumptions:

1. Limiting assumptions about fuel type, density, and manufacturing were made. For example, no credit is taken for natural or slightly enriched uranium axial blankets.

2. No credit was taken for non-fuel neutron absorbing materials, i.e., rack structure, burnable and soluble poisons, and U^{234} and U^{236} or other fission products.

3. Lateral (X and Y) dimensions were infinite. The Z dimension was assumed infinite for the full density moderation calculation and was explicitly modeled for the optimal density moderation case.

The results of the analyses demonstrate that the proposed new fuel storage configuration meets the acceptance criteria which require that k_{eff} be less than 0.95 for full density conditions and less than 0.98 for optimal moderation conditions. The calculated k_{eff} for normal conditions was 0.9312, and 0.9690 for optimal moderation conditions. The staff has concluded that the proposed changes are acceptable because the referenced analysis is very conservative, significantly bounding and it meets the acceptance criteria.

Analyses of the proposed spent fuel storage rack, region 1, were also performed using the KENO-Va code. The following assumptions were made:

1. Limiting assumptions about fuel type (the analysis used OFA assemblies), density, and manufacturing were made. Once again, no credit was taken for axial blankets.

2. Limiting assumptions about pool moderator and structural conditions were made. For example, the moderator was assumed to be 1.0 g/cm^3 density water at 68°F (20°C) and Boraflex loading was assumed at a minimum.

3. No credit was taken for non-fuel neutron poisons such as rack structural material, U^{234} and U^{236}, and soluble poisons.

4. The rack was assumed infinite in the lateral (X and Y) dimensions and finite in the axial (Z) direction. A reflector of clean, full density water was assumed on the top and bottom of the stored fuel.

In order to account for burnup, a technique called reactivity equivalencing was used. This technique consists of plotting constant k_{eff} contours on a burnup vs. initial enrichment plot. This technique offers two advantages. One, it implicitly considers the effect that the rack has on reactivity.

Second, it facilitates simpler analysis because it allows one to analyze "fresh" assemblies that have equivalent reactivity to higher enriched, "burned" assemblies. This technique was used to calculate the "fresh" 2.5 w/o value used as "burned" fuel for the region 1 criticality calculation. Reactivity equivalencing was done with the PHOENIX code (ref. 3) and has been accepted in many other fuel storage evaluations.

For region 1, two cases were analyzed; one used nominal conditions and the other used worst case assumptions. The nominal calculations yielded a k_{eff} of 0.9042 using a 95/95 uncertainty value of ±0.0054. The worst case assumptions involved using the limiting values for manufacturing tolerances to limit Boraflex loading and optimize moderation. Furthermore, fuel enrichments were increased by 0.05 w/o to account for enrichment variations. Using these worst case assumptions, k_{eff} was calculated to 0.9448. K_{eff} for both cases was calculated to be below the acceptance criteria requiring a $k_{eff} < 0.95$. Optimal moderation conditions are not required to be considered for this case. The staff finds the proposed changes acceptable because the referenced analysis is suitably conservative and it meets the acceptance criteria.

Region 2 analyses consisted of using reactivity equivalencing to extend the constant k_{eff} contour on the existing Prairie Island burnup vs. enrichment curves to 5.0 w/o U^{235}. This was necessary because the existing curves only covered the range from 3.87 to 4.27 w/o U^{235} and this modified curve will be the basis for determining whether or not fuel can be stored in region 2 using the new storage technique. Because the proposed region 2 storage configuration will be bounded by the existing analyses (the extension of the existing burnup vs. enrichment curve ensures that the fuel storage configuration of region 2 will be no more reactive than what is currently used in the spent fuel pool), no further analyses were necessary.

One can conceive of scenarios in which rack reactivity will increase, for example, a misloaded fuel assembly or an assembly being dropped into the pool. The licensee has considered these events and because credit may be taken for the presence of a soluble poison (in this case, boron) for accident situations, the rack k_{eff} will remain well below the 0.95 acceptance criterion for these limiting cases. In order to ensure the presence of boron, the licensee's proposed TS 3.8.E.2 requires that boron be present in the pool from the onset of fuel movement until a spent fuel pool verification is performed to ensure that no assemblies have been mis-positioned. The boron

concentration during fuel movement is required to be 1800 ppm or greater. Furthermore, surveillance requirements (in proposed TS table 4.1-2B) during fuel movement and storage have been updated to require weekly tests of the boron concentration when boron is required to be present in the pool. The staff has reviewed the proposed changes and finds them acceptable.

Based upon the above discussion, the staff concludes that the proposed TS changes relating the fuel storage, core design, and RWST boron concentration are acceptable. The staff's conclusion is based upon the following:

1. The referenced analysis used established techniques that were properly verified.
2. The analysis was suitably conservative and bounding.
3. The results met NRC acceptance criteria with respect to K_{eff}.
4. The consequences of limiting accidents are acceptable.
5. The proposed TS are consistent with the analyses provided.

2.4 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of postulated accidents, included means for their avoidance in the design and operation of the facility, and provided means for mitigation of their consequences should they occur. The licensee has evaluated the effect of the changes on the calculated consequences of a spectrum of postulated design basis accidents (DBA) and concluded that the effect of the proposed TS changes is small. The licensee concluded that the calculated consequences are within regulatory requirements and staff guideline dose values. Since the licensee proposes to utilize higher enrichment fuel, the staff reevaluated the fuel handling accident for Prairie Island to consider the effects of increased enrichment and burnups.

In its evaluation for Prairie Island, issued on September 28, 1972, the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

Since the applicant intends to utilize higher enrichment fuel, the staff reanalyzed the fuel handling DBA for this case. According to NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," (February 1989) increasing fuel enrichment to 5.0 w/o U^{235} with a maximum burnup of 60,000 MWD/T increases the doses for a fuel handling accident by a factor of 1.2.

The licensee proposes to limit enrichment to 5.0 w/o U^{235}. Therefore, the 1.2 factor increase in dose displayed in Table 1 below, bounds the dose consequences of the licensee's proposal. In Table 1, the new and old DBA doses are presented and compared to the guidelines doses in NUREG-0800 (established based on 10 CFR Part 100).

TABLE 1
RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING DESIGN BASIS ACCIDENT (REM)

	Exclusion Area Thyroid	Low Population Zone Thyroid
Staff Evaluation September, 1972	33	6
Bounding Estimates for Higher Enrichment Fuel Burnup*	39.6	7.2
Regulatory Requirement (NUREG-0800 Chapter 15.7.4)	75	75

* Factor of 1.2 greater than original estimate for iodine.

The staff concludes that the only potential increased doses resulting from the fuel handling accidents with increased enrichment are the thyroid doses; these doses remain well within the dose limits set forth in NUREG-0800 and are, therefore, acceptable.

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 was published in the Federal Register on August 30, 1993 (58 FR 45536).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of these amendments 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. Letter from USNRC to S.R. Tritch (Westinghouse Electric Corporation), accepting the "referencing of topical report WCAP-12610 VANTAGE+ Fuel Assembly Reference Core Report," July 1, 1991.

2. NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, Volume 1," Oak Ridge National Laboratory, 1982.

3. WCAP-11956, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," Westinghouse Electric Corporation, 1987 (Proprietary).

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Date: September 3, 1993