August 28, 1989

Docket Nos. 50-282
and 50-306

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Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

SUBJECT: AMENDMENT NOS. 90 AND 83 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60: INCREASING THE ENRICHMENT OF THE FUEL ASSEMBLIES FROM 3.9 TO 4.25 WEIGHT PERCENT URANIUM-235 (TAC NOS. 72972 AND 72973)

The Commission has issued the enclosed Amendment No. 90 to Facility Operating License No. DPR-42 and Amendment No. 83 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 (TSs) in response to your application dated April 6, 1989.

The amendments change the TSs by permitting the irradiation of the fuel assemblies with enrichment up to 4.25 weight percent (w/o) Uranium-235 and the storage of such assemblies prior to and subsequent to loading the assemblies in the reactors at the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2.

A copy of our related Safety Evaluation and Notice of Issuance are enclosed. The issuance of these amendments completes our work effort under TACs Nos. 72972 and 72973.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 90 to License No. DPR-42
2. Amendment No. 83 to License No. DPR-60
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

DOCUMENT NAME: P.I. AMEND TAC 72973/73

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
August 28, 1989

Docket Nos. 50-282
and 50-306

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Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

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cc w/enclosures:
See next page

Mr. T. M. Parker
Northern States Power Company

Prairie Island Nuclear Generating
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
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 April 6, 1989, 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. 90, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. Di Ianni /L.Y.

Lawrence A. Yandell, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 28, 1989

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
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 April 6, 1989, 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. 83 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. Dilanni /L.Y.

Lawrence A. Yandell, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 28, 1989

ATTACHMENT TO LICENSE AMENDMENT NOS. 90 AND 83

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

DOCKET NOS. 50-282 AND 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 marginal lines indicating the area of change.

REMOVE	INSERT
TS.3.8-3	TS.3.8-3
TS.3.8-4	TS.3.8-4
TS.3.8-5	TS.3.8-5
-	TS.3.8-6
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

D. Spent Fuel Pool Special Ventilation System

1. Except as specified in Specification 3.8.D.3 below, both trains of the Spent Fuel Pool Special Ventilation System and the diesel generators required for their operation shall be operable at all times.

2. a. The results of in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks respectively shall show $\geq$ 99% DOP removal for particles having a mean diameter of 0.7 microns and $\geq$ 99% halogenated hydrocarbon removal.

   b. The results of laboratory carbon sample analysis shall show $\geq$ 90% radioactive methyl iodide removal efficiency (130°C, 95% RH).

   c. The Spent Fuel Pool Special Ventilation System fans shall operate within ± 10% of 5200 cfm per train.

3. From and after the date that one train of the Spent Fuel Pool Special Ventilation System is made or found inoperable for any reason, fuel handling operations are permissible only during the succeeding seven days (unless such train is made operable) provided that the redundant train is verified to be operable daily.

4. If the conditions for operability of the Spent Fuel Pool Special Ventilation System cannot be met, fuel handling operations in the Auxiliary Building shall be terminated immediately.

E. Storage of Low Burnup Fuel

1. The following restrictions shall apply whenever fuel with an average assembly burnup less than 5,000 MWD/MTU is stored in the spent fuel pool (except as specified in 3.8.E.2 and 3.8.E.3 below):

   a. The boron concentration in the spent fuel pool shall be maintained greater than or equal to 500 ppm, and

   b. Fuel with an average assembly burnup less than 5,000 MWD/MTU shall not be stored in more than three storage locations of every four by four storage rack array.

2. If the conditions in 3.8.E.1.a above are not met, verify that the spent fuel pool storage configuration meets the requirements of specification 3.8.E.1.b and suspend all actions involving the movement of fuel in the spent fuel pool until the boron concentration is increased to 500 ppm or greater.

3. If the conditions in 3.8.E.1.b above are not met, suspend all actions involving movement of fuel in the spent fuel pool, verify the spent fuel pool boron concentration to be greater than or equal to 500 ppm and initiate corrective actions. Mis-positioned fuel assemblies shall be moved to acceptable locations prior to the resumption of other fuel movement in the spent fuel pool.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the precautions specified above, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (B. above) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

Under rodded and unrodded conditions, the K_{eff} of the reactor must be ≤ 0.95 and the boron concentration must be ≥ 2000 ppm as indicated in A.4. Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. A.9 above allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

No movement of fuel in the reactor is permitted until the reactor has been subcritical for at least 100 hours to permit decay of the fission products in the fuel. The delay time is consistent with the fuel handling accident analysis.(2)

The spent fuel assemblies will be loaded into the spent fuel cask after sufficient decay of fission products. While inserting and withdrawing the cask into pool No. 1, the cask will be suspended above the bottom of the pool up to a maximum of 42 feet. The consequences of potential load drops have been evaluated in accordance with NUREG-0612(5). Following is a discussion of the basis for the limitations which resulted from that evaluation.

The cask will not be inserted into the pool until all fuel stored in the pool has been discharged from the reactor a minimum of 5 years. Supporting analysis indicated that fuel stored in the pool for a period as short as 50 days would allow sufficient decay of the fission products such that their release would result in off-site doses less than 25% of the 10 CFR Part 100 guidelines. The five year decay period was selected in following the general principle that spent fuel with the longest decay time would result in the least off-site doses in the event of an accident, while providing the plant operational flexibility. The cask will not be inserted or withdrawn from the pool unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that if fuel is crushed by a cask drop, K_{eff} will be less than or equal to 0.95. The cask will not be inserted or withdrawn from the pool unless a cask impact limiter, crash pad, or combination thereof is in place with the capability to absorb energy of a cask drop such that no significant amount of water leakage results from pool structural damage. This is to ensure that at no time will water level drop below the top of the spent fuel stored in the pool. In loading the cask into a carrier, there is a potential drop of 66 feet(4). The cask will not be loaded onto the carrier for shipment prior to a 3-month storage period.

At this time, the radioactivity has decayed so that a release of fission products from all fuel assemblies in the cask would result in off-site doses less than 10 CFR Part 100. It is assumed, for this dose analysis, that 12 assemblies rupture after storage for 90 days. Other assumptions are the same as those used in the dropped fuel assembly accident in the SER, Section 15. The resultant doses at the site boundary are 94 Rems to the thyroid and 1 Rem whole body.

The number of recently discharged assemblies in Pool No. 1 has been limited to 45 to provide assurance that in the event of loss of pool cooling capability, at least eight hours are available under worst case conditions to make repairs until the onset of boiling.

The Spent Fuel Pool Special Ventilation System (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. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

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 bases for these allowances 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 requirements for the storage of low burnup fuel in the spent fuel pool ensure that the spent fuel pool will remain subcritical during fuel storage. Fuel stored in the spent fuel pool will be limited to a maximum enrichment of 4.25 weight percent U-235. It has been shown by criticality analysis that the use of the three out of four storage configuration will assure that the K_{eff} will remain less than 0.95, including uncertainties, when fuel with a maximum enrichment of 4.25 weight percent U-235 and average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool.

The requirement for maintaining the spent fuel pool boron concentration greater than 500 ppm whenever fuel with average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool ensures that K_{eff} for the spent fuel pool will remain less than 0.95, including uncertainties, even if a fuel assembly is inadvertently inserted in the empty cell of the three out of four storage configuration.

References

(I) FSAR Section 9.5.2
(2) FSAR Section 14.2.1
(3) FSAR Section 9.6
(4) FSAR Page 9.5-20a
(5) Exhibit C, NSP License Amendment Request Dated December 21, 1984

5.3 REACTOR

A. Reactor Core

1. The reactor core contains uranium in the form of slightly enriched uranium dioxine pellets. The pellets are encapsulated in Zircaloy-4 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 4.25 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 applcable 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

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. The design of the new fuel storage pit and racks (Reference 1) ensures a new fuel pit K_{eff} of less than or equal to 0.95, including uncertainties, even if unborated water were used to fill the pit. The new fuel rack configuration also ensures K_{eff} less than or equal to 0.98, including uncertainties, even if the new fuel racks were accidentally filled with a low density moderator which resulted in optimum low density moderation conditions. Fuel stored in the new fuel storage racks will have a maximum enrichment of 4.25 weight percent U-235.

The spent fuel storage rack design (Reference 1) and the limitations on the storage of low burnup fuel contained in Technical Specification Section 3.8.E ensure a spent fuel pool K_{eff} of less than or equal to 0.95, including uncertainties. The maximum enrichment of fuel to be stored in the spent fuel pool will be 4.25 weight percent U-235.

The criticality considerations as they relate to the dropping of a spent fuel cask (i.e., heavy load) drop onto the racks has been evaluated. The maximum K_{eff} has been calculated to be less than 0.900 for water/UO_2 ratios of between 2.0 and 2.3 with a boron concentration of 1800 ppm.

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. In addition, the spent fuel cask will have an impact limiter attached or a crash pad will be in place in the pool which will have the capability to absorb energy of impact due to a cask drop. This will result in no structural damage taking place to the pool which would result in significant leakage from the pool. Piping to the pool is arranged so that failure of any pipe cannot drain the pool below the tops of the stored fuel assemblies.

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 shipping casks for removal from the site. The casks will be handled by the auxiliary building crane.

The load drop consequences of a spent fuel cask for Prairie Island have been evaluated. It is not possible, due to physical constraints, for a cask to be dropped into the large pool (pool no. 2). A load path has been defined which provides for safe movement of the cask. Travel interlocks and mechanical stops prevent cask movement outside of this path. The only safety-related equipment that can be impacted directly during a cask drop along this path is the fuel stored in the small pool (pool no. 1). The consequences of this drop have been evaluated and found to meet the NRC staff criteria contained in NUREG-0612 if at least 50 days have elapsed since reactor shutdown for fission gas release considerations and the pool water contains at least 1800 ppm boron for criticality considerations. While 50 days was determined adequate, a minimum decay period of 5 years has been incorporated into these technical specifications to provide additional margin in meeting the criteria specified in NUREG-0612 for fission gas releases, while not restricting the plant's operational flexibility. A cask impact limiter or crash pad prevents significant structural damage to the pool floor.

The spent fuel cask will be lowered 66 feet from the auxiliary building to the railroad car for offsite transportation. Specification 3.8 will limit this loading operation so that if the cask drops 66 feet, there will not be a significant release of fission products from the fuel in the cask.

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 shipping 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 90 AND 83 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 April 6, 1989 (Ref. 1), Northern States Power Company (NSP), the licensee, requested a change to Facility Operating License Nos. DPR-42 and DPR-60 which would change Specifications 5.6.A and 5.3.A.2 and add a new Specification 3.8.E and associated Bases to the Technical Specifications for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PING). The proposed changes would permit the reload of fuel assemblies with enrichments up to 4.25 weight percent (w/o) Uranium-235 and the storage of such assemblies prior to and subsequent to loading in the PING reactors. The increase in fuel enrichment is needed to support longer fuel cycles for the two plants.

2.0 EVALUATION

2.1 Analytical Methods

The analytical methods used in the criticality analysis of the PING fuel storage racks use the AMPX (Ref. 2) system of codes for neutron cross section generation and the KENO-IV (Ref. 3) Monte Carlo computer code for reactivity determination. The 227 energy group neutron cross section library used in the AMPX system of codes is based on ENDF/B-V (Ref. 4). For processing the 227 group neutron cross section library to obtain multigroup cross sections for evaluating criticality experiments and the PING fuel storage racks, the NITAWL code (Ref. 2) is used to provide the self-shielded resonance cross sections.

NITAWL uses the Nordheim Integral Treatment for the resonances. The XSDRNPM (Ref. 2) one-dimensional S_n code is used to perform the energy and spatial weighting of cross sections. The multigroup neutron cross sections generated for a particular configuration are then input to the KENO-IV (Ref. 3) Monte Carlo code to evaluate the criticality of the critical experiments and PING fuel storage racks.

These methods were benchmarked by Westinghouse, the vendor performing the analysis of the fuel storage racks, by analyzing 33 critical experiments. These experiments covered water moderated, uranium-oxide fuel arrays separated by various materials that simulate light water reactor (LWR) fuel shipping and storage conditions (Ref. 5) to critical experiments using highly enriched uranium metal cylindrical arrays with various interspersed materials (Ref. 6). The results of the analysis of these critical experiments are: (1) the average calculated effective multiplication factor (k_{eff}) of the critical experiments is 0.992, (2) the standard deviation of the bias value is 0.0008 delta-k, and (3) the 95 percent probability with a 95 percent confidence level (95/95 probability/confidence level) uncertainty in reactivity of the analytical methods is 0.0018 delta-k.

The reactivity equivalencing performed to determine the fuel assembly discharge burnup as a function of initial fuel enrichment and poison gap size was performed with the PHOENIX (Ref. 8) and CINDER (Ref. 9) computer codes. A 25 energy group nuclear data library based on the British WIMS (Ref. 10) code is used with PHOENIX. The PHOENIX code is a depletable, two-dimensional, multigroup, discrete ordinate, transport theory code. The CINDER code is a point-depletion computer code to determine fission product activities. Using these two codes, Westinghouse determined that, as a function of discharge burnup, fuel assemblies had their maximum reactivity at 100 hours after discharge from a reactor. This time of maximum reactivity occurs at the point in time when the major fission product poison Xenon-135 has almost completely decayed away. Therefore, the most reactive point in time for a fuel assembly discharged from the reactor is conservatively approximated by Westinghouse by removing the Xenon-135 from the calculations.

The PHOENIX code was validated by Westinghouse by analyzing extensive benchmark critical experiments and by analyzing the isotopic composition that has been measured for fuel discharged from a reactor (Ref. 11). The data that has been analyzed indicate good agreement with measurements for both the critical experiments and the isotopic data. Westinghouse includes an additional bias of 500 MWD/MTU to account for uncertainty associated with burnup dependent reactivities.

We conclude that the use of the AMPX system of codes and the KENO-IV Monte Carlo code for the PING fuel storage racks is acceptable because the results obtained for the critical experiments are satisfactory for these codes that are widely used by the industry for fuel storage rack analyses. In addition, we conclude that the use of the PHOENIX and CINDER codes is acceptable because the results obtained for the critical experiments and isotopic compositions are satisfactory for these codes.

2.2 Spent Fuel Racks

The PING spent fuel storage racks have a nominal 9.5 inch center-to-center spacing. Boraflex strips are encased in each wall of a storage cell. The Boraflex contains the neutron absorber Boron-10 at a loading of 0.040 grams per square centimeter. The spent fuel racks can provide storage for 1582 fuel assemblies. Total storage is limited to 1386 fuel assemblies to allow insertion of a shipping cask in the spent fuel pool. The PING spent fuel racks are currently licensed to store fuel assemblies which do not exceed 39 grams of Uranium-235 per axial centimeter of fuel assembly. This corresponds to an enrichment of about 3.9 w/o Uranium-235 for the 14x14 fuel assemblies used at PING. The present submittal addresses the following issues; (1) the maximum enrichment fresh fuel assembly that can be stored in the spent fuel racks;

(2) the maximum enrichment fuel assembly that can be stored in the spent fuel racks with poison gaps (gaps in the Boraflex) with credit for fuel burnup; and (3) the maximum enrichment fuel assembly that can be stored in a modified checkerboard loading using 3-out-of-4 storage locations with poison gaps in the Boraflex.

2.2.1 Maximum Enrichment Fresh Fuel Assembly for Spent Fuel Racks

Analyses were performed to determine the maximum enrichment fresh fuel assembly that could be stored in the spent fuel racks. The analyses were performed for Westinghouse OFA fuel assemblies which give a larger k_{eff} than does the Westinghouse STD fuel assembly when both fuel assemblies have the same Uranium-235 enrichment. Other assumptions that were made are:

1. All fuel rods contain uranium dioxide fuel with the same Uranium-235 enrichment over the entire length of each fuel rod.

2. No credit is taken for the Uranium-234 or Uranium-236 in the fuel or for burnup.

3. The spent fuel pool water is at a temperature of 68°F with a conservative value of 1.0 gm/cc for the water density.

4. No credit is taken for spacer grids or sleeves.

5. The storage cells from an infinite array in the lateral dimensions. The axial dimension is taken to be finite.

6. A minimum poison material loading of 0.040 grams of Boron-10 per square centimeter is used for the Boraflex.

7. No credit is assumed for the soluble boron in the pool water.

Westinghouse considered the worst case location of a fuel assembly in the spent fuel racks and the worst case dimensions of the storage racks based on construction tolerances. Based on this worst case analysis, Westinghouse determined that an enrichment of 4.07 w/o Uranium-235 results in a k_{eff} of 0.9331. To this worst case k_{eff} biases are added to account for the biases in the calculational method and poison particle self-shielding. These biases add 0.0093 delta-k to the worst case k_{eff}. In addition, uncertainties at the 95/95 probability/confidence level are added to k_{eff} to account for the uncertainty in the method and in the KENO-IV Monte Carlo uncertainty on the worst case k_{eff}. These uncertainties add 0.0053 delta-k to the worst case k_{eff}. The corrected k_{eff}, including biases and uncertainties, is 0.9477.

We conclude that the unrestricted storage of fresh fuel assemblies (OFA and STD) enriched to 4.07 w/o Uranium-235, if no poison gaps exists in the spent fuel racks, is acceptable because the k_{eff}, including biases and uncertainties, of the spent fuel racks is less than the staff criterion of 0.95 and because suitably conservative analysis assumptions have been made.

Westinghouse analyzed accidents that could increase the reactivity of the spent fuel racks. For accident conditions, the staff position allows credit for the soluble boron in the spent fuel pool water. A heavy load dropped on top of the spent fuel racks was postulated. The methodology used by Westinghouse is the same as was previously used by Quadrex (Ref. 7). The same assumptions that were used to establish the 4.07 w/o fresh fuel assembly enrichment limit were used to analyze the heavy load drop except that the model is infinite in all directions. Calculations were performed with and without soluble boron as a function of water to uranium dioxide volume ratio. Additional uncertainties of 0.005 delta-k are added for the no boron result and 0.010 delta-k to the boron result for conservatism. The results show that the maximum spent fuel rack reactivity will be less than 0.90, including uncertainties at a 95/95 probability/confidence level, with 1800 ppm of boron in the spent fuel pool water.

We conclude that a heavy load drop on the spent fuel storage racks loaded with fresh fuel assemblies enriched to 4.07 w/o Uranium-235 is acceptable because k_{eff} is less than the staff criterion of 0.95 when credit is taken for 1800 ppm of soluble boron in the spent fuel pool water.

2.2.2 Maximum Enrichment Fuel Assemblies with Poison Gaps and Credit for Discharge Burnup

Analyses were performed to determine the maximum reactivity effect of gaps in the Boraflex. These analyses were also for the purpose of bounding expected shrinkage of the Boraflex, according to discussions with the licensee. Westinghouse determined that some of the enrichment and poison gap sizes exceeded the criterion on k_{eff} of being less than or equal to 0.95. The k_{eff} limit can be met in any one of three ways: (1) by taking credit for fuel burnup; (2) by taking credit for soluble boron in the pool water for fuel assembly misloadings; and (3) by a checkerboard loading of the fuel assemblies.

The PHOENIX (Ref. 8) and CINDER (Ref. 9) codes were used to determine the amount of fuel burnup required for a given fuel assembly enrichment and maximum assumed poison gap size using the method of reactivity equivalency. In this method reactivity calculations are performed to generate a set of fuel assembly enrichment and associated fuel assembly discharge burnup for an assumed poison gap size which will yield k_{eff} equal to or less than 0.95, including all uncertainties at the 95/95 probability/confidence level. Curves of fuel assembly discharge burnup as a function of initial fuel assembly enrichment in weight percent Uranium-235 for poison gap sizes of 0, 2 and 4 inches which meet the spent fuel storage rack criterion were generated. The results of the analysis show, as an example, that a fuel assembly, with an initial enrichment of 4.27 w/o Uranium-235 and with a 4 inch gap in the boraflex poison centered at the fuel assembly midplane throughout every spent fuel storage location, would meet the spent fuel rack k_{eff} criterion if the discharge burnup was at least 4,272 MWD/MTU.

Thus, using the AMPX/KENO-IV code to determine the storage rack k_{eff} for fresh fuel and the PHOENIX/CINDER codes to provide the discharge burnup in the reactivity equivalency analysis, the licensee has determined the storage requirements in the spent fuel pool when credit for discharge burnup is taken. In the analyses the k_{eff} of 0.95 includes all applicable biases and uncertainties at the 95/95 probability/confidence level. The analyses include the same assumptions as discussed in the previous section except that the axial gaps in the poison panels are positioned in the axial center of the active fuel in all poison panels in the spent fuel racks.

We conclude that the storage of OFA and STD fuel assemblies in the Prairie Island spent fuel pool is acceptable for the combination of initial fuel enrichment, fuel assembly discharge burnup, and potential poison gap size discussed in the Westinghouse report because the criterion of k_{eff} being equal to or less than 0.95, including all uncertainties at the 95/95 probability/confidence level, is met.

2.2.3 Modified Checkerboard Loaded Spent Fuel Racks

Westinghouse performed analyses to show that fresh fuel assemblies with an enrichment of 4.27 w/o Uranium-235 and 4 inch gaps in the poison panels could be stored in a modified loading of the spent fuel racks such that 3-out-of-4 storage locations were occupied by fresh fuel and the fourth location was empty. Except for enrichment, loading pattern and poison gaps the assumptions are the same as those discussed in Section 2.2.1. The axial gaps (4 inches) in the poison panels are positioned in the axial center of the active fuel in all poison panels in the spent fuel racks.

Westinghouse considered the worst case location of a fuel assembly in the spent fuel racks and the worst case dimensions of the storage racks based on construction tolerances. Based on this worst case analysis, Westinghouse determined that k_{eff} was equal to 0.9016. Biases are added to this worst case k_{eff} to account for biases in the calculational method and poison particle self-shielding. These biases add 0.0093 delta-k to the worst case k_{eff}. Finally, uncertainties at the 95/95 probability/confidence level are added to k_{eff} to account for uncertainty in the method and in the KENO-IV Monte Carlo uncertainty on the worst case k_{eff}. These uncertainties add 0.0049 delta-k to the worst case k_{eff}. The corrected k_{eff}, including all biases and uncertainties is 0.9158.

We conclude that the storage of fresh fuel assemblies (OFA and STD) enriched to 4.27 w/o Uranium-235 in a modified checkerboard array using only 3 of every 4 locations is acceptable because the k_{eff}, including biases and uncertainties, of this spent fuel rack configuration is less than the staff criterion of 0.95 and because suitably conservative analysis assumptions have been made.

2.2.4 Fuel Misloading and Credit for Soluble Boron

The modified checkerboard loading pattern without physically blocked unused locations and the taking of credit for a fuel assembly's discharge burnup means that fuel misloading errors are now possible for the PING spent fuel racks. The fuel misloading event is not discussed as such in the licensee's submittal. The limiting case is the misloading of fresh fuel assemblies with an enrichment of 4.27 w/o Uranium-235 in the racks if 4 inch axially centered poison gaps exist in every racks' poison panels. In such a misloading case credit may be taken for the soluble boron in the pool water. The licensee calculated that 250 ppm of soluble boron in the water, including a 10 percent conservatism in the soluble boron concentration, would yield a k_{eff} equal to 0.950. The limiting case also shows that k_{eff} is equal to 0.9810 when no credit is taken for the soluble boron in the water. Thus, the spent fuel pool racks would remain subcritical even without taking credit for soluble boron in the water in the event of the limiting case misloading event. The PING Technical Specifications will require a minimum soluble boron content of 500 ppm. A Surveillance

Requirement was added to the Technical Specifications. Based on these requirements, we conclude that administrative procedures are acceptable for the storage of fuel assemblies at PING in a 3 of every 4 configuration or for the storage of fuel assemblies with discharge burnups equal to or greater than 5,000 MWD/MTU. In addition, we conclude that credit may be taken for the soluble boron in the water for spent fuel rack misloading events.

2.3 New Fuel Racks

The licensee has provided an analysis of the criticality of the new fuel storage racks for both the fully flooded and low hydrogenous moderation conditions. The analyses used the same calculational methods and neutron cross section libraries that were used in the analyses for the spent fuel racks. The analyses were based on the following assumptions:

1. The fuel assembly containing the highest enrichment authorized, is at its most reactive point in life, and no credit is taken for any burnable poison in the fuel rods.

2. All fuel rods contain uranium dioxide at an enrichment of 4.27 w/o Uranium-235 over the entire length of each rod.

3. No credit is taken for any Uranium-234 or Uranium-236 in the fuel, nor is any credit taken for the buildup of fission products.

4. No credit is taken for any spacer grids or spacer sleeves.

For the fully flooded condition the water is taken to be at 68°F and at a density of 1.0 gm/cc. The analysis was performed for Westinghouse OFA fuel assemblies in an array that is infinite in both the lateral and axial dimensions. Westinghouse considered the worst case location of a fuel assembly in the new fuel racks and the worst case dimensions of the racks based on construction tolerances. Based on this worst case analysis, Westinghouse determined that an enrichment of 4.27 w/o Uranium-235 results in a k_{eff} of 0.8736. To this worst case k_{eff} a bias of 0.0083 delta-k is added to account for the bias in the calculational method. In addition uncertainties at the 95/95 probability/ confidence level are added to k_{eff} to account for the uncertainty in the method and in the Monte Carlo uncertainty on the worst case k_{eff}. These uncertainties add 0.0076 delta-k to the worst case k_{eff}. The corrected k_{eff}, including biases and uncertainties, is 0.8895. This k_{eff} is well below the staff criterion of 0.95 for the fully flooded case.

A similar analysis was performed for the case of low density hydrogenous moderation. For this case it was determined that a Westinghouse STD fuel assembly was more reactive than an OFA fuel assembly. Westinghouse determined that a full 8x11 array of fuel assemblies would exceed the staff criterion of k_{eff} less than or equal to 0.98. Westinghouse determined that a 5x11 array of fuel assemblies at the low density of maximum reactivity of 0.075 gm/cc would meet the criterion. The three deleted rows of fuel storage locations will be modified to prevent storage of fuel assemblies. The calculated k_{eff} was 0.9794 including the worst case k_{eff} of 0.9634, a bias of 0.0083 delta-k and an uncertainty at a 95/95 probability/confidence level of 0.0077 delta-k.

We conclude that the storage of new STD and OFA fuel assemblies having a maximum enrichment of 4.27 w/o Uranium-235 in no more than a 5x11 array of new fuel storage locations is acceptable because it will meet the staff criterion on k_{eff} of less than or equal to 0.95 for the fully flooded case and of k_{eff} of less than or equal to 0.98 for the low density hydrogenous moderation case.

2.4 Technical Specifications

The Technical Specifications for this proposed license amendment were reviewed and discussed with the licensee. As a result of these discussions a number of wording changes were made to the Specifications. All of these changes were agreed to with the licensee. Based on our review, we found the following Specifications to be acceptable.

1. Specification 3.8.E and Bases Statements

This Specification provides limitations on the storage of a fuel assembly with a burnup of less than 5000 MWD/MTU. The first limitation is that the concentration of boron in the pool shall be equal to or greater than 500 ppm.

The other limitation specifies that storage of such fuel shall be in a 3 of every 4 array with the fourth storage location empty. A monthly Surveillance Requirement is added to Table 4.1-2b for the pool boron concentration. The Bases statements for this Specification are consistent with the requirements of the Specification. This Specification and associated Bases are acceptable because they are based on the safety analysis for the spent fuel racks.

2. Specification 5.3.A.2

This Specification has been changed to allow a maximum fuel assembly enrichment of 4.25 w/o Uranium-235 in the reactor core. This change is acceptable because it is consistent with the safety analysis for the new and spent fuel racks and because cycle specific reload safety analyses will be performed for the maximum allowed enrichment fuel to be placed in the reactor core to ensure that all applicable safety criteria are met.

3. Specification 5.3.A.1

This change deletes the reference to the total weight of uranium in the core. This change is not related to the safety analysis for the new and spent fuel racks rather it is an editorial change. This is acceptable because the exact weight of uranium in the core has little significance to the safety of the reactor.

4. Specification 5.6.A

This Specification has been changed to allow a maximum enrichment of 4.25 w/o Uranium-235 for the new fuel storage pit and racks. This change is acceptable because it is supported by the safety analysis.

2.5 Design Basis Accident Analysis Relative to Fuel Burnup

We have evaluated the potential impact of fuel burnup on the radiological basis accident for the Prairie Island Nuclear Generating Plant. By letter dated December 28, 1983, the Commission issued Amendment Nos. 67 and 61 that evaluates off-site doses from potential radiological consequences for fuel burnup up to 55 GWD/MTU. The evaluation concludes that the extended burnup will not result in higher doses from those previously analyzed for postulated accidents nor will doses exceed the dose guidelines of 10 CFR 100.11. This conclusion is still applicable for this amendment since the licensee does not intend to irradiated fuel assemblies above the 55 GWD/MTU limit.

3.0 SUMMARY

Based on the review described above, the proposed Technical Specification modifications are acceptable and fuel assemblies having initial enrichments up to 4.25 weight percent Uranium-235 may be operated in the reactors and safely stored in the new fuel storage pit if the requirements of the Technical Specifications are met. The Surveillance Requirement for Specification 3.8.E requires that the spent fuel pool boron concentration be verified monthly. In addition, the new fuel storage racks will be modified to preclude the storage of more than an array of 5x11 new fuel assemblies by deleting a 3x11 array of storage cells.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, and environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on August 28, 1989 (54 FR 35543) . Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have 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, and (2) such activities will not be conducted in compliance with the Commission's regulations and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: August 28, 1989

6.0 REFERENCES

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UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DOCKET NOS. 50-282 AND 50-306

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 90 and 83 to Facility Operating License Nos. DPR-42 and DPR-60 issued to Northern States Power Company (licensee) which revised the Technical Specifications for operation of the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 located in Goodhue County, Minnesota.

The amendment is effective as of the date of issuance.

The amendment revised the Technical Specification by permitting the irradiation of the fuel assemblies with enrichment up to 4.25 weight percent (w/o) Uranium-235 and the storage of such assemblies prior to and subsequent to loading the assemblies in the reactors at the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on June 27, 1989 (54 FR 27083). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendments dated April 6, 1989, (2) Amendment Nos. 90 and 83 to Facility Operating License Nos. DPR-42 and DPR-60, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW, Washington, D.C., and at the Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V & Special Projects.

Dated at Rockville, Maryland this August 28, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

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