MAY 1 3 1981

Docket Nos. 50-282
and 50-306

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendments Nos. 48 and 42 to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated January 31, 1980 and as supplemented by letters dated June 10 and November 21, 1980, January 14, February 3, March 10, March 31 and April 20, 1981.

These amendments will allow an increase in the spent fuel storage capability up to a maximum of 1120 fuel assemblies in the spent fuel pool through the use of high density borated spent fuel racks. Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

Your application of January 31, 1980 requested an increase in the spent fuel pool storage capacity from the previously authorized capacity of 687 fuel assemblies to 1582 fuel assemblies. However, due to our evaluation of the handling of heavy loads issue we have limited our approval of the number of spent fuel assemblies which may be stored to 1120. The reasons for this limitation are discussed in Section 3.3 of the attached Safety Evaluation. Accordingly our Safety Evaluation and Environmental Impact Appraisal were prepared considering the higher number (1582) of fuel assemblies except for the heavy loads handling review which was based on a limit of 1120 fuel assemblies.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Copies of the Safety Evaluation, Environmental Impact Appraisal, and the Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Original signed by
Robert A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 48 to DPR-42
2. Amendment No. 42 to DPR-60
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice and Negative Declaration

cc: w/enclosures
See next page

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

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Docket No. 50-282 and 50-306

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHERN STATES POWER COMPANY, Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendments Nos. 48 and 42.
Referenced documents have been provided PDR.

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PMKreutzer/pn					
DATE→	5/15/81					

NRC FORM 102 7—79

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

May 13, 1981

Docket Nos. 50-282
and 50-306

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendments Nos. 48 and 42 to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated January 31, 1980 and as supplemented by letters dated June 10 and November 21, 1980, January 14, February 3, March 10, March 31 and April 20, 1981.

These amendments will allow an increase in the spent fuel storage capability up to a maximum of 1120 fuel assemblies in the spent fuel pool through the use of high density borated spent fuel racks. Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

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Sincerely,

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 48 to DPR-42
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3. Safety Evaluation
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5. Notice and Negative Declaration

cc: w/enclosures
See next page

Northern States Power Company

cc:

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Ms. Terry Hoffman
Executive Director
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

The Environmental Conservation Library
Minneapolis Public Library
300 Nicollet Mall
Minneapolis, Minnesota 55401

Mr. F. P. Tierney, Plant Manager
Prairie Island Nuclear Generating Plant
Northern States Power Company
Route 2
Welch, Minnesota 55089

Joclyn F. Olson, Esquire
Special Assistant Attorney General
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minneosta 55113

Robert L. Nybo, Jr., Chairman
Minnesota-Wisconsin Boundary Area
Commission
619 Second Street
Hudson, Wisconsin 54016

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #2, Box 500A
Welch, Minnesota 55089

Mr. John C. Davidson, Chairman
Goodhue County Board of Commissioners
321 West Third Street
Red Wing, Minnesota 55066

Bernard M. Cranum
Bureau of Indian Affairs, DOI
831 Second Avenue South
Minneapolis, Minnesota 55402

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U.S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming
dated: 1/31/80, 6/10/80. 3/21/80, 1/14/81
2/3/81, 3/10/81, 3/31/81, 4/20/81
Chairman, Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison Wisconsin 53702

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. 48
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 January 31, 1980 as supplemented by filings dated June 10, 1980, November 21, 1980, January 14, February 3, March 10, March 31 and April 20, 1981, 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, Facility Operating License No. DPR-42 is amended by changes to the Technical Specifications as indicated in the attachment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 , 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

T M Novak

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 13, 1981

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. 42
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 January 31, 1980 as supplemented by filings dated June 10, 1980, November 21, 1980, January 14, February 3, March 10, March 31 and April 20, 1981, 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, Facility Operating License No. DPR-60 is amended by changes to the Technical Specifications as indicated in the attachment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 42 , 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

T M Novak

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 13, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Pages

3.8-2
5.3-1
5.6-1
5.6-3

9. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

10. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 100 hours.

11. The radiation monitors which initiate isolation of the Containment Purge System shall be tested and verified to be operable immediately prior to a refueling operation.

B. During fuel handling operations, the following conditions shall be satisfied:

1. No heavy loads will be transported over or placed in either part of the spent fuel pool when irradiated fuel is stored in that part.*

2. Prior to spent fuel handling in the auxiliary building, tests shall be made to determine the operability of the spent fuel pool special ventilation system including the radiation monitors in the normal ventilation system that actuate the special system and isolate the normal systems.

3. Prior to fuel handling operations, fuel-handling cranes shall be load-tested for operability of limit switches, interlocks, and alarms.

4. When the spent fuel cask contains one or more fuel assemblies, it will not be suspended more than 30 feet above any surface until the fuel has decayed more than 90 days.

5. In the event that all of the fuel assemblies (121) or a majority are discharged from the reactor into the spent fuel pool, no more than 45 of these recently discharged assemblies shall be located in the small pool (pool no. 1).

C. If any of the specified conditions in 3.8.A or 3.8.B above are not met, refueling or fuel-handling operations shall cease. Work shall be initiated to correct the violated conditions so that the specifications are met, and no operations which may increase the reactivity of the core shall be performed.

* For the purpose of completing the fuel storage pool reracking modification, the movement and placement of loads are permitted as described in the licensee's submittals dated January 31, 1980 and January 14, March 10 and March 31, 1981.

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.[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 fuel is stored vertically in an array with the center-to-center distance between assemblies sufficient to assure $k_{eff} \leq 0.95$ even if unborated water were used to fill the pit. In addition, fuel in the storage pool shall have a U-235 loading of $\leq$ 39.0 grams of U-235 per axial centimeter of fuel assembly (average).

The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations or whenever there is fuel in the pit.

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.[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.

5.3 REACTOR

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide 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.

2. The average enrichment of the reload core is a nominal 2.90 weight percent of U-235. The highest Uranium-235 loading is a nominal 39 grams of U-235 per axial centimeter of fuel assembly (average).

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.[2]

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements.[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.[4] The system is intended to trip the reactor upon the abnormal dropping of more than one control rod.[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) FSAR, Section 3.2.3
(2) FSAR, Sections 3.2.1 and 3.2.3
(3) FSAR, Table 4.1-9
(4) FSAR, Section 7

D. Spent Fuel Storage Capacity

The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, would provide up to 1582 storage locations. With the four storage racks in the southeast corner of pool 1 removed a total of 1386 storage locations would be provided. The storage of spent fuel assemblies discharged as a result of normal refuelings is limited to a total of 1120 fuel assemblies in accordance with a stipulation agreed upon by the licensee (NSP), the NRC staff and the State of Minnesota. The stipulation, which is reproduced below, is hereby incorporated into these Technical Specifications.

> "Until such time as NRC issues to NSP a license amendment authorizing insertion and withdrawal of a spent fuel shipping cask into and from pool #1 when spent fuel is stored in that pool, NSP shall store in the spent fuel pools no more than 1120 spent fuel assemblies discharged as a result of normal refuelings. This limitation shall not apply to storage of any fuel which is to be returned to the reactor. NSP may store spent fuel in pool #1 so long as there are storage locations in pool #2 into which all spent fuel in pool #1 can be placed prior to insertion of a spent fuel shipping cask."

Reference

(1) FSAR, Section 9.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS.48 AND 42 TO

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

RELATING TO MODIFICATION OF THE SPENT FUEL POOL

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 January 31, 1980 the Northern States Power Company (NSP) proposed to change the spent fuel pool storage design for the Prairie Island Nuclear Generating Plant (PINGP) Unit Nos. 1 and 2. The presently approved design was reviewed and approved in Amendment Nos. 22 and 16 to Facility Operating License Nos. DPR-42 and DPR-60 issued August 16, 1977. The presently installed storage capacity is 687 fuel assemblies in a compartmented double pool which serves both units. The proposed modifications would make available up to 1582 storage locations of which only 1120 will be allowed to contain spent fuel assemblies. In response to our questions NSP submitted supplemental information by letters dated June 10, and November 21, 1980, January 14, February 3, March 10, March 31, and April 20, 1981.

2.0 Background

The Prairie Island Nuclear Generating Plant (PINGP) spent fuel pool (SFP) was originally designed with a storage capacity of about one and two-thirds cores (198 fuel assemblies) felt to be adequate for the storage of the discharge of the 40 assemblies per reactor year for a period of one year prior to shipment off-site for reprocessing.

By Amendment Nos. 22 and 16 dated August 16, 1977, we approved NSP's request to expand their SFP capacity to a total of 687 fuel assemblies for both units, through the use of high density spent fuel racks. NSP further realized that an additional increase in SFP capacity would likely be necessary before any reprocessing facility or offsite storage facility is ready. By letter dated January 31, 1980, NSP submitted their request to expand the SFP capacity to 1582 fuel assembly storage locations with high capacity poison racks. The licensee proposes to fill up to 1362 of these locations with spent fuel resulting from normal refueling. As discussed in Section 3.3 of this report, current approval is limited to a total storage of 1120 fuel assemblies due to the current status of resolution of the heavy loads handling issues for the PINGP.

Our reviews except for the heavy loads handling issue were based on the design proposed by the licensee.

NSP, as the licensee, is responsible for the modification to the SFP. Nuclear Services Corporation is retained to design the spent fuel racks, contract for fabrication, perform analysis pertinent to the modification, and provide technical assistance during installation.

3.0 Discussion and Evaluation

In reviewing the SFP modification, we considered: (1) criticality analysis, (2) spent fuel cooling, (3) installation of racks and fuel handling (4) structure design, (5) fuel and other heavy loads handling, (6) occupational radiation exposure, (7) radioactive waste treatment, and (8) material compatibility.

The proposed new higher density racks are to be made up of individual double-walled containers which are about fourteen feet long. The inner wall of each of these containers will be made from 0.090 inch thick sheet of 304 stainless steel which will be formed into a square cross section container with an inside dimension of 8.27 inches. The outer, or external wall will be a sheet of 0.036 inch thick stainless steel. Borated, neutron absorbing plates, which are 8.05 inches wide and 0.125 inches thick will be placed in each of the four spaces between the two walls. Thus each of the four sides of every container will have a borated plate in it which, as NSP states in its January 31, 1980 submittal, will initially contain at least 0.04 grams of boron-ten per square centimeter of plate. NSP also shows in this submittal that the average center-to-center pitch between fuel assembly storage tubes will be maintained at 9.50 $\pm$ 0.060 inches by the external sheets and by welded spacers.

3.1 Criticality Considerations

The Prairie Island fuel pool criticality calculations were performed under the assumption of 39 grams of U-235 per centimeter of assembly length, clean unborated water in the pool, fresh fuel without burnable poison, and an array of infinite extent in the lateral direction. Calculations were done for the standard Westinghouse 14x14 fuel assembly, the slightly different Exxon replacement design, and a proposed optimized (for uranium utilization) Westinghouse design. Since the latter two designs had smaller pellet diameters than the Westinghouse standard design, it follows that larger enrichments were used for them. Credit was taken for axial leakage but none was taken for rack structure (other than the storage cans) or for the fuel assembly spacers.

The calculations were performed for Northern States by Nuclear Services Corporation. The CHEETAH and XSDRN computer codes were used to generate cross sections for use in the CITATION diffusion theory program. The results of these calculations were verified against critical experiments which included boral blades, and against a Monte Carlo calculation which used the KENO-IV code with cross sections prepared by the AMPX code package.

The effect of "normal" variations - uncertainties in box dimensions and variations in pool temperature have been considered along with calculational biases and uncertainty in arriving at the total uncertainty in the calculation. The result for the optimized fuel case is a K-effective value of 0.942 with all uncertainties included.

3.1.1 Evaluation

A comparison of the calculation presented in this application with others made for similar configurations shows it to be acceptably accurate. This is confirmed by the results of the comparison of calculated and experimental results referred to above which showed that the calculation tended to overestimate the multiplication factor.

In order to assure that the appropriate amount of neutron poison is present a strict quality assurance procedure will be followed during the manufacture of the fuel storage boxes. In addition, after the racks are manufactured and are on-site a final verification of the presence of boron will be made by using a neutron source and detector combination. Each storage box will be examined. Continued presence of boron in the racks will be verified by removal of sample coupons periodically in order to test for deterioration.

3.1.2 Conclusion

We conclude that the design of the proposed spent fuel storage racks is acceptable for storage of fuel for the Prairie Island reactor with respect to potential criticality in normal usage and in credible accident configurations. Our conclusion is based on the following:

1. Standard state-of-the-art analysis methods are used for the analysis.

2. The methods were verified by comparison with critical experiment.

3. The result for these storage racks is consistent with that for similar rack designs calculated by other applicants using other methods.

4. Normal or expected variations in design parameters have been treated, either by assuming worst case values or by performing sensitivity studies.

5. Credible "abnormal" configurations have been analyzed.

6. Our criterion of less than or equal to 0.95 for the calculated effective multiplication factor has been met for both normal and abnormal storage configurations.

7. Tests will be conducted to assure the presence of the poison in the racks on the site.

Accordingly, we find that the proposed racks are safe with respect to criticality for the storage of fuel assemblies containing up to 39 grams of U-235 per cm. of assembly length.

3.2 Spent Fuel Cooling

The shared PINGP spent fuel storage facility consists of a small (pool #1) and a large (pool #2) storage pool housed within a seismic Category I structure. Transfer slots, with pneumatically sealed gates, permit the fuel to be moved from the fuel transfer canal into either pool or between pools. The elevation of the bottom of the transfer slots is above the top of the stored fuel.

Above the operating floor a seismic Category I concrete reinforced structure encloses the two spent fuel storage pools and the new fuel storage pit. The walls and roof of the enclosure serve as a tornado missile barrier. The passage of heavy loads in or out of the enclosure, using the overhead crane, is restricted to a slot in the roof and walls of the enclosure. The slots are located such that all loads being handled by the overhead crane would travel over pool #1, the smaller storage pool.

The new storage racks will be modular assemblies consisting of stainless steel storage tubes held and supported on a 9.5 pitch by upper and lower grids. The grids are a box like structure fabricated from heavy plates. A typical storage rack is shown in Figure 3.2-1 Exhibit-C of the January 31, 1980 submittal.

3.2.1 Evaluation

The licensed thermal power for each of Units 1 and 2 is 1,650 MWt. During a normal annual refueling cycle one third of a core's 121 fuel assemblies is discharged to the storage pools. The refueling cycles for the two units are scheduled such that a refueling operation takes place about every six months. The licensee's March 10, 1981 submittal provided the resultant decay heat loads. These heat loads have been calculated in accordance with Branch Position ASB 9-2. Further, in evaluating the adequacy of the cooling system it has been conservatively assumed that 100 hours will be required to prepare the facility for refueling and that either a normal core discharge or a full core discharge can be accomplished in 150 hours following shutdown, i.e., 50 hours are required for transferring fuel between the reactor vessel and the storage pools.

The spent fuel pool cooling system consists of two pumps and two heat exchangers. These are cross connected such that the loss of any one pump or heat exchanger will not prevent the operation of the remaining components. The decay heat is removed from the spent fuel pool heat exchangers by either Unit 1 or Unit 2's Component Cooling Water System. Each unit's Component Cooling Water System consists of two, 100% capacity, normally interconnected parallel loops each comprised of one pump and one heat exchanger having a rating of 29×10^6 BTU/HR. In the unlikely event that a LOCA should occur in the unit whose Component Cooling Water System is connected to the Spent Fuel Pool Cooling System, the operator would conservatively have more than one hour to transfer the pool cooling system to the unaffected unit's Components Cooling Water System.

The free volume of pool #2 is slightly less than 2.5 times the free volume of pool #1. The volume of water, storage racks and spent fuel in the respective pools are essentially in the same ratio. The water volumes in pool #1 and pool #2 are 12,692 ft^3 and 29,593 ft^3 respectively. Each pool has a high and low water level sensor which provides input to each control room's control board. The spent fuel pool liner seam leakage is directed to a common open sight drain trough for monitoring and then to the waste disposal system.

Temperature detectors are installed at both pools. A high temperature alarm for each pool, nominally set at 130°F, is located on each control room alarm panel. The auxiliary building operator, as a routine shift responsibility, monitors the spent fuel pool water level, temperature, radiation and the leak detection system. The control room operators, as part of their routine shift responsibility, monitor the spent fuel radiation levels.

In the course of reconfirming the cooling system's heat removal capability NSP found that the hydraulic flow resistance of the two heat exchangers was unequal. Therefore the flow distribution to the two heat exchangers was revised and the maximum pool water temperature was recalculated using the revised heat loads, and flows through the two heat exchangers. The assumed conditions and resultant maximum pool water temperatures are as follows:

(a) 1362 normally discharged fuel assemblies (11.97×10^6 BTU/HR peak heat load) are stored in pools 1 and 2. The pools are cooled by either the main heat exchanger and one of the two pumps or the backup heat exchanger and both of the pumps. The maximum pool water temperature will not exceed 137°F.

(b) 1362 normally discharged fuel assemblies plus a freshly off loaded core consisting of 121 fuel assemblies (25.09×10^6 BTU/HR peak heat load) are stored in pools 1 and 2. The pools are cooled by both the main and backup heat exchangers and both pumps. The maximum pool water temperature will not exceed 145°F.

(c) 1362 normally discharged fuel assemblies plus a freshly loaded core consisting of 121 fuel assemblies (25.09×10^6 BTU/HR peak heat load) are stored in pools 1 and 2. The analyses assumed the failure of either one pump or one heat exchanger. The maximum pool water temperature will not exceed 183°F.

Based on the above results we conclude the spent fuel pool cooling system is adequate and therefore acceptable.

NSP investigated the elapsed time before pool boiling under two sets of assumptions following the loss of all external pool cooling.

The two postulated conditions and results are as follows:

(a) Pools 1 and 2 contain 1362 normally discharged fuel assemblies plus a full core discharge, and it was assumed that complete mixing of the water in pools 1 and 2 would occur. Calculations indicate that boiling would occur in 8 hours. Further, since this is the maximum heat load the maximum boil off rate would be 51.4 gpm.

(b) Pool #1 contains one full core discharge plus 266 recently discharged fuel assemblies. It was further assumed that there would be no mixing of pool 1 water with pool 2 water. Calculations indicate that boiling would occur in 2.9 hours.

We concur that it is reasonable to assume that 8 hours is an adequate amount of time to perform minor maintenance on the cooling system and restore it to an operable condition and therefore the time established in the first postulated condition is acceptable. In order to avoid the short boil off time in the second postulated case, we believe that it would be prudent to distribute the 121 freshly off loaded core fuel assemblies in pools 1 and 2. By distributing these fuel assemblies in a ratio approximating the water volumes of pools 1 and 2 the time to boil would approach 8 hours. Further, by adopting this requirement the assumption made in the first case regarding complete mixing of water in pools 1 and 2 becomes less crucial in establishing the time to boiling. Therefore we will require that no more than 45 of the assemblies be placed in pool #1 and the remaining fuel assemblies be placed in pool #2.

Exhibit C of the January 31, 1980 submittal identified six available sources of makeup water in the unlikely event that all spent fuel pool cooling is lost and boiling occurs. Attachment 1 to the June 10, 1980 letter indicates the available makeup rate from each source and states that ten minutes or less is required to line up the valves or to carry out the steps necessary in order to make the water available. The six sources of makeup water and their makeup rates are as follows: (a) Chemical and Volume Control System - 300 gpm, (b) Chemical and Volume Control System Blender - 100 gpm, (c) Refueling Water Storage Tank - 80 gpm, (d) Reactor Makeup Storage Tanks - 80 pm, (e) four demineralized water hose stations, each station rated at 20 gpm, and (f) the fire protection system - there are two fire hose stations near the spent fuel pool each rated at 95 gpm.

Regarding the maximum required makeup rate of 51.4 gpm and the number of makeup sources and their respective makeup rates, we find the makeup sources to be adequate and therefore acceptable.

3.2.2 Conclusion

We have reviewed the calculated decay heat values and find them to be consistent with Branch Technical Position ASB 9-2 and therefore acceptable. The described spent fuel pool cooling system performance has been reviewed and found to be adequate and therefore acceptable. The available time required before the water will be available has been reviewed and found to be acceptable.

In regard to the time before boiling occurs, following a full core discharge, we find the time interval of 8 hours acceptable provided the stored fuel assemblies are distributed in both pools as specified above. This distribution of fuel assemblies has been addressed by Technical Specification 3.8.B.5.

3.3 Installation of the New Storage Racks and Load Handling

In order to change the storage racks within the reinforced concrete enclosure, a 15 ton temporary crane will be erected inside the enclosure. In addition, a temporary laydown area will be provided above pool #1 in order to permit the transfer of loads from one crane to the other one. The temporary laydown area will consist of covers placed above pool #1 in order to support the storage racks while the load transfer is being accomplished.

To accomplish the modification the following operations will be performed:

With all fuel assemblies (442 assemblies) stored in the larger pool (pool #2), the 15 ton temporary crane is erected over the smaller pool (pool #1) and placed on the fuel handling bridge rails. The temporary crane and 25 ton capacity overhead auxiliary hoist will be used in removing the existing empty storage racks in pool #1 and installing the new storage racks. All of the stored spent fuel will then be moved from the old storage racks in pool #2 and placed in the new storage racks in pool #1. The protective covers will be placed over pool #1 in order to protect the stored fuel and to provide a temporary laydown area. Using the 15 ton temporary crane, the existing storage racks in pool #2 will be individually lifted, transported and placed on the laydown area provided by the protective covers where the 25 ton overhead auxiliary hoist will pick them up and remove them for disposal. Similarly the new racks will be individually moved, by the 25 ton overhead hoist, into the enclosure and placed on the protective covers where the temporary crane will lift, transport, and locate them in pool #2. When all the new racks have been installed the 15 ton temporary crane will be disassembled and removed by the overhead hoist. The covers will be removed from pool #1 and the stored fuel will be moved to pool #2 using the fuel handling bridge. The weights of the old and new storage racks are equal to or less than 12.4 tons. The above procedures effectively limit the possibility of dropped load type damage to the stored fuel to the time that loads are being handled above pool #1 protective covers.

The 25 ton auxiliary hoist, which is mounted on the Auxiliary Building Crane, will be employed for the movement of loads outside the enclosure and within the enclosure slot. Outside of the enclosure, the travel paths of the loads are such that no equipment essential in the safe shutdown of the reactor is located beneath, adjacent to or otherwise within the area of influence of a dropped load. The FSAR indicates this crane has been designed, fabricated and qualified in accordance with the Electric Overhead Crane Institute (EOCI) Standard #61 and the American Standard Institute Standard B30.2-1967. Considering that the heaviest rack is less than one half of the 25 ton hoist's capacity, the safety margin for these operations will be twice what would exist when the hoist is handling its rated load of 25 tons.

While it is not anticipated to move loads, other than those associated with the modifications, through the enclosure and over pool #1, NSP has indicated that should that become necessary these loads will not be moved without first removing all fuel from storage pool #1.

The 15 ton temporary crane is a double leg gantry type unit having motorized drives for vertical and north/south motions. East/west motions will be powered manually.

Aside from one rack that will require slings, lifting rigs will be employed in handling the old and new storage racks. The lifting rigs will have four hooks to engage the four corners of the rack. The vertical dimensions of the rigging will be such that the carrying height of heavy loads above the temporary laydown area will not exceed 6 inches, i.e., the hook will be essentially at its upper limit of travel. All rigging will have an overall factor of safety of 10.

NSP states only trained and experienced NSP plant personnel will be permitted to operate the cranes during this modification.

Since neither of the above cranes are single-failure-proof cranes, the potential exists for dropping their load. A load drop onto the covers above pool #1, while the pool contains all 442 spent fuel assemblies, would potentially be the most severe accident. The adequacy of the covers to withstand load drops is discussed in section 3.4.

The Prairie Island Nuclear Generating Plant Units 1 and 2 Technical Specification 3.8.B.1 states "No heavy loads will be transported over or placed in either part of the spent fuel pool when irradiated fuel is stored in that part". This limitation makes it impossible to remove and insert the storage racks as described in this modification as well as utilizing pool #1 for the long term storage of spent fuel. As in the previous pool expansion program, NSP is requesting that a temporary waiver be granted to Technical Specification 3.8.B.1 during the load handling operations associated with the pool modification program. In this regard considering the limited number of such operations, the described precautionary measures and safety margins that will exist when carrying out these operations we find the request acceptable. In regards to the proposal contained in the present submittal to have spent fuel stored in pool No. 1 when heavy loads such as the four temporary storage racks or the spent fuel shipping cask are inserted or withdrawn from the pool, the following stipulation was agreed upon by NSP, the NRC staff and the State of Minnesota on September 23, 1980:

> "Until such time as NRC issues to NSP a license amendment authorizing insertion and withdrawal of a spent fuel shipping cask into and from pool #1 when spent fuel is stored in that pool, NSP shall store in the spent fuel pools no more than 1120 spent fuel assemblies discharged as a result of normal refuelings. This limitation shall not apply to storage of any fuel which is to be returned to the reactor. NSP may store spent fuel in pool #1 so long as there are storage locations in pool #2 into which all spent fuel in pool #1 can be placed prior to insertion of a spent fuel shipping cask".

3.3.1 Evaluation

The described sequence of steps by which the modifications are to be accomplished is such that to the extent possible the stored spent fuel assemblies will be removed from the areas where heavy load handling operations take place. To protect the stored spent fuel assemblies during the removal and installation of the storage racks protective covers will be placed over pool #1. These covers have been analyzed assuming the storage racks are dropped from a height of six inches. To preclude greater drop heights the vertical distance of the rigging will be so arranged that the respective hoists will be at their upper limit of travel when the bottom of the rack is six inches above the protective covers. The rated load capacity of both the 25 ton auxiliary hoist and the temporary 15 ton gantry exceeds the weights of both the old and new storage racks. The rigging will have a safety margin of 10. Further, only trained and experienced personnel familiar with the equipment will be permitted to operate the cranes during this modification.

In the past we have limited our reviews of spent fuel pool load drop accidents to loads equal to or greater than the weight of one fuel assembly and or its associated handling tool. Subsequently, it has become apparent that lighter loads, i.e., other normally handled loads, weighing less than one fuel assembly, and/or its associated handling tool, could potentially, if dropped, impact on stored spent fuel with a greater amount of kinetic energy and hence possibly cause greater damage to the stored spent fuel.

NSP's March 31, 1981 submittal elaborated on their March 10, 1981 response on this subject. It states that there are three normally used handling tools which, if dropped from their maximum drop height, could potentially possess more kinetic energy than one fuel assembly and its associated handling tool when dropped from its maximum carrying height above stored spent fuel. These tools are: the Burnable Poison Rod Assembly Handling Tool, the Spent Fuel Handling Tool, and the Thimble Plug Handling Tool.

To prevent the potential energy of these three long handled tools from exceeding 3,250 foot-pounds (i.e. the potential energy of one fuel assembly and its handling tool, 1,625 pounds, when dropped from its maximum carrying height, 2 feet, above the stored spent fuel) NSP will administratively limit the carrying height of the tools such that the products of their weight times the drop height will not exceed 3,250 foot-pounds. To assist the operator in accomplishing this, identification marks will be located on the shank of the tool at the pool's water level when the tool is at this elevation. In addition, only trained and experienced NSP personnel will handle the tools.

Both of the hoists employed in handling the tools (i.e. the 6,000 pound capacity spent fuel pool bridge crane hoist and the 25 ton capacity Auxiliary Building Crane hoist) have keepers on the hook in order to prevent the tool's lifting bail from inadvertently becoming disengaged from the hook. The bails on the handling tools (in the loaded condition) have a factor of safety ranging from 4.8 to 8.9.

We believe that the identification marks placed on the shank of the handling tools will assist the operator to administratively limit the carrying height of the tools above stored spent fuel. This, in conjunction with keepers on the hoist hooks and the reserve load capacity of the hoists, leads us to conclude that reasonable assurance has been provided to prevent a light load drop on stored spent fuel which would exceed the damage potential of a dropped fuel assembly and is, therefore, acceptable.

3.3.2 Conclusion

In conjunction with our findings set forth in Section 3.4 of this report regarding the structural adequacy of the pool #1 cover, we conclude that the request for a temporary exemption to Technical Specification 3.8.B.1 using the described handling equipment and handling procedures is acceptable in order to complete the proposed spent fuel pool modifications.

The above procedures and conditions will reduce the possibility of dropping a rack or a fuel cask onto stored fuel assemblies to an acceptable level. From this and the protection provided by the protective covers, we conclude that the health and safety of the public will not be endangered by reracking the spent fuel pools and is therefore acceptable.

The information supplied regarding the handling of light loads above stored spent fuel includes a description of the identification marks on the shank of the handling tools that will assist in limiting the carrying height of tools above the stored spent fuel. This, in conjunction with keepers on the hoist hooks and the reserve load capacity of the hoists, leads us to conclude that reasonable assurance has been provided to prevent a light load drop on stored spent fuel which could exceed the damage potential of a dropped fuel assembly and is therefore acceptable.

3.4 Structural Design

The design for the racks, fabrication, and installation criteria; the structural design and analysis procedures for all loading, including seismic and impact loadings; the load combinations; the structural acceptance criteria; the quality assurance requirements for design, and applicable industry codes were all reviewed in accordance with the applicable portions of the current "OT Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications", dated April 1978, including revisions, dated January 1979.

The design of the spent fuel storage modules utilized the AISC Code "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings". The basic material allowables were taken from the ASME Code, Section III, Division 1 and Standard Review Plan Section 3.8.4 for the applicable load combinations. The fabrication and installation of the modules are in accordance with the ASME Code Section III, Division 1, Subsection NF with the following exceptions: (1) material traceability is preserved for each module and not for each individual piece of the module, (2) the neutron absorber material was purchased in accordance with ANSI documents since the ASME code does not address this material, and (3) the welds used to install the Boraflex into the fuel racks were made by the tungsten inert gas welding process. These welds are not structural welds and a random failure of 50% of these welds would not result in dislocation of the Boraflex.

3.4.1 Evaluation

The seismic analysis of the racks utilized a time history analysis including structural damping consistent with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants" and Regulatory Guide 1.92.

The structural evaluation of the proposed racks was based on the results from previous seismic analyses contained in the report entitled "Revised Earthquake Analysis for Prairie Island Nuclear Generating Plant" by John A. Blume and Associate Engineers, February 16, 1971. The spectrum corresponding to the spent fuel pool floor was used in the analysis. Although the racks have no floor attachment nor lateral supports, it was assumed in the analysis that the coefficient of friction was such that the racks would not slide thus maximizing the internal stresses produced by seismic forces.

The 7X8 rack was used in the analysis to determine loads, stresses and deflections, since this rack had the greatest potential for tipping and would develop the greatest internal forces due to seismic and deadweight loadings. Two models were used in the analysis of the fuel rack. A three dimensional finite element model was used to determine stresses in the rack resulting from seismic, thermal, grapple, buoyancy and dead weight and a one-dimensional model was used to determine maximum rack sliding distance during an SSE. All the stress analyses on the finite element model were performed using the computer program STARDYNE and the stresses resulting from dead weight and buoyancy loads were evaluated simultaneously.

A nonlinear sliding analysis was performed to determine the maximum displacement and velocity of the rack relative to the pool floor under the action of SSE vibratory motion. The coefficient of friction between the stainless steel liner and the rack leveling legs used in the analysis was conservatively chosen to be 0.2, based on the information provided in a report by E. Rabinowicz of the Massachusetts Institute of Technology entitled "Friction Coefficients of Water Lubrication Stainless Steel for a Spent Fuel Rack Facility" dated November 5, 1976. The result of this analysis indicates that during an SSE, the proposed racks which are free-standing may slide towards each other and impact in a random fashion. The methodology used by the licensee to predict the stresses generated by the impact has been found to be acceptable.

The postulated fuel assembly drop was considered in the analysis of the racks. The energy balance method was used to determine the effects of the impact of a fuel assembly dropped from a maximum height of 18 inches at the most critical location on the rack. Three postulated drops were analyzed, namely, the straight drop of a fuel assembly through an individual cell, an inclined drop on the rack and a vertical impact on the rack where a point load was assumed instead of finite impact area. In all cases, the impact energy is dissipated by local yielding or crushing, however gross stresses in the rack remain below allowable and the overall structural integrity is maintained.

The effects of a postulated stuck fuel assembly due to the attempted assembly withdrawal was considered and the stresses due to this postulated accident were computed by an elastic analysis.

In order to protect fuel assemblies in the small spent fuel pool against the accidental drop of a heavy load, a protective cover over the pool is provided. The pool cover is made of 3/16 inch stainless steel plate welded to a grid of structural tees and built-up wide-flange beams which are made of structural steel ASTM A588 Grade A. Underneath each end of the beams, one pad made of one-inch thick compressible material is used between the cover and the concrete floor.

The licensee had evaluated the protective cover when subjected to a postulated drop of 24,800 pounds at a height of 6 inch clearance above the cover. The results of the evaluation show that although local plastic deformation may occur, the overall structural integrity of the cover will be maintained. Thus, the effect of the postulated drop of this heavy load is considered to be within the acceptable limit.

The loads and load combinations considered in the analysis of the spent fuel storage racks are in accordance with SRP Section 3.8.4. Results of the analysis show that the racks are capable of withstanding the loads associated with all the design loading conditions without exceeding allowable stresses.

The spent fuel pool is constructed of concrete walls and floor, lined with a stainless steel liner and reinforced in both vertical and horizontal directions. The fuel pool concrete, reinforcing steel and liner were analyzed to account for the additional loadings imposed by the new racks. The structural adequacy was verified using conventional concrete building codes (ACI 318).

Results of the analysis for the most severe loading conditions indicate that the maximum stresses are within the allowables, and that the structural members of the fuel pool are adequate to withstand the additional loads imposed by the new racks and additional fuel.

3.4.2 Conclusion

The analysis, design, fabrication, and criteria for establishing installation procedures of the proposed new spent fuel racks are in conformance with accepted codes, standards and criteria. The structural design and analysis procedures for all loadings, including seismic, thermal, and impact loading; the acceptance criteria for the appropriate loading conditions and combinations; and the applicable industry codes are in accordance with appropriate sections of the NRC Staff "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

Allowable stress limits for the combined loading conditions are in accordance with the AISC specifications. Yield stress values at the appropriate temperatures were obtained from Section III of the ASME B&PV Code. The quality assurance codes and criteria for the materials, fabrication and installation of the new racks are in accordance with the accepted requirements of the ASME Code.

The effects of the additional loads on the existing pool structure due to the new fuel racks, existing fuel racks, and equipment have been examined. The pool structural integrity is assured by conformance with Standard Review Plan Section 3.8.4.

Results of the seismic structural analyses indicate that the racks are capable of withstanding the loads associated with all design loading conditions. Also, impact due to fuel assembly/cell interaction has been considered, and will result in no damage to the racks or fuel assemblies. The methodology used by the licensee to predict the stresses generated by the impact during the SSE has been found acceptable.

Results of the dropped fuel assembly analyses show that local rack deformation will occur, but indicate that gross stresses meet the applicable allowables and that the integrity of the racks is maintained.

Results of the dropped heavy loads over the protective pool cover indicate that although local damage and plastic deformation may occur, the overall structural integrity of the cover is maintained and is within the acceptable limits.

Results of the stuck fuel assembly analyses show that the stresses are below those allowed for the applicable loading combinations.

We find that the subject modification proposed by the licensee is acceptable and satisfies the applicable requirements of the General Design Criteria 2,4,61, and 62 of 10 CFR, Part 50, Appendix A.

3.5 Occupational Radiation Exposure

We have reviewed the licensee's plan for the removal and disposal of the existing racks that were installed during a previous modification in 1977/1978 and the installation of the new racks with respect to occupational radiation exposure. The occupational exposure for this operation is conservatively estimated by the licensee to be about 40 man-rems. We consider this to be a reasonable estimate because it is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job is being performed.

The modification will be performed by reracking pool 1, transferring spent fuel elements from pool 2 into pool 1 and then reracking pool 2. The spent fuel will then be returned to pool 2. The licensee has indicated that alternative plans are being evaluated for the disposal of the present racks which include removing and crating the racks intact for shipment offsite versus removing, cleaning by electropolishing and subsequently disposing of the racks. The licensee has estimated that either alternative will result in an occupational exposure of four man-rems or less. This contribution to occupational exposure has been included in the over all estimate of 40 man-rems discussed above. Selection of a disposal method has not been finalized. The licensee will estimate the exposures associated with the different ways to dispose of the present racks from measurements of the activity levels on them when they are removed from the pool and are ready for disposal. At that time taking into account alternative disposal costs and exposures, the licensee will select the method of disposal so that exposures will be kept to levels that are as low as is reasonably achievable. All work will be effected in accordance with a radiation permit to identify all protection requirements. Health physics personnel will be available to assure that ALARA radiation exposures prevail.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel area from radionuclide concentrations in the SFP water and deposited on the SFP walls. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible impact. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in additional exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

3.6 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation dated September 1972. There will be no change in the waste treatment system or in the conclusions given in Section 11.0 of the evaluation of these systems because of the proposed modification. Our evaluation of the SFP cleanup system, in light of the proposed modification, has concluded that any resultant additional burden on the system is minimal and therefore the existing SFP cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water within acceptably low levels.

Our evaluation of the radiological considerations supports the conclusion that the proposed modification to the Prairie Island 1 and 2 spent fuel pools is acceptable because:

(1) The conclusions of the evaluation of the waste treatment systems, as found in the Prairie Island Safety Evaluation Report of 1972, are unchanged by the modification of the SFP.

(2) The existing SFP cleanup system is adequate for the proposed modification.

3.7 Material

The proposed spent fuel storage racks are fabricated of Type 304 stainless steel with the exception of the adjusting bolts of the rack feet. These bolts are made from Type 17-4 PH stainless steel. The existing spent fuel pool liner is stainless steel.

The high density spent fuel storage racks will utilize Boraflex sheets as a neutron absorber, within an annulus formed by concentric inner and outer, square stainless steel tubes. The Boraflex is composed of boron carbide powder in a rubber-like silicone polymeric matrix. The Boraflex sheets will have a minimum boron-ten content of 0.04 gm/cm2 of sheet surface area. The spent fuel storage rack is composed of individual storage cells interconnected to form an integral grid structure. The annulus region between the concentric tubes that contains the Boraflex is vented at both the top and bottom. The Boraflex is not attached to either of the stainless steel tubes. It is captured in the annulus and is supported on a stainless steel strip at the bottom of the annulus.

The pool contains oxygen-saturated demineralized water containing boric acid, generally controlled to a temperature below 130°F.

3.7.1 Evaluation

The pool liner, rack lattice structure and fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.00×10^{-5} inch in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. The Boraflex is composed of non-metallic materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material. The evaluation tests have shown that the Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex to 1.03×10^{11} rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boraflex. Long term borated water soak tests at high temperatures were also conducted. The test showed that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The annulus space which contains the Boraflex is vented to the pool at each corner storage tube assembly. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the inner stainless steel tube.

The tests have shown that neither irradiation, environment nor Boraflex composition has a discernible effect on the neutron transmission of the Boraflex material. The tests also show that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally in the Boraflex will typically contain 0.1 wt percent of soluble boron. The test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble specie from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell's walls. These specimens will be removed and examined periodically.

3.7.2 Conclusion

From our evaluation as discussed above, we conclude that the corrosion that will occur in the Prairie Island spent fuel storage pool environment should be of little significance during the 40-year life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life of 40 years.

We further conclude that the environmental compatibility and stability of the materials used in the Prairie Island expanded spent fuel storage pool is adequate, based on the test data cited above and actual service experience in operating reactors.

We have reviewed the surveillance program and we conclude that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. We therefore find that the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison.

4.0 Technical Specification

As indicated in the criticality analysis of this Safety Evaluation and in the licensee's referenced submittals the maximum Uranium-235 content is specified in Technical Specification 5.3 and 5.6 to be 39 grams per axial centimeter of fuel assembly. Therefore fuel assemblies that are bound by the fuel assembly designs described in the licensee's referenced submittals may be stored in the spent fuel pool.

Technical Specification 3.8.B.1 is modified to permit its suspension during the fuel pool reracking operation so that the old racks may be removed and the new racks installed.

Technical Specification 3.8.B.5 is added to implement the specification associated with a discharge of a complete core of fuel assemblies as discussed in Section 3.2.1 of this report.

Technical Specification 5.6.D has been added to implement the stipulation on allowable spent fuel storage discussed in Section 3.3 of this report.

5.0 Safety 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 be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 13, 1981

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 48 AND 42 TO LICENSES NOS. DPR-42 AND DPR-60

RELATING TO MODIFICATION OF THE SPENT FUEL POOL

NORTHERN STATES POWER COMPANY

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

DOCKET NOS. 50-282 AND 50-306

1.0 DESCRIPTION OF PROPOSED ACTION

By letters dated January 31, 1980 as supplemented by letters dated June 19, 1980, November 21, 1980, January 14, 1981, February 3, 1981, March 10, 1981 and March 31 and April 20, 1981, the Northern States Power Company (NSP) proposed to change the spent fuel pool (SFP) storage design for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2 (PINGP) from the design which was reviewed and approved in Amendment Nos. 22 and 16 to Facility Operating License Nos. DPR-42 and DPR-60 issued August 16, 1977. This approved spent fuel storage capacity is 687 fuel assemblies.

The modification evaluated in this environmental impact appraisal is the proposal by the licensee to replace the existing spent fuel storage racks with high density borated storage racks. Several numbers are used herein which bear explanation. If the SFP were totally filled with fuel assembly racks this would provide 1582 storage spaces. However, the licensee would not store spent fuel assemblies from normal refueling operations in all of these spaces since that would leave no room for the eventual placement of a spent fuel shipping cask into the pool to remove the spent fuel assemblies from the pool. Therefore not more that 1386 assemblies resulting from normal refueling operations are proposed to be stored in the SFP by the licensee. This value has been further limited to 1120 assemblies pending further evaluation and resolution of the heavy loads handling issue in the future.

Therefore the staff's evaluation and appraisal is being performed for a total of 1582/1386 as proposed by the licensee with the only exception being that the facility Technical Specifications and license will limit the number of assemblies resulting from normal refueling operations to 1120.

2.0 NEED FOR INCREASED STORAGE CAPACITY

The PINGP SFP was originally designed with the storage capacity of 198 fuel assemblies (1-2/3 cores). The first refueling of PINGP Unit 1 was on March 4, 1976 at which time 40 fuel assemblies were replaced and stored in the SFP. The first 40 spent fuel assemblies from Unit 2 were placed in the SFP in October 1976. At this rate, 80 assemblies per year from both units are discharged from the reactor to the SFP.

By letter dated August 16, 1977, we approved NSP's request to expand their SFP capacity to 687 fuel assemblies which would extend the storage capability of the pool midway through 1983 and leave room for a complete core discharge.

Spent fuel is not currently being processed on a commercial basis in the United States and storage capacity away from reactor sites is available only on an emergency basis as is discussed in Sections 6.1 and 6.2 of this appraisal.

Based on the above information, there is clearly a need for additional onsite spent fuel storage capacity to assure continued operation of the PINGP units, with full core off-load capability, after the Spring of 1983. The expansion of the SFP capacity to 1120 assemblies would provide this capability through the Fall of 1989 using annual refueling cycles.

3.0 THE FACILITY

The PINGP units are described in the Final Environmental Statement (FES), issued by the Commission in May 1973, related to the section on operation of the facilities. Each unit is a Pressurized Water Reactor (PWR) which produces 1650 megawatts thermal (MWt) and has a gross electrical output of 530 megawatts (MWe). Pertinent descriptions of principal features of the plant as it currently exists are summarized below to aid the reader in following the evaluations in subsequent sections of this appraisal.

3.1 Fuel Inventory

Each PINGP reactor contains 121 fuel assemblies. The fuel assemblies are a cluster of 179 fuel rods or sealed tubes arranged in a 14 by 14 array. The weight of the fuel, as UO_2, is approximately 120,000 pounds. About one-third of the assemblies are removed from the reactor and replaced with new fuel each year. Present scheduling is for the refueling outage to be in the first few months of each year for Unit No. 2 and the last few months of each year for Unit No. 1.

The proposed modification of the SFP would not change the quantity of uranium fuel used in the reactor over the anticipated operating life of the facility and would not change the rate at which spent fuel is generated by the facility. The added storage capacity would increase the number of spent fuel assemblies that could be stored in the SFP and the length of time that some of the fuel assemblies could be stored in the pool.

3.2 Purpose of the SFP

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core and they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

3.3 SFP Cooling System

The spent fuel pool cooling system consists of two pumps and two heat exchangers. These are cross connected such that the loss of any one pump or heat exchanger will not prevent the operation of the remaining components. The decay heat is removed from the spent fuel pool heat exchangers by either Unit 1 or Unit 2's Component Cooling Water System. Each unit's Component Cooling Water System consists of two, 100% capacity, normally interconnected parallel loops each comprised of one pump and one heat exchanger having a rating of 29 X 10^6 BTU/HR. In the unlikely event that a LOCA should occur in the unit whose Component Cooling Water System is connected to the Spent Fuel Pool Cooling System, the operator would conservatively have more than one hour to transfer the pool cooling system to the unaffected unit's Component Cooling Water System.

In the course of reconfirming the cooling system's heat removal capability NSP found that the hydraulic flow resistance of the two heat exchangers was unequal. Therefore the flow distribution to the two heat exchangers was revised and the maximum pool water temperature was recalculated using the revised heat loads and flows through the two heat exchangers. The assumed conditions and resultant maximum pool water temperatures are as follows:

(a) 1362 normally discharged fuel assemblies (11.9 X 10^6 BTU/HR peak heat load) are stored in pools 1 and 2. The pools are cooled by either the main heat exchanger and one of the two pumps or the backup heat exchanger and both of the pumps. The maximum pool water temperature will not exceed 137°F.

(b) 1362 normally discharged fuel assemblies plus a freshly off loaded core consisting of 121 fuel assemblies (25.09 X 10^6 BTU/HR peak heat load) are stored in pools 1 and 2. The pools are cooled by both the main and backup heat exchangers and both pumps. The maximum pool water temperature will not exceed 145°F.

(c) 1362 normally discharged fuel assemblies plus a freshly loaded core consisting of 121 fuel assemblies (25.09 X 10^6 BTU/HR peak heat load) are stored in pools 1 and 2. The analysis assumes the failure of either one pump or one heat exchanger. The maximum pool water temperature will not exceed 183°F.

Based on the above results our Safety Evaluation finds the spent fuel pool cooling system is adequate and therefore acceptable.

3.4 SFP Purification System

The SFP purification loop consists of filters, a mixed bed demineralizer and the required piping, valves and instrumentation. The SFP cooling system pumps draw water from the pool or the refueling cavity. A fraction of this flow is passed through the SFP purification loop. The water is returned to the pool or the refueling cavity.

Because we expect only a small increase in the radioactivity released to the pool water as a result of the proposed modification as discussed in Section 4.4 of this environmental impact appraisal, we conclude the SFP filtering system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels which have existed prior to the modification.

3.5 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems are evaluated in the FES dated May 1973. There will be no change in the waste treatment systems described in Section III.D.2 of the FES because of the proposed modification.

4.0 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

4.1 Land Use

The external dimensions of the SFP will not change because of the proposed expansion of its storage capacity; therefore, no additional commitment of land is required. The SFP is intended to store spent fuel assemblies under water for a period of time to allow shorter-lived radioactive isotopes to decay and to reduce their thermal heat output. This type of use will remain unchanged by the modification but the additional storage capacity would provide for an additional sixteen normal refuelings. Thus, the proposed modification would result in more efficient use of the land already designed for spent fuel storage.

4.2 Water Use

There will be no significant change in plant water consumption or use as a result of the proposed modifications. As discussed subsequently, storing additional spent fuel in the SFP will slightly increase the heat load on the SFP cooling system. This heat is transferred in turn to the component cooling water system and to the service water system. The modifications will not change the flow rate within these cooling systems. The temperature of the SFP water during normal refueling operations with only one SFP cooling pump running is expected to remain below 137°F, as compared to the 120°F used as the design basis in the FSAR. Therefore, evaporation and thus the need for makeup water will not be significantly changed by the proposed modifications.

4.3 Nonradiological Effluents

There will be no change in the chemical or biocidal effluents from the plant as a result of the proposed modification.

The only potential offsite nonradiological environmental impact that could arise from this proposed action would be additional discharge of heat to the atmosphere and to the Mississippi River. Storing spent fuel in the SFP for a longer period of time will add more heat to the SFP water. The SFP heat exchangers are cooled by the component cooling water system which in turn is cooled by the plant cooling water system. The maximum incremental heat load resulting from the SFP modification is 8.09×10^6 BTU/HR. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools. Compared with the existing heat load (8.4×10^9 BTU/HR) on the plant cooling tower water system, this small additional heat load from the SFP cooling system will be negligible.

4.4 Radiological Impacts

4.4.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is fuel which has decayed at least eleven years based on a pre-expansion capacity of 687 storage locations. During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90 are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

4.4.2 Effect of Fuel Failure on the SFP

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel pool water appears to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months.

Based on the operational reports submitted by the licensee and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

Experience indicates that there is little radionuclide leakage from Zircaloy-clad spent fuel stored in pools for over a decade. Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircaloy-clad fuel elements which developed defects during reactor exposure, e.g., Ginna, Oyster Creek, Nine Mile Point, and Dresden Units Nos. 1 and 2. Based on the operational reports submitted by licensees and discussions with the operators, there has not been any significant leakage of fission products from spent reactor fuel stored in the MO pool or the NFS pool. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the MO pool without need for isolation in special cans. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity.

A Battelle Northwest Laboratory (BNL) report, "Behavior of Spent Nuclear Fuel in Water Pool Storage" (BNWL-2256 dated September 1977), states that radioactivity concentrations may approach a value up to 0.5 μCi/ml during fuel discharge in the SFP. After the refueling, the SFP ion exchange and filtration units will reduce and maintain the pool water in the range of 10^{-3} to 10^{-4} μCi/ml.

In handling defective fuel, the BNL study found that the vast majority of failed fuel does not require special handling and is stored in the same manner as intact fuel. Two aspects of the defective fuel account for its favorable storage characteristics. First, when a fuel rod perforates in-reactor, the radioactive gas inventory is released to the reactor primary coolant. Therefore, upon discharge, little additional gas release occurs. Only if the failure occurs by mechanical damage in the basin are radioactive gases released in detectable amounts, and this type of damage is extremely rare. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels. The second favorable aspect is the inert character of the uranium oxide pellets in contact with water. This has been determined in laboratory studies and also by casual observations of pellet behavior when broken rods are stored in pools.

4.4.3 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defective fuel. However, we have conservatively estimated that an additional 33 curies per year of Krypton-85 may be released from both units when the modified pools are completely filled. This increase would result in an additional total body dose to an individual at the site boundary of less than .00004 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than .0002 person-rem/year. This is significantly less than the natural fluctuations in the dose this population would receive from natural background radiation. Under our conservative assumptions, these exposures represent an increase of less than 0.1% of the exposures from the plant evaluated in the FES (1973) for the individual at the site boundary (Table V-2) and the population (Table V-3). Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

4.4.4 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations while reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, should be minor because of the capability of the cleanup system to remove radioactivity to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we generally agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two resin beds a year due to the increased operation of the spent fuel pool cleanup system. The annual average volume of solid waste shipped from the Prairie Island Plant during 1974 through 1979 was 7600 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 40 cubic feet per year, the increase in total waste volume shipped would be less than 1% and would not have any significant additional environmental impact.

The licensee indicates that alternative plans are being evaluated for the disposal of the present racks. The alternatives include removing and crating the racks for shipment offsite for disposal as low level solid waste (a volume of about 15,000 cubic feet) versus removing, cleaning by electropolishing and subsequent disposal. Selection of a disposal method has not been finalized. Averaged over the lifetime of the plant the disposal of the racks intact as low level waste would increase the total waste volume shipped from the facility by less than seven percent. This will not have a significant additional environmental impact.

4.4.5 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. Since the SFP cooling and cleanup system operates as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of this modification. The SFP demineralizer resin removes soluble radioactive matter from the SFP water. These resins are periodically flushed with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin might increase slightly due to the additional spent fuel in the pool, but the soluble radioactivity should be retained on the resins. If any activity is transferred from the spent resin to the flush water, it will be removed by the liquid radwaste system since the sluice water is returned to the liquid radwaste system for processing. After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

4.4.6 Occupational Exposure

We have reviewed the licensee's plans for the removal and disposal of the existing racks that were installed during a previous modification in 1977/78 and the installation of the new racks with respect to occupational radiation exposure. The occupational exposure for the entire operation is estimated by the licensee to be about 40 man-rems. We consider this to be a reasonable estimate because it is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. Performance of this operation is expected to be a small fraction of the total man-rem burden from occupational exposure at the plant.

We have estimated the increment in onsite occupational doses resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel pool area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to the dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed actions represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modifications should add less than one percent to the total annual occupational radiation exposure burden at both units. Thus, we conclude that storing additional fuel in the two pools will not result in any significant increase in doses received by occupational workers.

4.4.7 Impacts of Other Pool Modifications

As discussed above, the additional environmental radiological impacts in the vicinity of PINGP-1&2 resulting from the proposed modification are very small fractions (less than 1%) of the impacts evaluated in the PINGP-1&2 FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that an SFP modification at any other facility should not significantly contribute to the environmental impact of PINGP-1&2 and that the PINGP-1&2 SFP modification should not contribute significantly to the environmental impact of any other facility.

4.4.8 Impacts on the Community

The new storage racks were fabricated offsite and shipped to the PINGP, where they are stored. Only a few truck or rail shipments would be involved in shipment of these racks and disposal of the present ones. The impacts of dismantling the present racks and installing the new ones will be limited to those normally associated with metal working activities. No significant impact on the community is expected to result from the fuel rack conversion or subsequent operation with increased storage of spent fuel in the SFP.

4.5 Evaluation of Radiological Impact

As discussed above, the proposed modification does not significantly change the radiological impact evaluated in the FES.

5.0 ENVIRONMENTAL IMPACT OF POSTULATED ACCIDENTS

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the FES for PINGP dated May 1973.

Additionally, the NRC staff has under way a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because the PINGP has the TS requirement to prohibit the movement of heavy loads over the fuel assemblies in the SFP, we have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is under way.

6.0 ALTERNATIVES

The staff has considered the following alternatives to the proposed expansion of the SFP storage capacity at PINGP-1&2: (1) reprocessing the spent fuel; (2) shipment of spent fuel to a separate fuel storage facility; (3) shipment of spent fuel to another reactor site; (4) reduced plant operation; and (5) shutdown of facility. These alternatives are discussed below.

6.1 Reprocessing of Spent Fuel

As discussed earlier, none of the three commercial reprocessing facilities in the United States is currently operating. The General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois is in a decommissioned condition. Nuclear Fuel Services informed the NRC on September 22, 1976, that it was "withdrawing from the nuclear fuel reprocessing business". The Allied-General Nuclear Services (AGNS) reprocessing plant at Barnwell, South Carolina, received a construction permit on December 18, 1970. In October 1973, AGNS applied for an operating license for the reprocessing facility; construction of the reprocessing facility is essentially complete but no operating license has been granted. On July 3, 1974, AGNS applied for a materials license to receive and store up to 400 MTU of spent fuel in the onsite storage pool, on which construction has also been completed but hearings with respect to this application have not been held and no license has been granted.

In 1976, Exxon Nuclear Company, Inc. submitted an application for a proposed Nuclear Fuel Recovery and Recycling Center (NFRRC) to be located at Oak Ridge, Tennessee. The plant would include a storage pool that could store up to 7,000 MTU in spent fuel. However, licensing review of this application was discontinued in 1977 as discussed below.

On April 7, 1977, the President issued a statement outlining his policy on continued development of nuclear energy in the U. S. The President stated that: "We will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U. S. nuclear power programs. From our own experience, we have concluded that a viable and economic nuclear power program can be sustained without such reprocessing and recycling".

On December 23, 1977, the NRC terminated the fuel cycle licensing actions involving mixed oxide fuel (GESMO) (Docket No. RM-50-5), the AGNS' Barnwell Nuclear Fuel Plant Separation Facility, Uranium Hexafluoride Facility and Plutonium Product Facility (Dockets Nos. 50-332, 70-1327 and 70-1821), the Exxon Nuclear Company, Inc. NFRRC (Docket No. 70-564), Westinghouse Electric Corporation (recycle fuels plant, Docket No. 70-1432) and the NFS West Valley Reprocessing Plant (Docket No. 50-201). The Commission also announced that it would not at this time consider any other applications for commercial facilities for reprocessing spent fuel, fabricating mixed-oxide fuel, and related functions. At this time, any consideration of these or comparable facilities has been deferred for the indefinite future. Reprocessing is not a reasonable alternative to the proposed expansion of the PINGP SFP. Accordingly, no estimate of cost is considered appropriate.

6.2 Independent Spent Fuel Storage Faciltiy

An alternative to expansion of onsite SFP storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1,000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. The fuel storage pools at MO and NFS are functioning as smaller ISFSIs although this was not the original design intent. The license for the General Electric (GE) facility was amended on December 3, 1975 to increase the storage capacity to about 750 MTU; and, as of August 30, 1978, 310 MTU was stored in the pool in the form of 1196 spent fuel assemblies.

An application for an 1100 MTU capacity addition is pending and the schedule called for completion in 1980 if approved. However, by a motion dated November 8, 1977, GE requested the Atomic Safety and Licensing Board to suspend indefinitely further proceedings on this application. This motion was granted.

With regard to the status of storage space at MO, we have been informed that GE is primarily operating the MO facility to store either fuel owned by GE (which had been leased to utilities on an energy basis), or fuel which GE has previously contracted to reprocess. We were also informed that the present GE policy is not to accept spent fuel for storage except fuel for which GE has a previous commitment. There is no such commitment for PINGP spent fuel. Storage of the PINGP spent fuel at the existing reprocessing facilities is not a viable alternative to the expansion of the PINGP spent fuel pools.

The NFS facility has capacity for about 260 MTU, with approximately 170 MTU presently stored in the pool at West Valley. Although the storage pool is not full, NFS has indicated that it is not accepting additional spent fuel, even from the reactor facilities with which it had reprocessing contracts.

We also considered under this alternative the construction of new ISFSIs. The staff had estimated that at least five years would be required for completion of an ISFSI. This estimate assumes one year for preliminary design; one year for preparation of the license application, environmental report, and licensing review in parallel with one year for detail design; two and one-half years for construction and receipt of an operating license; and one-half year for plant and equipment testing and startup.

Industry proposals for additional independent spent fuel storage facilities are scarce to date. In late 1974, E. R. Johnson Associates, Inc. and Merrill Lynch, Pierce, Fenner and Smith, Inc. issued a series of joint proposals to a number of electric utility companies having nuclear plants in operation or contemplated for operation, offering to provide independent storage services for spent nuclear fuel. A paper on this proposed project was presented at the American Nuclear Society meeting in November 1975 (ANS Transactions, 1975 Winter Meeting, Vol. 22, TANSAO 22-1-836, 1975). In 1974, E. R. Johnson Associates estimated the construction cost would be equivalent to approximately $9,000 per spent fuel assembly.

Several licensees have evaluated construction of an ISFSI and have provided cost estimates. In 1975, Connecticut Yankee, for example, estimated that an independent facility with a storage capacity of 1,000 MTU (BWR and/or PWR assemblies) would cost approximately $54 million and take about five years to put into operation. The Commonwealth Edison Company estimated the construction cost of an ISFSI in 1975 at about $10,000 per fuel assembly. To this would be added the costs for maintenance, operation, safeguards, security, interest on investment, overhead, transportation and other costs.

On December 2, 1976, Stone and Webster Engineering Corporation submitted a Topical Report requesting NRC approval for a standard design ISFSI intended for siting near nuclear power facilities. Based on discussions with Stone & Webster, we estimated that the present day cost for such a fuel storage installation would be about $24 million, exclusive of site preparation costs. On July 12, 1978 we concluded that the proposed approach and conceptual design are acceptable.

Base on the above facts, on a short-term basis (i.e., prior to 1986), an ISFSI is not available as an alternative. One would not be available in time to meet the licensee's needs. It is also unlikely that the environmental impacts of this alternative, on a delayed availability basis, would be less than the minor impacts associated with the proposed PINGP modification. This is based on the fact that offsite transportation would be involved and a structure, pool, and supporting systems would have to be erected and installed for an ISFSI, whereas for PINGP modification, only new storage racks are involved.

For the long term, DOE is modifying its program for nuclear waste management to include design and evaluation of a long term repository to provide Government storage of unreprocessed spent fuel rods in a retrievable condition. It is estimated that the long term storage facility will start accepting commercial spent fuel in the time frame of 1995 to 2000. The criterion for acceptance is that the spent fuel must have decayed a minimum of ten years so it can be stored in dry condition without need for forced air circulation.

DOE has recently revised its policy with respect to the provision by DOE of interim fuel storage facilities. By letter dated March 27, 1981, addressed to the Presiding Officer for the ongoing Waste Confidence Rulemaking proceeding, DOE indicated that it had reached a decision to discontinue its efforts to provide Federal government - owned or controlled away-from-reactor storage facilities. DOE intends to redirect its effort to support the development of alternative means to be employed by utilities to further increase spent fuel storage capabilities. This leaves the task of developing interim storage capacity to private industry. We conclude that Government - sponsored interim storage is not a viable alternative to the proposed SFP modification.

6.3 Storage at Another Reactor Site

NSP owns and operates the Monticello Nuclear Plant. The Monticello facility is a boiling water reactor whereas the PINGPs are pressurized water reactors. The fuel handling and storage equipment for fuel assemblies from the two plants are not compatible. Monticello has been operating since December 1970 and is also confronted with the similar problem of spent fuel storage capacity. The licensee cannot assuredly rely on other power facilities to provide additional storage capability except on a short term emergency basis. If space were available in another reactor facility, the costs would probably be comparable to the cost of storage at a commercial storage facility.

6.4 Reduced Plant Output

Nuclear plants are usually base-loaded because of their lower costs of generating a unit of electricity compared to other thermal power plants on the system. Therefore, reducing the plant output to reduce spent fuel generation is not an economical use of the resources available. The total production costs remain essentially constant, irrespective of plant output. Consequently, the unit cost of electricity is increased proportionately at a reduced plant output. If the plant is forced to substantially reduce output because of spent fuel storage restriction, the licensee would be required to purchase replacement power or operate its higher cost fossil-fired units, if available, without any accompanying environmental advantage. The cost of electricity would therefore be increased without any likely reduction of environmental impact.

6.5 Shutdown of Facility

Storage of spent fuel from the PINGP units in the existing racks is possible but only for a short period of time. As discussed above, if expansion of the SFP capacity is not approved and if an alternate storage facility is not located, NSP would have to shut down Unit No. 1 in late 1984 and Unit No. 2 in early 1984 due to a lack of spent fuel storage facilities, resulting in the cessation of at least 1040 MWe net electrical energy production.

The incremental cost for providing replacement power if both units were shutdown would be approximately $160 million dollars per year. This would be the cost of increased use of NSP's coal-fired and oil-fired generating facilities and the purchase of some replacement power from other utilities. This does not reflect that the $370 million dollar investment would be idle and that the PINGP would have to be maintained in standby or decommissioned.

6.6 Comparison of Alternatives

In Section 4 of this environmental impact appraisal the incremental environmental impacts of the proposed expansion of the SFP storage capacity were evaluated and were found to be insignificant. Therefore, none of the alternatives to this action offers a significant environmental advantage. Furthermore, alternatives (1), reprocessing, and (2), storage at an independent spent fuel storage facility, are not presently available to the licensee and are not likely to become available in time to meet the licensee's need. Alternative (3), shipment to another reactor site, would be a short term emergency solution but would eventually involve shipment to another temporary storage facility. Alternatives (4), reducing the plant output, and (5), shutdown of the facility, would both entail substantial additional expense for replacement electrical energy.

Table 1 presents a summarized comparison of the alternatives, in the order presented in Subsections 6.1 through 6.5. From inspection of the table, it can be seen that the most cost effective alternative is the proposed SFP modification, which is included as alternative 6. The SFP modification would provide the required storage capacity, while minimizing environmental effects, capital cost and resources committed. The staff therefore concludes that expansion of the PINGP SFP storage capacity is superior to the alternatives available or likely to become available within the necessary time frame.

TABLE 1

COMPARISON OF ALTERNATIVES

	Alternative	Cost	Benefit
1.	Reprocessing of Spent Fuel	N/A	Continued production of electrical energy by Units 1 & 2. This alternative is not available now.
2a.	Storage at Reprocessor's Facility	\$3,000 to \$6,000/ assembly per yr* plus shipping costs of \$12,000 per assembly.	Continued production of electrical energy by Units 1 & 2. This alternative is not available now or in the foreseeable future.
2b.	Storage at a new Independent Facility	\$20,000-\$40,000/assembly plus operating and transportation costs, and environmental impacts related to development of a new facility.	Continued production of electrical energy by Units 1 & 2. This alternative could not be available in time to meet the present storage needs of the PINGP.
3.	Storage at Other Nuclear Plants	Costs of shipment to other facility plus cost for subsequent shipment to an ISFSI; increased environmental costs of extra shipping and handling.	Continued production of electrical energy. However, this alternative is unlikely to be available.
4.	Reduction in Plant Output	See below for replacement electricity costs. Amount of replacement required would be equivalent to at least 50% reduction in rated output of Units 1 and 2.	Continued production of electrical energy by Units 1 and/or 2 - but at much higher unit cost. The generation of replacement electricity elsewhere would probably create no less impact.

*Since NFS and MO are not accepting fuel for storage, the cost range reflects prices that were quoted in 1972 to 1974.

TABLE 1

COMPARISON OF ALTERNATIVES

	Alternative	Cost	Benefit
5.	Reactor Shutdown	Increased electric production expenses are estimated to be approximately $160 million/yr ($438,000/day) if both units are shut down, plus the costs of maintenance and security of the plant.	Environmental impacts associated with plant operation would cease but the generation of replacement electricity elsewhere would probably create no less impact.
6.	Increased Storage Capacity of PINGP SFP	$7,600/added assembly storage space	Continued production of electrical energy by PINGP Units 1 & 2.

7.0 EVALUATION OF PROPOSED ACTION

7.1 Unavoidable Adverse Environmental Impacts

7.1.2 Radiological Impacts

Expansion of the storage capacity of the SFP will not create any significant additional adverse radiological effects. As discussed in Section 4.4, the additional total body dose that might be received by an individual at the site boundary or the estimated population within a 50-mile radius is less than 0.00004 mrem/yr and 0.0002 person-rem/yr, respectively, and is significantly less than the natural fluctuations in the dose this population would receive from background radiation. The total dose to workers during removal of the present storage racks and installation of the new racks is estimated to be about 40 man-rem. Operation of the plant with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the present total annual occupational exposure at this facility.

7.2 Relationships Between Local Short Term Use of Man's Environment and the Maintenance and Enhancement of Long Term Productivity

Expansion of the SFP storage capacity would permit more efficient use of the land already committed to this purpose. There would be no other significant changes from the evaluation in the FES.

7.3 Irreversible and Irretrievable Commitments of Resources

7.3.1 Water, Land and Air Resources

The proposed action will not result in any significant change in the commitments of water, land and air resources as identified in the FES. No additional allocation of land would be made; the land area now used for the SFP would be used more efficiently by reducing the spacings between fuel assemblies.

7.3.2 Material Resources

Under the proposed modification, the present spent fuel storage racks will be replaced by new racks that will increase the storage capacity of the SFP by 699 spent fuel assemblies. The new spent fuel storage racks consist of storage tubes on 9.5 inch centers which have three components: an inner type 304 stainless steel square tube with an inside dimension of 8.27 inches, a layer of neutron absorbing material sandwiched in between the inner and outer tubes and a 0.090 inch thick outer square tube of type 304 stainless steel. The largest storage rack consists of a 7 X 8 array of individual storage boxes, a base with four legs, and various bracing and support members. The fuel assemblies sit on bars across the bottom of each storage box. The tops of the storage boxes are flared to form a lead-in funnel. A total of sixteen 7 X 8 racks and fourteen 7 X 7 racks each weighing less than 12.4 tons will be used resulting in a total weight of less than 744,000 pounds.

Thus, the resources to be committed for fabrication of the new spent fuel storage racks total less than 744,000 pounds of stainless steel. The amount of stainless steel used annually in the U. S. is about 2.82×10^{11} lbs. The material is readily available in abundant supply. The amount of stainless steel required for fabrication of the new racks is a small amount of this resource consumed annually in the U. S. and therefore can be ignored in this Appraisal. The amount of boron required in the borated rack is insignificant. We conclude that the amount of material required for the new racks at PINGP is insignificant and does not represent a significant irreversible commitment of material resources.

8.0 BENEFIT-COST BALANCE

This section summarizes and compares the cost and the benefits resulting from the proposed modification to those that would be derived from the selection and implementation of each alternative. Table 1 presents a tabular comparison of these costs and benefits. The first three alternatives are not possible at this time or in the foreseeable future except on a short term emergency basis. Alternatives 4 and 5 have higher cost and no less environmental impacts than that of increasing storage capacity of PINGP SFP.

From examination of the table, it can be seen that the most cost-effective alternative is the proposed spent fuel pool modification. As evaluated in the preceding sections, the environmental impacts associated with the proposed modification would not be significantly changed from those analyzed in the Final Environmental Statement for PINGP Units 1 and 2 issued in May 1973.

9.0 BASIS AND CONCLUSION FOR NOT PREPARING AN ENVIRONMENTAL IMPACT STATEMENT

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51. We have determined that the proposed license amendment will not significantly affect the quality of the human environment and that there will be no significant environmental impact attributable to the proposed action other than that which has already been predicted and described in the Final Environmental Statement for PINGP dated May 1973. Therefore, the staff has found that an environmental impact statement need not be prepared, and that pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Dated: May 13, 1981

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 48 and 42 to Facility Operating License Nos. DPR-42 and DPR-60, respectively, issued to Northern States Power Company (the licensee), which revised the Technical Specifications for operation of the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments authorize replacement of the existing racks in the spent fuel pool of the facility with borated racks of a design which would provide a maximum of 1582 storage locations. This capacity will be limited to allow the storage of up to 1120 assemblies resulting from normal operation of Units 1 and 2. The modification and subsequent use of the pool permits a total of 1120 fuel assemblies to be stored instead of the previously authorized total of 687 assemblies.

The application for the amendments 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 amendments. Notice of Consideration of Proposed Modification to Facilities Spent Fuel Storage Pool in connection with this action was published in the Federal Register on March 12, 1980 (45 FR 16056).

A Request for Hearing and Petition for Leave to Intervene was filed by the State of Minnesota, by its Minnesota Pollution Control Agency and by its Attorney General (the petitioner) on April 9, 1980. Subsequently the Petitioner requested leave to withdraw its Petition for Leave to Intervene pursuant to a Stipulation duly executed by the Licensee, the NRC staff, and the Petitioner. The Stipulation has been incorporated into these Amendments to the License. Therefore, there being no issue to be heard by the Atomic Safety and Licensing Board, by ORDER dated October 24, 1980, the Board dismissed this proceeding.

The Commission has prepared an environmental impact appraisal of the action being authorized and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action significantly greater than that which has already been predicted and described in the Commmission's Final Environmental Statement for the facility dated May 1973, and the action will not significantly affect the quality of the human environment.

For further details with respect to this action, see (1) the application for amendment dated January 31, 1980, as supplemented June 10, and November 21, 1980, January 14, February 3, March 10, March 31 and April 20, 1981, (2) Amendment Nos. 48 and 42 to License Nos. DPR-42 and DPR-60, (3) the Commission's concurrently issued Safety Evaluation, and (4) the Commission's concurrently issued Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Environmental Conservation Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single 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 Licensing.

Dated at Bethesda, Maryland, this 13th day of May, 1981.

FOR THE NUCLEAR REGUALTORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing