NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

July 21, 1978

Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42
50-306 DPR-60

Control of Heavy Loads Near Spent Fuel

Your May 17, 1978 generic letter to licensees of power reactors asked nine questions pertaining to the control of heavy loads near spent fuel. This issue was discussed and documented extensively on the Prairie Island docket as part of a program to increase the spent fuel pool storage capacity and the ensuing public hearing. Of particular relevance is the following quote from the the Staff Safety Evaluation from D K Davis to L O Mayer dated April 20, 1977.

> "In order to ensure that no offsite exposures due to possible cask handling accidents in the spent fuel pool area are in excess of those calculated for the fuel handling accident in the Safety Evaluation Report of September 1972, we will require that the licensee demonstrate within six to nine months prior to using a cask in the auxiliary building either (1) that no scenario exists whereby any shielded cask which may be employed may tip into the large pool (no. 2) or (2) that fuel stored in the large pool (no. 2) which may be struck by any such tipping cask will have decayed for at least 120 days before a shielded cask is moved into the auxiliary building. Existing technical specifications (T.S.3.8-2.B.1) preclude the use of any heavy loads over or in either spent fuel pool when fuel is stored in that pool."

The requests for information of your May 17 letter are repeated below followed by our response.

NRC REQUEST 1

Provide a diagram which illustrates the physical relation between the reactor core,the fuel transfer canal,the spent fuel storage pool and the set down, receiving or storage areas for any heavy loads moved on the refueling floor.

NSP Response:

These diagrams are attached. Additional diagrams can be found in the Prairie Island FSAR

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NRC REQUEST 2

Provide a list of all objects that are required to be moved over the reactor core (during refueling), or the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram of the movement path utilized (including carrying height) and the frequency of movement.

NSP Response:

The Prairie Island FSAR includes an analysis of an accident involving the drop of a fuel assembly from the maximum possible height. Consequences of this postulated event were shown to be acceptable. This response will therefore consider only those loads weighing greater than a fuel assembly. License requirements (Technical Specification 3.8-2.B.1) preclude the handling of heavy loads over the spent fuel pool when fuel is stored in the pool. There are three loads required to be moved over the reactor cavity during refueling, the first two of which are too large to fit into the reactor vessel.

Missile Shield - This is a rectangular component 10 feet by 24.5 feet by 1 foot high, weighing 20 tons. It must be lifted 30 feet, and moved to a temporary location on the refueling floor each refueling outage.

Vessel Head - This is a 13 foot diameter by 5.7 foot high dome weighing 80 tons. It must be lifted 30 feet and moved to a temporary storage location on the second floor each refueling outage.

Upper Reactor Internals - This is a 10.5 foot diameter by 10 foot high cylindrical component weighing 25 tons. It must be lifted 20 feet and moved to a temporary storage location in the refueling cavity each refueling outage.

NRC REQUEST 3 & 4

What are the dimensions and weights of the spent fuel casks that are or will be used at your facility?

Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC Staff.

NSP Response:

An analysis of a cask drop accident is included in the Prairie Island FSAR along with a description of the spent fuel shipping cask assumed in the analysis. Additional information on shipping casks and the ability to safely handle the cask and other heavy loads will be included in the study required by the April 20, 1977 Staff safety evaluation.

NRC REQUEST 5

Identify any heavy loads that are carried over equipment required for safe shutdown of a plant that is operating at the time the load is moved. Identify what equipment could be affected in the event of a heavy load handling accident (piping, cabling, pumps, etc.) and discuss the feasibility of such an accident affecting this equipment. Describe the basis for your conclusion.

NSP Response:

No heavy loads are moved in the reactor containment building unless the reactor is already shutdown.

During reactor operations, heavy loads are not routinely moved in the auxiliary building or turbine building over any equipment required for safe shutdown of the plant. However, in the turbine building it has on rare occasions been necessary to move heavy loads over the safeguards switchgear rooms. Static analysis of turbine building floor strenth has shown that it will support the maximum load (low presure turbine rotor - 80 tons).

NRC REQUEST 6

If heavy loads are required to be carried over the spent fuel storage pool or fuel transfer canal at your facility, discuss the feasibility of a handling accident which could result in water leakage severe enough to uncover the spent fuel. Describe the basis for your conclusions.

NSP Response:

The heaviest load expected to be moved over the spent fuel pool, is the spent fuel shipping cask. (Because of space limitations and practical travel path restrictions, it is very unlikely that any other loads greater than the weight of a fuel assembly will be moved over the spent fuel pool). The Prairie Island FSAR discusses the analysis of the drop of a shipping cask and shows the consequences to be minimal, if any, leakage.

NRC REQUEST #7

Describe any design features of your facility which affect the potential for a heavy load handling accident involving spent fuel, e.g., utilization of a single failure proof crane.

NSP Response:

This subject is discussed in the Prairie Island FSAR and cask tip will be further discussed in the study required by the April 20, 1977 Staff safety evaluation.

NRC REQUEST 8

Provide copies of all procedures currently in effect at your facility for the movement of heavy loads over the reactor core during refueling, the spent fuel storage pool, or equipment required for safe shutdown of a plant that is operating at the time the move occurs.

NSP Response:

Northern States Power has consistently maintained the position that operating procedures are inappropriate documents for docketing purposes. There are numerous reasons for this position. Operating procedures are written for use by operators and not as licensing documents. They are often bulky, use terminology not readily understood by those outside the plant, and make reference to non-docketed information. Procedures are reviewed and revised frequently as required by the technical specifications. Distribution is controlled and restricted to avoid the existence of out-of-date and possibly incorrect information. We believe the following two alternatives to the NRC request to be acceptable . First, knowledgeable members of your staff could meet in our offices to review and discuss the operating procedures. Preferably, the Region III Office of Enforcement and Inspection personnel, who have a working knowledge of our operations and procedures, might be consulted. We suggest that a review of our operating practices be deferred until the Staff reviews the study required by the April 20, 1977 safety evaluation.

NRC REQUEST 9

Discuss the degree to which your facility complies with the eight (8) regulatory positions delineated in Regulatory Guide 1.13 (Revision 1, December, 1975) regarding Spent Fuel Storage Facility Design Basis.

NSP Response:

The Prairie Island Plant conforms with the requirements of this guide with one exception. Position statements 3 and 5a discuss interlocks and design features to prohibit carrying heavy loads over spent fuel. The design of the fuel storage facility is such that it is possible but difficult to move any object over the spent fuel because of physical travel path restrictions. The existing design meets the remaining regulatory positions of Regulatory Guide of 1.13.

Yours very truly,

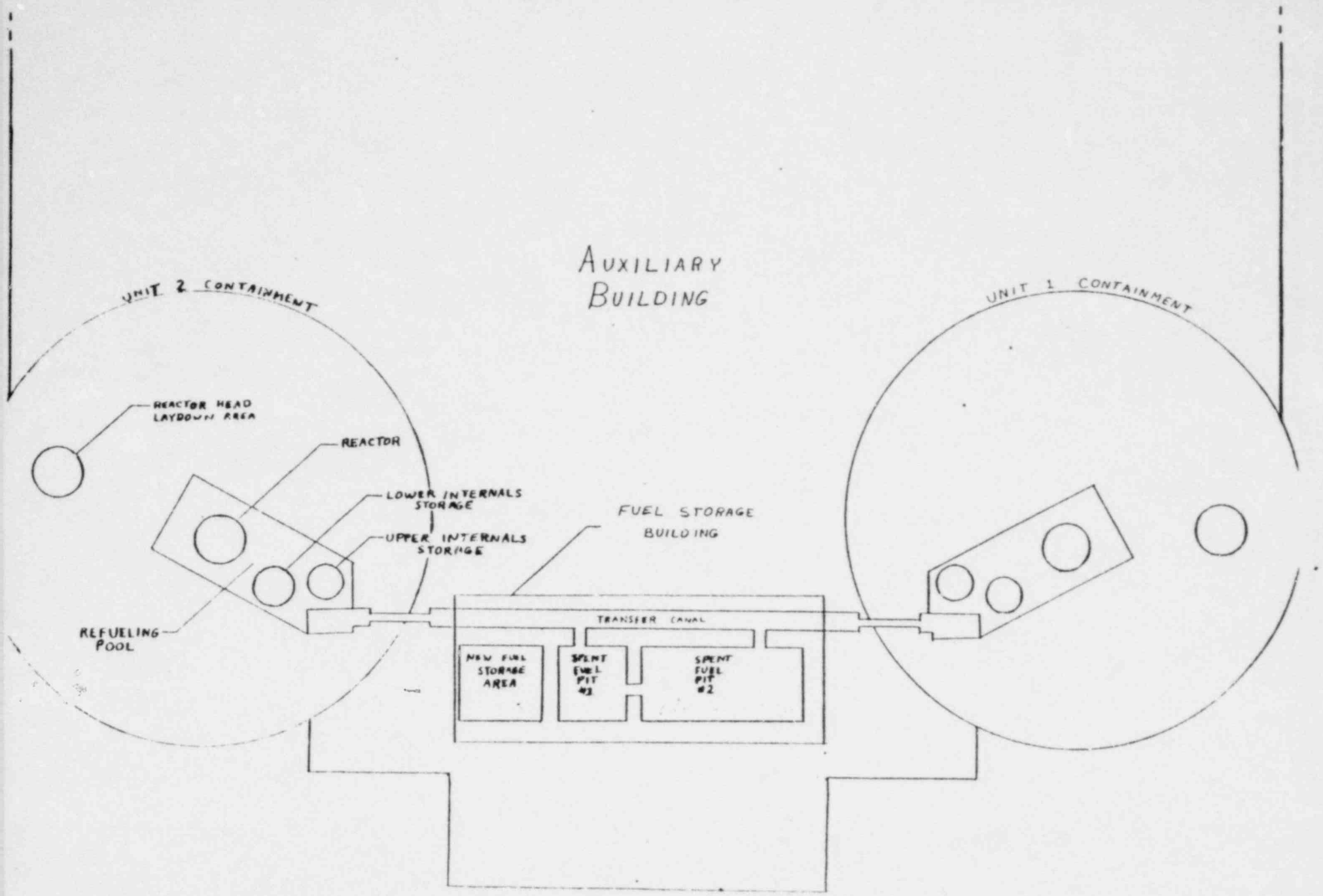
L.O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/deh

Attachment

cc: J G Keppler
G Charnoff


AUXILIARY BUILDING
UNIT 2 CONTAINMENT
UNIT 1 CONTAINMENT
REACTOR HEAD LAYDOWN AREA
REACTOR
LOWER INTERNALS STORAGE
UPPER INTERNALS STORAGE
REFUELING POOL
FUEL STORAGE BUILDING
TRANSFER CANAL
NEW FUEL STORAGE AREA
SPENT FUEL PIT #1
SPENT FUEL PIT #2

Prairie Island Unit 1 & 2

APR 26 1977

M. GROTENHUIS

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO MODIFICATION TO THE SPENT FUEL POOL
FACILITY LICENSE NOS. DPR-42 AND DPR-60
NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2
DOCKET NOS. 50-282 AND 50-306

March 21, 1977
Updated 14 1977

1.0 INTRODUCTION

By letter dated November 24, 1976 supplemented by submittals of March 1 and March 11, 1977, Northern States Power Company (NSP) requested an amendment to Facility License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). This request was made to permit the provision of additional storage capacity in the PINGP Spent Fuel Storage Pool (SFP). The proposed modification would increase the capacity of the SFP from the present capacity of 198 fuel elements to a future capacity of 687. The extra capacity would come from installing new racks in place of the old racks, with the exception of the use of one of the present racks which holds 12 fuel elements. The expansion of the SFP would allow the PINGP to operate until 1982 without shipping fuel from the site. This would still leave room for a complete core discharge.

The PINGP has a compartmented double pool which serves both units. The smaller pool (pool no. 1) will primarily be used for core off-loading. The larger pool (pool no. 2) will be used for extended storage.

The spent fuel storage racks are made up of modular stainless steel structures, comprised of individual cavities 13.3 inch on centers. The location of the spent fuel storage facility within the plant is shown in Figures 3.1 through 3.5 of the Design Report. The arrangement and details of the new fuel racks in the storage pools are shown in Figures 3.6 and 3.7 of the November 24, 1976 submittal.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

The proposed replacement racks for the spent fuel assemblies are to be made up of individual containers which are approximately 14 feet long and which have an overall square cross section of

8.674 inches along each side. These containers are to be fabricated from 0.187 inch thick type 304 stainless steel plate, and are to be spaced in the racks on a nominal 13.3 inch pitch. Allowing for fabrication and assembly tolerances, the minimum pitch is to be 13.227 inches. At the nominal pitch, there will be more than 4.5 inches of water between neighboring storage containers. The 13.3 inch pitch combined with the overall dimension of the fuel assembly, which is 7.784 inches, leads to a fuel region volume fraction of 0.34 for the storage lattice.

NSP states that a uranium-235 enrichment of 3.5 weight percent was used in its calculations for the neutron multiplication factor. For a maximum uranium dioxide density of 95 percent of the theoretical density in the 14x14 array with 179 fuel rods per fuel assembly, this 3.5 percent enrichment results in a fuel loading of 39 grams of uranium-235 per axial centimeter of fuel assembly.

The Nuclear Associates International Corporation (NAI) performed the criticality analysis for NSP. Their CHEETAH-P computer program, which is a modified version of the LEOPARD program, was used to obtain four energy group cross sections for use in PDQ-7 diffusion theory calculations. NAI also used the AMPX program to obtain 123 group cross sections for use in a KENO-IV Monte Carlo calculation. These computer programs were first used to calculate the neutron multiplication factor, k_∞, for an infinite array of infinitely long fuel assemblies, i.e., the cell k_∞, in the nominal storage lattice. This resulted in a maximum cell k_∞ of 0.867 for the nominal lattice of fuel assemblies on a 13.3 inch pitch and at a temperature of 68°F.

The sensitivity of k_∞ to a decrease in the spacing of the fuel assemblies was then determined by calculating a storage lattice with a 12.8 inch pitch. This resulted in a cell k_∞ of 0.884. Calculations of the neutron multiplication factor were also made for four higher temperatures. These showed that the increase in cell k_∞ temperature up to 212°F will be less than 0.002.

Since there will be unobstructed water areas between the periphery of the proposed storage arrays and the sides of the pool, the possibility will exist for a fuel assembly being incorrectly placed into one of these areas and allowing it to have a closer spacing to a stored fuel assembly than it could if it were correctly placed in the rack. In its submittal, NSP stated that to take this into account the periphery of the new racks will be designed to maintain a minimum of four inches of water between an exterior fuel assembly and any fuel assembly stored in the

racks. To conservatively calculate the neutron multiplication factor for this situation, NAI assumed that a whole row of fuel assemblies, rather than just one, was brought up to the outside of the filled rack, and they assumed that the six closest fuel assemblies in the rack were loaded off-center towards this exterior row of fuel assemblies. For the minimum four inches of water between this exterior row of fuel assemblies and the fuel assemblies in the rack, NAI calculated a neutron multiplication factor of 0.92.

The above quoted results of the licensee's calculations compare very favorably with results of parametric calculations made with another method for a similar fuel pool storage lattice. By assuming new, unirradiated fuel with no burnable poison these calculations give the maximum neutron multiplication factor that could be obtained throughout the core life of the nominal fuel assembly. This includes the buildup of plutonium. A major part of the remaining uncertainty in the accuracy of the calculated neutron multiplication factor is in the effective cross sections for the stainless steel. For this and other reasons, the uncertainty in the calculation of the neutron multiplication factor decreases as the amount of stainless steel is decreased.

Thus, since there will be only one thickness of stainless steel between two fuel assemblies in the accident condition described, i.e., where a fuel assembly is inadvertently brought as close as possible to the outside of one of the fuel rack modules, the accuracy of the calculated neutron multiplication factor of 0.92 for this situation will be better than that for the storage lattice. Thus, the actual neutron multiplication factor for this case, when the maximum temperature effect and all uncertainties are included, will be less than 0.94. Since this is less than the NRC acceptance criterion of 0.95, we find it to be acceptable.

We find that the maximum neutron multiplication factor in these pools will be obtained when a fuel assembly is brought up to the outside of a fully loaded storage rack. We also find that when the fuel assemblies in the pool have no more than 39 grams of uranium-235 per axial centimeter of assembly and there is a minimum of four inches of water between an exterior fuel assembly and any stored fuel assembly, the neutron multiplication factor will be less than 0.94. Since this factor is less than our acceptance criterion of 0.95, we find the proposed design acceptable. On this basis, we conclude that when the plant's Technical Specifications are modified to prohibit the storage of fuel assemblies that contain more than 39 grams of uranium-235 per longitudinal centimeter of assembly in the pool, the rack design will preclude criticality.

18, 19, 20, 21

2.2 Spent Fuel Cooling

The licensed thermal power for each of the two units of the Prairie Island Nuclear Generating Plant is 1650 MWt. The refueling of each of the two units, which consists of replacing about 40 of the 121 fuel assemblies, is done annually, and the refuelings of the two units are staggered so that a refueling at the plant occurs about every six months. In its submittal NSP assumed that there would be a cooling period of 150 hours after the reactor is shut down before the unloading of fuel assemblies from the reactor vessel is started. With this assumption NSP stated that the maximum heat load in the fuel pool, which will be for a full core offload which fills the capacity of the racks, will be approximately 17.0×10^6 BTU/hr (5 MWt).

In the Prairie Island FSAR it is stated that the spent fuel pool cooling system consists of two half-capacity pumps, each with a design flow rate of 700 gallons per minute, and two full-capacity heat exchangers, each rated for removing 7.89×10^6 BTU/hr (2.3 MWt) with about a 11°F drop in the spent fuel pool outlet water temperature. In the FSAR one of these heat exchangers is designated for cooling 120°F water while the other, which is called a backup, is designated for cooling 140°F water. In their submittal NSP stated that for the maximum heat load of 17×10^6 BTU/hr (5 MWt) from the unloading of a full core the fuel pool outlet water will heat to a maximum of 143°F. The FSAR also states that the Component Cooling System, which is used to cool the spent fuel heat exchanger, has four heat exchangers each of which is designed to remove 29×10^6 BTU/hr (8.5 MWt) and that the normal design duty of the Component Cooling System with the reactor at power is 30.4×10^6 BTU/hr (8.9 MWt).

The total volume of water in the two spent fuel storage pools is stated to be 4.75×10^4 cubic feet. NSP also stated that fuel pool number 2, the larger pool, will contain about 2-1/2 times the water inventory in the fuel pool number 1. For this ratio there will be about 1.36×10^4 cubic feet of water in the fuel pool number 1 and 3.39×10^4 cubic feet of water in fuel pool number 2.

By comparing NSP's calculated heat loads with those obtained by using the method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, we find that for a burnup of 33,000 MWD/MTU the maximum heat load of 17×10^6 BTU/hr (5 MWt) will be obtained if about ten days are taken to transfer the full core to the fuel pool after the 150 hours of cooling time. We also find that

82 percent of this heat load is from the full core. Thus the heat load from 566 fuel assemblies, which were assumed to be in the pool at the time the full core was offloaded, is .18 of 17×10^6 BTU/hr or 3.1×10^6 BTU/hr. This is a sufficiently small percentage of the design duty of the Component Cooling System that it can be taken care of by the conservatism in the design.

For the normal refuelings at six month intervals we find that the heat load in the spent fuel pool will always be less than the design value of 7.89×10^6 BTU/hr. NSP's calculated fuel pool outlet water temperatures are consistent with the stated flow rates and design capabilities of the heat exchangers. Thus we find that for the normal refuelings the fuel pool outlet water temperature will not be more than 120°F.

If both spent fuel pool cooling systems happened to fail just after transferring a full core to the fuel pool number 1, it would take about four hours to heat the 1.36×10^4 cubic feet of water to the boiling point. Since this is the smaller pool and has the maximum heat load, this would be the minimum time to boiling. Four hours would be sufficient time to open the fuel transfer slots to the fuel pool number 1 if they were not already open. This would establish communication between the fuel pool number 1, the fuel pool number 2 and the storage canal. This would make an adequate amount of cooling water available for a period of more than ten hours. The PINGP received a Construction Permit prior to the issuance of Regulatory Guide 1.13, December 1975, therefore it was not built specifically according to that guidance. It does, however, satisfy the intent of the guide. The Chemical and Volume Control Systems holdup tanks are available as a siesmic Class 1 source of water to the spent fuel pool. This also represents a backup source of water for the fuel pool.

We find that the stated cooling capacity, as discussed above and in the EIA, will be sufficient to maintain the spent fuel pool outlet water temperature below 143°F. We also find that in the unlikely event that all fuel pool cooling is lost, ten hours would be sufficient time to either make repairs or find an alternate source of cooling water for the spent fuel pool.

2.3 Installation of Racks and Fuel Handling

There are 30 fuel assemblies stored in the spent fuel pools at this time. NSP proposes to move these to the east end of the pool number 2 prior to installing the new racks in the pool

number 1. This approach will preclude the movement of racks over stored fuel assemblies. After the new racks are installed in the pool number 1, all stored spent fuel assemblies will be transferred to the new racks and a structural cover will be placed over the pool number 1. This cover will be designed to prevent the heaviest spent fuel rack from falling into the stored spent fuel when dropped from a height of six inches above the cover, and administrative controls, which could conceivably require a manual measurement of the rack height, will be used to prevent lifting racks higher than six inches above the structural cover.

In order to ensure that no offsite exposures due to possible cask handling accidents in the spent fuel pool area are in excess of those calculated for the fuel handling accident in the Safety Evaluation Report of September 1972, we will require that the licensee demonstrate within six to nine months prior to using a cask in the auxiliary building either (1) that no scenario exists whereby any shielded cask which may be employed may tip into the large pool (no. 2) or (2) that fuel stored in the large pool (no. 2) which may be struck by any such tipping cask will have decayed for at least 120 days before a shielded cask is moved into the auxiliary building. Existing technical specifications (T.S. 3.8-2.B.1) preclude the use of any heavy loads over or in either spent fuel pool when fuel is stored in that pool.

The above procedures should eliminate the possibility for dropping a rack or a fuel cask on to stored fuel assemblies. We conclude therefore that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.

2.4 Structural and Mechanical

A review based on the applicable parts of Sections 3.7 and 3.8 of the Standard Review Plan was completed for the following items: the supporting arrangements for the racks including restraints; quality control for design, fabrication and installation procedures; the structural design and analysis procedures for all loads, including seismic and impact; the required load combinations; the structural acceptance criteria; and the application of industry codes. The fuel pool is supported above grade at the elevation of 713'6" by the auxiliary building. The seismic input to the rack modules is described by amplified floor response spectra obtained from the Technical Report, "Amendment I of the FSAR Northern States Power Company Prairie Island Nuclear Generating

Plant." The applicable floor response spectra correspond to mass point 25. Three components of earthquake input were simultaneously applied to the racks and the maximum responses were combined as per Regulatory Guide 1.92 entitled "Combining Modal Responses And Spatial Components In Seismic Response Analysis." The existing pool structure was analyzed to account for the increased dead load and seismic loads. The load combinations and the acceptance criteria satisfy parts of Section 3.8.4.II.3 of the Standard Review Plan, which are applicable to concrete structures.

The performance requirements during service life of type 304 stainless steel, used in the fabrication of the spent fuel racks, were reviewed for material compatibility in the spent fuel environment. The proposed modification of fuel racks does not involve the use of any poison material.

Analyses, design, fabrication, and installation of the proposed racks satisfy accepted criteria and conform to the rules of Subsection NF, Linear Components Class 2, of Section III, Division 1 of the ASME Boiler and Pressure Vessel Code. The racks are designed as seismic Category I structures. The existing pool structure has been qualified to account for the additional loads due to high density storage of the fuel assemblies. The qualification of the pool structure satisfies the acceptance criteria described in Section 3.8.4.II.3 of the Standard Review Plan. We conclude that the subject modification proposed by the licensee is acceptable.

2.5 Radiation Levels

We have reviewed the licensee's plan for removal, disassembly and offsite shipment of the old racks and installation of the new racks. The occupational radiation exposure for this operation is estimated to be about 28 man-rem. The racks will be installed over a short period of time, that is weeks rather than years. Therefore, since this is a one-time exposure it is not directly comparable to the annual doses during normal operation in the SFP. We consider this to be a reasonable estimate.

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 and by utilizing realistic assumptions for radionuclide concentrations in the SFP water and for occupancy times. This analysis indicates that the occupational radiation exposure resulting from this proposed action represents less than one percent of

the present total annual occupational burden at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses as low as reasonably achievable and within the limits of 10 CFR 20. The waste treatment system is adequate and the fuel integrity is not expected to deteriorate while stored in the pool as discussed in the EIA. Thus, we conclude that storing additional fuel for the longer period requested by NSP in the SFP will not result in any significant increase in doses received by occupational workers.

The only change in offsite dose would be that associated with the small incremental increase in effluents released from the facility. As is discussed in the accompanying Environmental Impact Appraisal, the effect of that increment would be negligible.

3.0 SUMMARY

Our evaluation supports the conclusion that the proposed modification to the SFP at PINGP is acceptable because:

(1) The physical design of the new storage racks will preclude criticality for any moderating condition with the limits imposed.

(2) The SFP cooling system has adequate cooling capacity.

(3) The installations and use of the proposed fuel handling racks can be accomplished safely.

(4) The structural design and the materials of construction are adequate.

(5) The increase in radiation doses due to the storage of additional fuel in the SFP would be negligible.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 15, 1977

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations. The lifting device is permanently attached to the reactor vessel head.

Reactor Internals Lifting Device

The reactor internals lifting device is a structural frame used in moving the reactor internals. The device is lowered onto the guide tube support plate of the internals and is manually bolted to the support plate by three bolts. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package.

Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the floor along the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position. The manipulator can lift only one fuel assembly at a time.

f) An interlock on the hoist drive circuit in the up direction, which permits the hoist to be operated only when either the OPEN or CLOSED indicating switch on the gripper is actuated

g) An interlock of the bridge and trolley drives, which prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the refueling canal centerline. The trolley drive is locked out when the bridge is beyond the edge of the core.

Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing and the manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a Design Basis earthquake.

Spent Fuel Pool Bridge

The spent fuel pool bridge is a wheel-mounted walkway, spanning the spent fuel pool which carries electric monorail hoists on an overhead structure. The fuel assemblies are moved within the spent fuel pool by means of a long handled tool suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

The Auxiliary Building Crane

The Prairie Island Auxiliary Building Crane, which is used for handling spent fuel shipping casks, is designed to minimize the possibility of dropping such a cask. The crane has been designed, fabricated and qualified in accordance with the Electric Overhead Crane Institute (EOCI) Standard No. 61, American National Standard Institute Standard B-30.2.0, 1967 Edition (and Pioneer

Service and Engineering Company Standard Specification for Powerhouse Overhead Electrical Traveling Cranes). The rated load on the crane cable and hook is 125 tons. All crane structural members have been designed to withstand impact loads per applicable specifications. A seismic evaluation has been performed for the loaded condition. Numerous safety features have been incorporated in the design of the crane. Among these are the following:

1. Design of the hoist cables incorporate a safety factor of five based on the rated load and efficiency of lifting tackle.

2. All parts subjected to dynamic strains such as gears, shafts, drums, blocks and other integral parts have a safety factor of five.

3. Two separate magnetic brakes are provided as well as an eddy cunent brake. Each magnetic brake provides a braking force in excess of 200% of rated load. The eddy cunent brake assures that smooth lowering and hoisting speeds can be maintained regardless of the load on the hook andthat the load can be safely lowered even if both magnetic brakes fail.

4. The crane is capable of raising, lowering and transporting occasional loads of 125 percent of rated load without damage or distortion to any crane part.

5. The crane is provided with a type 302 stainless steel cable with a high strength steel core. The cable is made up of 7 strands consisting of 37 wires to each strand.

6. During acceptance inspection, the hook is subjected to a 200% overload test followed by a magnetic particle-inspection.

7. Motor controllers, with at least five uniformly proportional steps for control in each direction are provided for all crane motions. Built-in time delays are provided between steps. This will provide a smooth uniform acceleration which eliminates sudden jerking of the cask.

8. A fail-safe remote radio control system is provided for the crane. The radio controls parallel the master switch connections and all of the safety features built into the control system also apply when the radio transmitter is used. The radio has a very complex system for transmitting and receiving signals so it is impossible to duplicate this signal by any other means. Also, the signal differs for every crane so the transmitter for some other crane would not actuate the receiver panel for this crane.

Fuel Transfer System

The fuel transfer system, shown in Figure 9.5-1 is an underwater air-motor driven conveyor car that runs on tracks extending from the refueling canal through the transfer tube and into the fuel transfer canal. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then is raised to a vertical position in the fuel transfer canal.

During plant operation, the conveyor car is normally stored in the fuel transfer canal. A blind flange is bolted on the refueling canal end of transfer tube to seal the reactor containment. The terminus of the tube
24 outside the containment is closed by a gate valve.

9.5.3 SYSTEM EVALUATION

Underwater transfer of spent fuel provides essential ease and safety in handling operations. Water is an effective, economic, and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

37 a) Gamma radiation levels in the containment refueling area and fuel storage areas are continuously monitored (see Section 11.2.3). These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the control room of an abnormal core flux level.

b) Violation of containment integrity is not permitted when the reactor vessel head is removed, unless adequate shutdown margin is maintained.

c) Whenever new fuel is added to the reactor core, a reciprocal curve of source range neutron count rate is plotted to verify the sub-criticality of the core.

d) The design of the spent fuel storage facility incorporates a Class I fuel pool enclosure for additional protection against the effects of a tornado.

In all, four barriers protect stored spent fuel against the effects of a tornado. They are:

1. The Class I fuel pool structure

2. The twenty-five feet of water that normally covers the spent fuel

3. The fuel storage racks

4. The Class I fuel pool enclosure

The potential effects of a tornado striking the fuel storage pool of existing light-water reactors, i.e. with no Class I fuel pool enclosure, has been thoroughly investigated.[(1), (2), (3)] Two key areas were examined:

(i) whether sufficient water could be removed from the pool to prevent cooling of the fuel and

(ii) whether missiles could potentially enter the pool and damage the stored fuel.

As these investigations[(1), (2), (3)] show, existing fuel pool designs are inherently capable of withstanding the effects of a tornado. The design of the fuel pool makes the removal of more than five feet of water due to tornado action highly improbable. With 25 feet of water covering the fuel racks, the removal of five feet of water is of no concern. Protection is provided against a wide spectrum of tornado-generated missiles by the water which covers the fuel racks.

Results of an evaluation of a spectrum of tornado-driven missiles indicates that a four inch or smaller pipe, wooden debris from the cooling towers or metal panel siding ripped off the building are the most probable sources of missiles.

Such debris would not damage the fuel because of the buoyant force exerted by the twenty-five feet of water covering the fuel. Only by arbitrarily assuming long cylindrical objects to be hurled to the fuel pool by winds acting on their maximum cross-sectional area and then impacting the pool with minimum cross-sectional area could a potential for damage be shown[2]. The probability of such an event has been shown to be less than 10^{-7}. Even in this highly unlikely case, a wide spectrum of these missiles can hit the pool without resulting in fuel damage or liner penetration.

In addition to this inherent protection provided by the fuel pool structure, the 25 feet of water covering the spent fuel and the fuel storage racks, a Class I fuel pool enclosure has been provided for additional protection. It is, therefore, concluded that adequate protection for tornado-generated missiles has been provided.

e) Protection against a dropped cask accident is provided by the design of both the auxiliary building crane, which is used to handle spent fuel shipping cask, and by the design of the fuel pool enclosure which limits the area of the spent fuel pool over that heavy objects can be moved.

To provide additional protection, prior to cask lifting operations, a detailed visual inspection is made of all mechanical and electrical components of the crane. Following the visual inspection an operational test will be conducted with no load on the hook. This test will verify that all controls are operating correctly. Following these tests a load test will be conducted by raising the fuel cask approximately one foot off its rail car. The prime purpose of this test is to verify that there is no load movement after a fixed period of time. Hoisting and lowering rates will then be determined to see that they comply with the vendor's recommendations.

However, an analysis of the pool has been made to determine its capability of withstanding the impact of a dropped fuel cask. The following assumptions
11 were used in this evaluation.

a. Cask weight of 100 tons

b. Cask is 6 feet in diameter

c. The cask is lifted only far enough above the refueling floor to provide the required clearance to permit manipulation of the cask. This would result in the potential for a 42 foot cask drop.

d. The cask utilizes lead for shielding

e. The cask has a drag co-efficient of 0.86 in falling through the water. This results in an equivalent free fall through air of 30 feet.

f. Approximately 40% of the energy is absorbed in deformation of the cask. This assumption is based on work performed by Knapp Mills Inc., Wilmington, Delaware and the Oak Ridge National Laboratory, Oak Ridge, Tenn.

The results of our analysis show the small fuel pool can withstand the consequences of a dropped cask with minimal, if any, leakage. 11

The layout of walls above and below the fuel pool bottom slab in the vicinity of the path of travel of the cask limits the clear span of the bottom slab to about 9 feet. This is approximately 1-1/2 times the pool bottom slab thickness. The impact load will, hence, be carried by arch action and not result in severe flexural cracking. The penetration of the cask computed as for a missile amounts to only about an inch. The stresses in the concrete and reinforcing steel, although exceeding yield, stay below the ultimate strength of the reinforcing steel. In our analysis, we assumed a decceleration path based on conventional collapse mechanism (limit design) principles. The energy was assumed to be absorbed by yielding of the reinforcing steel and by heat generation due to crushing of the concrete.

No leakage would result unless the SS liner were punctured or torn in the accident. Flexural cracking of the concrete is minimized due to arching action.

The above analysis was performed for the cask being dropped in the small pool. Use of the fuel transfer cask in the large pool is precluded by the fuel pool enclosure which only allows cask accessibility to the small pool.

It is difficult to postulate any cask drop in the small pool regardless of
11 assumptions which would result in sufficient damage to jeopardize the water-tight integrity of the large pool in which fuel is stored.

Pool leakage resulting from a cask drop accident through cracks and possible loss-of-integrity of the SS pool liner would be in the few gpm range. Expected leakage rate would be such to allow sufficient time (hours) to install gates in the fuel transfer slats to limit leakage to the small pool. In addition, the pool is provided with makeup capacity from the normal source of approximately 100 gpm.

When moving the fuel cask from the position in which it was loaded in the pool (bottom of cask at elevation 713' - 6") to its position on a railroad car, the cask will follow the following path:

1. Move vertically up thru the pool to its surface at elevation 753' - 6".

19 2. Continue vertically up thru air until the bottom of the cask is at about elevation 756' - 0", one foot above the fuel handling floor and 42' - 6" above the bottom of the pool.

3. Move horizontally south 33' - 4". During this move the cask will be over the small pool for 6' - 0"; over the fuel handling floor for 5' - 0"; then along the edge of an opening in the fuel handling floor for 22' - 4".

4. Move horizontally east about ten (10) feet.

5. Move vertically down about 66 feet until the cask rests on the railroad car whose top surface will be at about elevation 699' - 0".

Thus, the cask could be dropped 42' - 6" when over the small pool, or 66 feet when over the railroad car. It has been calculated that the speed of the cask after the 42' - 6" drop, mostly thru water, would be about the same as a 30 ft. drop thru air.

The cask has not yet been designed or purchased. It is expected that it will be designed to meet the requirements of federal regulation 49CFR - Part 173 which requires that the cask be capable of sustaining a 30-foot free drop onto a unyielding surface without releasing radioactive materials. If the cask, as finally purchased, is not capable of being dropped 66 ft. without damage, then it will be necessary to temporarily add energy absorbing material around the cask during the time it is being lowered to the railroad car. 19

Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane operator is available whenever changes in core geometry are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Malfunction Analysis

An analysis is presented in Section 14 concerning damage to one complete assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel handling incident.

Any suspected defective fuel assembly is placed in a failed fuel can and sealed to provide an isolated chamber for testing for the presence of fission products.

The failed fuel cans are stainless steel cylinders with lids that can be closed in place remotely. An internal gas space in the lid provides for

water expansion and for collection and sampling of fission product gases. Various remotely operable quick-disconnect fittings permit connection of the can to sampling loops for continuous circulation through the can.

If sampling shows the presence of fission products indicative of a cladding failure, the sampling lines are closed off by valves on the can and the encapsulated fuel assembly is removed to the spent fuel storage pit to await shipment. Design of the cans complies with federal regulation 10 CFR 71 so that the defective fuel can be stored and shipped while sealed in the failed fuel can.

9.5.4 TEST AND INSPECTION CAPABILITY

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests can be performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the in-core thermocouples, and the reactor vessel head water temperature thermocouple can be tested at the time of installation. The tests can be repeated on these electrical items before initial plant operation.