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

January 23, 2001

Mr. Mark Reddemann
Site Vice President
Kewaunee and Point Beach Nuclear Power Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT (TAC NO. MA7278)

Dear Mr. Reddemann:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 150 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated November 18, 1999, as supplemented on August 7, 2000.

The amendment revises the Kewaunee Nuclear Power Plant Technical Specifications to increase the allowable number of spent fuel assemblies stored in the spent fuel pools.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

John G. Lamb

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 150 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

Kewaunee Nuclear Power Plant

cc:

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
License No. DPR-43

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee) dated November 18, 1999, as supplemented on August 7, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 150 , are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 23, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 150

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
TS ii	TS ii
TS iii	TS iii
TS vi	TS vi
TS 3.8-1	TS 3.8-1
TS B 3.8-1	TS B 3.8-1
TS 5.4-1	TS 5.4-1
TS 5.4-2	TS 5.4-2
Figure 5.4-1	Figure 5.4-1

LIST OF FIGURES

Note:

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

3.8 REFUELING OPERATIONS

APPLICABILITY

Applies to operating limitations during REFUELING OPERATIONS.

OBJECTIVE

To ensure that no incident occurs during REFUELING OPERATIONS that would affect public health and safety.

SPECIFICATION

a. During REFUELING OPERATIONS:

1. Containment Closure

    a. The equipment hatch shall be closed and at least one door in each personnel air lock shall be capable of being closed[(1)] in 30 minutes or less. In addition, at least one door in each personnel air lock shall be closed when the reactor vessel head or upper internals are lifted.

    b. Each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool shall be monitored continuously.

3. The reactor will be subcritical for 148 hours prior to movement of its irradiated fuel assemblies. Core subcritical neutron flux shall be continuously monitored by at least TWO neutron monitors, each with continuous visual indication in the control room and ONE with audible indication in the containment whenever core geometry is being changed. When core geometry is not being changed at least ONE neutron flux monitor shall be in service.

4. At least ONE residual heat removal pump shall be operable.

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(1) Administrative controls ensure that:

- Appropriate personnel are aware that both personnel air lock doors are open,
- A specified individual(s) is designated and available to close the air lock following a required evacuation of containment, and
- Any obstruction(s) (e.g., cables and hoses) that could prevent closure of an open air lock can be quickly removed.

BASIS

The equipment and general procedures to be utilized during REFUELING OPERATIONS are discussed in the USAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident occurs during the REFUELING OPERATIONS that would result in a hazard to public health and safety.[1] Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (TS 3.8.a.2) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. A boron concentration of 2100 ppm, as required by TS 3.8.a.5, is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.[2] With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Interlocks are utilized during REFUELING OPERATIONS to ensure safe handling. Only one assembly at a time can be handled. The fuel handling hoist is dead weight tested prior to use to assure proper crane operation. It will not be possible to lift or carry heavy objects over the spent fuel pool when fuel is stored therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred forty-eight hour decay time following plant shutdown is consistent with the spent fuel pool cooling analysis and also bounds the assumption used in the dose calculation for the fuel handling accident. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass

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[1] USAR Section 9.5.2

[2] USAR Section 14.1

5.4 FUEL STORAGE

Applies to the capacity and storage arrays of new and spent fuel.

OBJECTIVE

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

SPECIFICATION

a. Criticality

i. The spent fuel storage racks are designed and shall be maintained with:

(1) Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter;

(2) $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties.

ii. The new fuel storage racks are designed and shall be maintained with:

(1) Fuel assemblies having a maximum enrichment of 56.067 grams Uranium-235 per axial centimeter;

(2) $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties;

(3) $k_{eff} < 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties.

iii. The spent fuel pool is filled with borated water at a concentration to match that used in the reactor refueling cavity and refueling canal during REFUELING OPERATIONS or whenever there is fuel in the pool.

b. Capacity

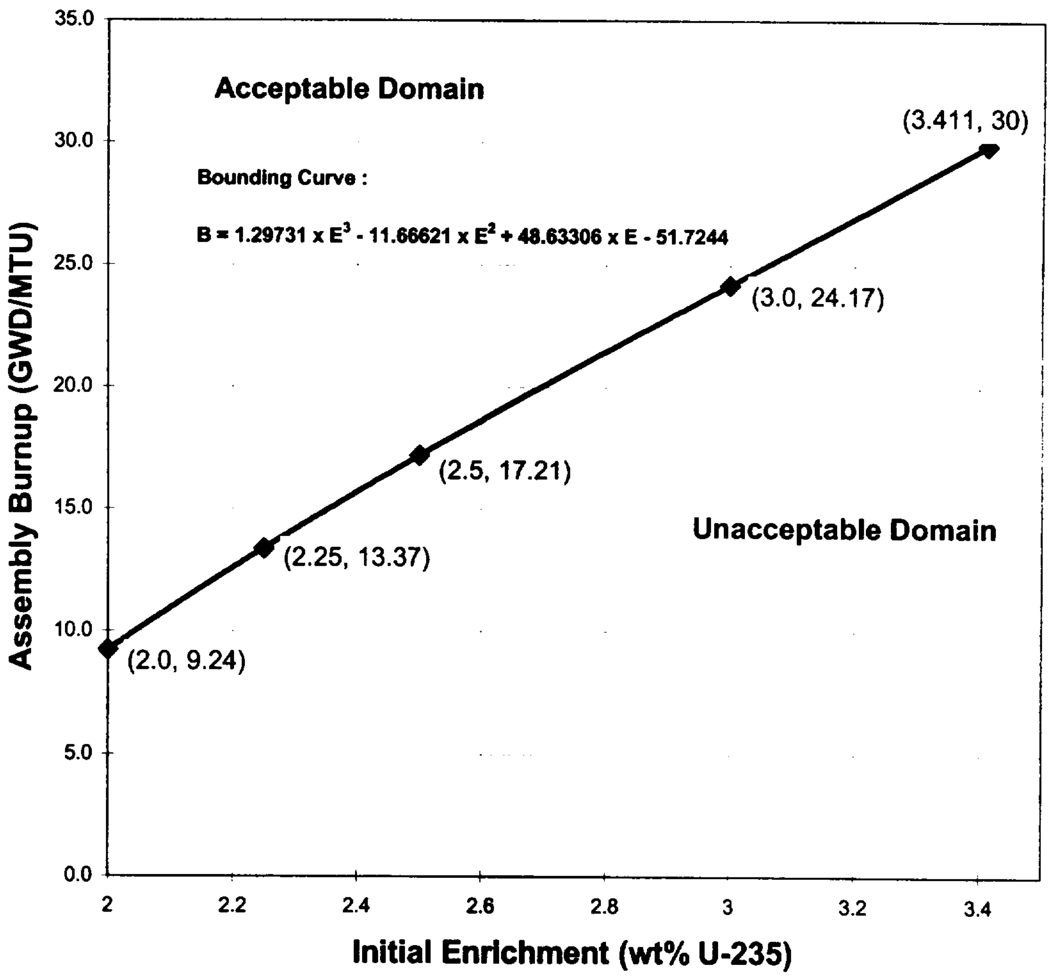
The spent fuel storage pool is designed with a storage capacity of 1205 assemblies and shall be limited to no more than 1205 fuel assemblies.

c. Canal Rack Storage

Fuel assemblies stored in the canal racks shall meet the minimum required fuel assembly burnup as a function of nominal initial enrichment as shown in Figure TS 5.4-1. These assemblies shall also have been discharged prior to or during the 1984 refueling outage.

FIGURE TS 5.4-1

MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF NOMINAL INITIAL ENRICHMENT TO PERMIT STORAGE IN THE TRANSFER CANAL



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

Wisconsin Public Service Corporation (WPSC) submitted a letter dated November 18, 1999, and Nuclear Management Company, LLC (NMC) submitted a letter dated August 7, 2000, requesting an amendment to the Technical Specifications (TSs) for the Kewaunee Nuclear Power Plant (KNPP). The amendment proposes changes to the KNPP TSs to allow 215 spent fuel assemblies (SFAs) to be stored in the new north canal pool. Subsequently, WPSC was succeeded by NMC, as the licensed operator of the KNPP. By letter dated October 5, 2000, NMC (the licensee) requested the NRC staff to continue to process and disposition licensing actions previously docketed and requested by WPSC.

Brookhaven National Laboratory (BNL), in support of the Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) reviewed the proposed changes regarding the thermal hydraulics and the control and handling of heavy loads, and requested clarifying and additional information, which was submitted by the NRC to the licensee by letter dated May 23, 2000.

KNPP is a pressurized water reactor (PWR) which commenced commercial operation in 1974, and its current operating license will expire in December 2013. Initially, KNPP was designed to accommodate 168 SFAs. The last phase of re-racking the spent fuel pool (SFP) at KNPP was completed in 1987, which provided for the current storage capacity of 990 SFAs. The KNPP currently has two storage pools. The larger south pool contains racks with storage capacity for 720 SFAs, and the smaller north pool contains racks with a storage capacity for 270 SFAs. At present, there are 718 SFAs stored in the south pool and 106 SFAs stored in the north pool. As a result of the present unavailability of an off-site spent fuel storage facility and the current rate of fuel discharge (approximately 40 assemblies per cycle), the KNPP will lose full-core reserve capability after the fall 2001 outage. The addition of the 215 storage locations in the new north canal pool will extend the full-core reserve capability until after the 2009 outage, and increase the total capacity to 1,205 SFAs.

To provide for this new storage area, the licensee plans to construct a transversal wall in the existing fuel transfer canal to form the new north canal pool storage area. However, the amendment does not include the installation of this wall modification, which the licensee plans to construct in accordance with the requirements of 10 CFR 50.59, and which is not addressed in this Safety Evaluation Report (SER).

The proposed changes would expand the present spent fuel storage capability to allow the use of spent fuel racks in the new north canal pool. The KNPP spent fuel storage pool is divided into two storage compartments and a fuel transfer canal. The north pool is designed for the transfer of SFAs into shipping casks. The fuel transfer canal and the two storage areas are interconnected by fuel transfer slots, which can be closed off by pneumatically sealed gates. Both the fuel transfer canal and the spent fuel storage pools are Class I reinforced concrete structures with seam-welded stainless steel plate liners. The licensee will be constructing a transversal wall, which will divide the fuel transfer canal into a north and south canal. This wall will be located north of the existing south pool/canal access gate. The licensee proposes to install four new spent fuel storage rack modules, with a capacity of 215 SFAs, in the newly created area in the transfer canal. The new racks will contain Boral as the active neutron absorbing poison, and will allow for SFA storage with a maximum initial enrichment of up to 3.411 wt% U-235 with a minimum burnup requirement of 30 GWD/MTU. The fuel to be stored in the north canal pool will be the fuel discharged during the 1976-1984 outages, which are also the oldest and coolest SFAs.

To accommodate this proposed modification, the following three Technical Specification changes have been proposed:

(1) TS 3.8.a.3, "Refueling Operations" and related Basis would be modified to increase the time the reactor will be sub-critical prior to movement of irradiated fuel assemblies (in the reactor core) to 148 hours (from the previous specified 100 hours).

(2) TS 5.4.b, "Fuel Storage-Capacity" would be added to include the 215 storage space capacity of the new racks in the spent fuel transfer canal. This increases the total SFP capacity to 1,205 assemblies.

(3) TS 5.4.c, "Fuel Storage-Canal Rack Storage" and Figure TS 5.4-1 would add the minimum required fuel assembly burnup as a function of nominal initial enrichment for the SFAs being stored in the spent fuel transfer canal.

Brookhaven National Laboratory's (BNL) evaluation of the licensee's submittal focused on the thermal hydraulic analyses and the control of heavy loads as described in the proposed modifications to Technical Specifications by KNPP TSs 3.8.a.3 and 5.4.b. The TS 5.4.c and Figure 5.4-1 deals with criticality considerations, and was reviewed by the NRC staff. In addition, the NRC staff reviewed the following aspects of the licensee's submittal: structural integrity and adequacy, occupational radiation exposure, radioactive waste, and fuel handling accidents.

This Safety Evaluation Report (SER) presents the results of the review of the amendment in the areas of the safe handling of heavy loads, thermal hydraulics, criticality, structural integrity and adequacy, occupational radiation exposure, radioactive waste, and fuel handling accidents.

2.0 EVALUATION

2.1 Safe Handling of Heavy Loads

2.1.A Background

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides guidelines and recommendations to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over stored spent fuel assemblies, fuel in the reactor core, safety-related equipment, and equipment needed for decay heat removal. The NUREG defines a heavy load as any load carried in a given area during the operation of the plant that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool.

Phase I of NUREG-0612 provided guidelines for reducing the likelihood of dropping heavy loads and limiting the resulting potential consequence of a drop. The guidelines are focused on establishing safe load paths, procedures for load handling operations, training of crane operators, the design of lifting devices and the design, testing, inspection, and maintenance of cranes. Phase II of NUREG-0612 provided guidelines for mitigating the consequences of dropped loads, including the use of a single-failure proof crane, use of electrical interlocks and mechanical stops to restrict crane travel, or performance of load drop and consequence analyses to assess the impact of dropped loads on plant safety. Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 25, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety. Based upon specific instances of heavy load handling concerns, the NRC requested licensees, in NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor, or Over Safety-Related Equipment," to provide specific information detailing their extent of compliance, and how, with these guidelines. In response to this request, the licensee stated by letter dated May 17, 1996, that they were in compliance with the provisions for both Phases I and II of GL 85-11.

2.1.B Hoisting System Evaluation

NUREG-0612 recommends that when licensees handle heavy loads in the proximity of safe shutdown equipment or irradiated fuel in the spent fuel pool, specific actions be implemented to minimize the potential for an accidental drop. These actions include the use of cranes and special lifting devices which are inspected, tested, and maintained to specific guidelines; the development of specific procedures to cover the load handling operations; and the use of trained and qualified crane operators and other personnel.

The licensee states that the four new spent fuel rack modules will be delivered in the horizontal position. They will be removed from the shipping trailer in the horizontal position using two chain hoists and a spreader beam suspended from the fuel handling crane. As stated by the licensee, the fuel handling crane has been designed, fabricated, and qualified in accordance with the guidelines of Section 5.1.1(6) of NUREG-0612, the Electric Overhead Crane Institute Standard No. 61, and Chapter 2-2 the American National Standard Institute Standard B-30.2.0, 1976. The rated load on the crane main hook and cable is 125 tons. As stated, each of the two hoists will be rated for 65 percent (minimum) of the total lifted weight to account for unequal loading and dynamic load factors. The fuel handling crane will be used to lift the upending frame off the ground, and the chain hoists will be used to rotate the racks to the vertical position. The new racks will then be raised to the refueling floor using a combination of a specially designed, remotely engageable lift rig and the fuel handling crane.

The licensee states that the lifting rig is similar to the design used to re-rack numerous other plants. The rig complies with all the provisions of Section 5.1.6 of NUREG-0612 and American National Standards Institute (ANSI) 14.6-1978. The rig has redundancies in the lift legs and lift eyes such that there are four independent load members. Failure of any one load-bearing member would not result in the uncontrolled lowering of a rack module.

The maximum combined weight for this operation will be less than 4 tons (6810 lbs. (nominal) for the 5 X 11 rack module plus 1000 lbs. (max) for the lift rig). These modules are considerably lighter than the previously installed 9 X 10 modules, which weigh approximately 42,000 lbs. Because the new racks are lighter than the previously installed racks, the licensee states that an accidental drop of these proposed racks does not present a new or different type of event and is bounded by the previous reracking effort. The KNPP proposed change involves the installation of racks that will permit closer spacing of SFAs, and will not utilize any new or unproven technology.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintain safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee plans to implement measures using administrative controls and procedures in each of these areas. The licensee states that only crane operators who have been trained and qualified in accordance with the KNPP crane training program and Chapter 2-3 of ANSI B30.2-1976 will be utilized. Additional licensee personnel who will provide hand signals to the crane operator during rack movement will also have completed the KNPP Control of Heavy Loads training. As previously discussed, the specially-designed redundant lifting rig complies with all the provisions of ANSI 14.6-1978. The spent fuel crane is tested, maintained and inspected in a manner which satisfies Chapter 2-2 of ANSI B30.3-1976. KNPP procedures require the performance of a pre-use crane inspection of hook latches, the crane hook, and wire hoist rope. As stated by the licensee, this procedure also incorporates a functional check of the crane controller prior to crane use. The KNPP program was previously reviewed by the NRC in 1984, and found to be consistent with the guidelines of NUREG-0612.

Based upon this evaluation, the NRC staff believes that the fuel handling crane, coupled with the lifting rig and other lifting devices, will enable the licensee to handle the heavy loads with little or no risk to the safety of the proposed reracking operation. Additionally, the NRC staff believes that the licensee's proposed personnel training, equipment inspections and functional checks, use of redundant lift rigs, and procedural controls provides adequate defense-in-depth to maintain safety during the handling of the new spent fuel racks.

2.1.C Safe Load Paths and Load Handling Accident Analysis Evaluation

In addition to the guidelines discussed above, NUREG-0612 also discusses the identification of safe load paths for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the SFP, or to impact safe shutdown equipment. The licensee states that the load paths to be used for the installation of the new racks will be defined to minimize the potential of impact if dropped on irradiated fuel and safe shutdown equipment. KNPP Technical Specification 3.8.a.8 prohibits movement of heavy loads over either spent fuel pool. Placement of additional storage racks is permitted by KNPP Technical Specifications, however, these racks must not traverse directly above stored spent fuel. The licensee has stated that electrical interlocks have been installed to prevent any inadvertent travel over either pool. Specific procedures, which will cover the entire modification effort, will be developed and implemented. The route which will be used in moving the new rack modules will not traverse any of the pools, and will be reviewed and approved by the Plant Operations Review Committee (PORC) prior to use. No spent fuel is stored in the area where the new racks will be installed. For rack movements along the pool floor, the height of the racks above the liner will not exceed six inches, except in the case of floor obstructions.

The proposed installation of the racks in the new north canal pool is bounded by the previous reracking effort when significantly larger and heavier rack modules were installed. Because of this, the licensee states that the proposed installation of the new racks does not represent a new or different kind of accident that was not previously analyzed. The licensee states that the installation of these racks will not traverse safety-related equipment, the existing spent fuel pools, or stored spent fuel. Because of this, no previously unanalyzed event that would result in a fuel configuration change, fuel release, or compromise of the pool structure leading to the loss of the coolant is postulated to occur.

The NRC staff concurs that the proposed installation of these racks will result in little or no risk to the safety of the proposed reracking operation, and does not represent any new or unanalyzed accident scenario that could result in damage to the pool structure or stored spent fuel. In addition, the NRC staff believes that, through the implementation of electrical interlocks and procedural controls, the safe load paths to be used during the installation of the new spent fuel racks will adequately minimize the potential for damage to the stored spent fuel and safety-related equipment.

2.1.D Fuel Handling Considerations Evaluation

The licensee has evaluated the potential of an accident involving the dropping of a spent fuel assembly associated with this proposed modification. As stated by the licensee, the proposed addition of the new spent fuel modules in the north canal pool will involve no additional spent fuel movement. These new racks will be used to store spent fuel that was discharged during the 1976-1984 outages. The licensee further states that the fuel stored in these racks will remain there until it is ultimately removed from the pool. No additional fuel shuffling within the pool is necessary to accommodate the installation of the new spent fuel (SF) racks. The licensee states that the KNPP fuel handling system will have sufficient capability to access all the cells of the proposed new racks. The method of handling the spent fuel during the loading of the proposed new racks will be the same as the current fuel handling methods, use the same fuel handling equipment, and the same procedures.

The licensee has evaluated the amount of spent fuel movement required to load the proposed new racks once installed, and concluded that it will be insignificant in comparison to the fuel movement that has occurred since 1974, as well as the total fuel movement planned over the remaining licensed operating period. In the event the proposed racks were not installed, the licensee would have to take alternate actions to provide sufficient spent fuel storage once the current capacity of the SFP was attained. These actions (e.g., use of spent fuel casks) would require an equivalent amount of spent fuel handling to load the casks plus the additional heavy load handling associated with the casks.

The NRC staff concurs with the licensee's evaluation that the increase in spent fuel handling associated with the installation of the proposed new racks is insignificant when compared to the spent fuel handling which has occurred and is planned to occur. This will not have a significant effect on the probability of an accidental fuel drop. In addition, the new spent fuel racks will preclude the need for additional heavy loads associated with spent fuel casks which might have been required to provide adequate on-site spent fuel storage.

2.1.E Summary of Heavy Load Considerations

Based upon the preceding discussions, the NRC staff concludes that the control of heavy loads aspects associated with the proposed changes to the TSs to allow for the additional storage of spent fuel in the new north canal pool area is in accordance with NUREG-0612, GL 85-11 and NRC Bulletin 96-02. Compliance with the specified administrative controls and procedures will result in the safe handling of heavy loads associated with the placement of the proposed new spent fuel storage racks. These changes will enable the licensee to increase the spent fuel storage capacity while not increasing the likelihood of damage to existing stored spent fuel and the pool structures.

2.2 Thermal-Hydraulic Considerations

2.2.A Background

NUREG-0800, "Standard Review Plan" provides criteria related to the design and performance of the spent fuel pool. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" provides methods acceptable for the licensee to implement General Design Criteria 61 of Appendix A to 10 CFR Part 50, which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. The NRC memorandum entitled "Office Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, and modified by Addendum dated January 18, 1979, provides key design criteria and regulatory guidance for new spent fuel storage racks.

2.2.B Spent Fuel Pool Cooling System Evaluation

The SFP cooling system (SFPCS) transfers decay heat from spent fuel stored in the SFP to the service water (SW) system. The SFPCS has two pumps and a heat exchanger. The SFPCS heat exchanger is a shell and tube unit; the cold shell side flow is supplied from the SW system and the hot tube side water is from the SFPs. During full-core offloads, following the completion of fuel transfer, the heat removal capacity of the SFPCS may be increased by aligning a Residual Heat Removal (RHR) system heat exchanger in parallel with the SFPCS heat exchanger. In this mode of operation, both SFPCS pumps circulate water from the SFP through the SFPCS heat exchanger and the "A" RHR heat exchanger. The cold shell side flow of the RHR heat exchanger is supplied from the component cooling water (CCW) system, which is in turn cooled by the SW system.

During partial-core discharge, a single SFPCS pump and the SFPCS heat exchanger provide cooling for the SFPs. During full-core discharge, both SFPCS pumps and the SFPCS heat exchanger will provide cooling for the SFP until the completion of fuel transfer. After fuel transfer is completed, and if the SW temperature exceeds 60 °F, one RHR system heat exchanger will be aligned in parallel with the SFPCS heat exchanger to cool the SFPs. The licensee stated in the submittal that the Reactor Engineering procedure that controls fuel movement during refueling outage would be revised to ensure the RHR alignment.

The heat removal capacities of heat exchangers depend on the flow rates and temperatures of the shell and tube side water. The following table shows the heat removal capabilities of the SFPCS and RHR system in the SFP cooling mode for various configurations, as functions of the SFP temperature; the cold shell side water inlet temperatures are assumed to be 80 °F for the SFPCS heat exchanger (SW), and 88 °F for the RHR heat exchanger (CCW).

System Configuration	SFP Temperature (°F)	Heat Removal Capability (10^6 Btu/hr)
1 SFPCS pump and 1 SFPCS heat exchanger	100 125 150	3.10 7.05 11.06
2 SFPCS pumps and 1 SFPCS heat exchanger	100 125 150	3.86 8.78 13.77
2 SFPCS pumps, 1 SFPCS heat exchanger and 1 RHR heat exchanger	100 125 150	4.84 12.12 19.50

Since the proposed increase in the SFP storage capacity would result in the increase of SFP heat load for all discharge scenarios, the licensee reevaluated the effects of the increased SFP storage capacity on the SFP heat loads and temperatures.

Three discharge scenarios were postulated for the bulk pool thermal-hydraulic evaluation:

(1) A normal partial-core discharge (48 fuel assemblies) 132 hours after a reactor shutdown.

(2) An emergency full-core discharge (121 fuel assemblies) 30 days after a normal partial core discharge.

(3) A planned full-core discharge (121 fuel assemblies) 148 hours after a reactor shutdown.

The licensee calculated the decay heat load using Holtec's QA validated LONGOR program, which incorporated the ORIGEN2 computer code for decay heat calculations. To determine the bounding cases for maximum decay heat calculations, the licensee made the following conservative assumptions regarding reactor thermal power levels, burnup levels, and number of SFAs stored in the SFP:

1. All reactor thermal power levels are increased by 2 percent to account for the plant's reactor thermal power calorimetric uncertainty.

2. A reactor thermal power uprate (about 4.3 percent higher) is assumed for all projected (i.e., after 1999) discharge batches.

3. Expected bounding parameters (i.e., burnup, batch size, initial enrichment, etc.) are used for all projected discharges.

4. The total fuel inventories stored in the SFPs are assumed to slightly exceed the 1205 maximum storage locations.

5. For a planned full-core discharge, the assemblies in the core are split into three regions with burnup levels corresponding to once-, twice-, and thrice-burned. The thrice-burned and twice-burned regions are each assumed to be the size of the maximum refueling batch size, resulting in the maximum number of assemblies having the highest possible burnups.

6. For an emergency full-core discharge, the assemblies in the core are split into three regions with burnup levels corresponding to 30 days at power, once-burned plus 30 days at power, and twice-burned plus 30 days at power. The twice-burned plus 30 days and the once-burned plus 30 days regions are each assumed to be the size of the maximum refueling batch size, resulting in the maximum number of assemblies having the highest possible burnups.

7. For an emergency full-core discharge, the refueling outage immediately before the core offload is assumed to be zero days long. Thus, the two reactor shutdowns are separated by exactly 30 days and the second shutdown occurs after 30 days of operation.

BNL concurs with the methodology and conservative assumptions the licensee used to calculate the decay heat loads.

The coincident net decay heat load for each scenario estimated by the licensee was:

Discharge Scenario	Coincident Net Decay Heat Load (10^6 Btu/hr)
A normal partial-core discharge	9.39
An emergency full-core discharge	18.43
A planned full-core discharge	18.62

In evaluating the maximum SFP bulk temperature, the licensee also made the following conservative assumptions regarding the heat exchanger fouling factors, tube plugging allowance, the coolant water inlet temperatures to the heat exchangers, and thermal capacities of other structures in the SFP.

a. For the bulk temperature analysis, the two spent fuel pools and the north end of the transfer canal are all assumed to be at the same temperature.

b. The thermal performance of the SFPCS and RHR System heat exchangers is determined with all heat transfer surfaces fouled to their design-basis maximum levels.

c. The thermal performance of the SFPCS and RHR System heat exchangers is determined incorporating a 10 percent tube plugging allowance.

d. Bounding maximum temperatures are used for the coolant water inlet temperatures for the SFPCS and CCW System heat exchangers.

e. The thermal inertia (thermal capacity) of the SFPs is based on the net water volume only. This conservatively neglects the thermal inertia of the fuel assemblies, stainless steel racks and stainless steel SFP liners.

The maximum pool bulk temperature for the normal partial-discharge scenario was calculated assuming only one SFPCS pump and one SFPCS heat exchanger were available, which is consistent with the single active failure recommendation of the SRP guidelines. Both of the full-core discharge scenarios assumed that 2 SFPCS pumps, 1 SFPCS heat exchanger and 1 RHR heat exchanger were available. The licensee solved the differential equations representing the transient heat balance and the thermal response of the SFP, using the Holtec QA validated computer program STER, to obtain the bulk pool temperature. This program utilizes the above data on heat removal capability of the heat exchangers, as well as the heat exchanger geometric data obtained from the manufacturers. The assumptions discussed above were also incorporated into the model. The following table shows the maximum pool bulk temperature calculated for each scenario:

Discharge Scenario	Maximum Pool Bulk Temperature (°F)
A normal partial-core discharge	139.53
An emergency full-core discharge	149.51
A planned full-core discharge	149.73

As shown in the above table, the maximum bulk temperatures of the SFP water remain below 150 °F for all scenarios. These results are also confirmed with the observation that the heat removal capability of the SFPCS exceeds the decay heat generated for Normal Partial Core Discharge scenario and the heat removal capability of the SFPCS and RHR in the SFP cooling mode exceeds the decay heat generated for both Full Core Discharge scenarios, when the SFP temperature is 150 °F. BNL concurs with the methodology and conservative assumptions the licensee used to calculate the SFP bulk temperatures.

Based on the NRC staff's review of the results and the methodology the licensee provided in the submittals, the NRC staff finds that the design and operation of the SFPCS meet the intent of the guidance described in SRP for the SFPs, provided that the additional Reactor Engineering procedure that controls fuel movement during refueling outage will be in place to ensure the RHR heat exchanger alignment for the full-core discharge scenarios. Further, the NRC staff finds that the SFP will be maintained below its design temperature of 150 °F for all three scenarios.

2.2.C Effect of SFP Boiling Evaluation

In the unlikely event that there is a complete loss of cooling, the SFP water temperature will begin to rise and eventually reach the boiling temperature.

The licensee performed an analysis to demonstrate the minimum time-to-boil and the maximum boil-off rate. The calculated minimum time from the loss-of-pool cooling at peak pool water temperature until the pool boils, based on the heat load for the full core offload, is 8.3 hours. The calculated maximum boil-off-rate is 41 gpm and the minimum time-to-minimum shielding depth (10 feet above the racks) is 48.5 hours. These results show that there would be at least 8.3 hours available for corrective actions prior to SFP boiling in the unlikely event of a failure of forced cooling to the SFP, and a minimum of 48.5 hours before the water boils to below the minimum shielding depth. The maximum boil-off rate of 41 gpm is much less than the emergency makeup capacity of 1000 gpm available from the service water system, which is a seismically qualified QA Category 1 system. The licensee stated that makeup from the service water system, which is initiated by opening one manual valve in the SFP heat exchanger room, can be performed in considerably less than 8.3 hours. Additionally, emergency makeup water can also be provided from the reactor makeup water or fire protection systems.

Based on this review, the NRC staff finds that, in the unlikely event that there is a complete loss of cooling, the licensee is capable of aligning the makeup water from various sources to the pool before boiling begins and that make-up water will be supplied at a rate that exceeds the boil-off rate. The NRC staff concludes that cooling the SFP at KNPP by adding makeup water during the unlikely event that there is a complete loss of SFP cooling provides adequate protection and conforms with the guidance described in the SRP and is, therefore, acceptable.

2.2.D Summary of Thermal-Hydraulic Considerations

The NRC staff has reviewed the licensee's submittal for compliance with guidelines and recommendations on SFP storage as provided in the Standard Review Plan (SRP), Regulatory Guide 1.13 and the Office Technical (OT) Position Paper, and has concluded that the thermal-hydraulic aspects of the proposed license amendment request are acceptable, provided that KNPP implements the administrative controls and procedures to ensure that the RHR heat exchanger is aligned for the full-core discharge scenarios.

2.3 Criticality

The proposed expansion would consist of installation of high density spent fuel storage racks in the north canal pool. The new racks were designed by Holtec International and are free-standing and self-supporting. The principal construction material is stainless steel. The only non-stainless material is the neutron absorber material which is the boron carbide and aluminum composite sandwich called Boral.

The primary analysis of the reactivity effects of fuel storage in the KNPP racks was performed with the MCNP4a continuous energy three-dimensional Monte Carlo code. Independent verification calculations were performed with the KENO5a three-dimensional multigroup Monte Carlo code. The CASMO-4 two-dimensional transport theory code was also used for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the KNPP spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, the two independent methods of analysis (MCNP4A and KENO5a) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. The NRC staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the proposed KNPP storage racks with a high degree of confidence.

General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50, requires the prevention of criticality in fuel handling and storage. PWR spent fuel pools contain soluble boron, which maintains the stored fuel assemblies approximately 25 percent or more subcritical during normal storage. However, for conservatism, the NRC acceptance criterion for subcriticality is that the effective multiplication factor (k-eff) in the spent fuel pool storage racks shall be no greater than 0.95, including uncertainties at a 95 percent probability, 95 percent confidence level (95/95), even under the extreme accident condition of a complete boron dilution event (i.e., flooded by unborated water). The criticality analyses were performed with several additional assumptions, which tend to maximize the rack reactivity. These include:

(1) Racks were fully loaded with the most reactive fuel authorized to be stored in the facility,

(2) Unborated pool water at the temperature yielding the highest reactivity (4 °C) over the expected range of water temperatures,

(3) Assumption of infinite array (no neutron leakage) of storage cells except for the assessment of peripheral effects and certain accident assessments,

(4) Neutron absorption in minor structural material is neglected (i.e., spacer grids are analytically replaced by water).

The NRC staff concludes that appropriately conservative assumptions were made.

The Westinghouse original design 14x14, Westinghouse Optimized Fuel Assembly (OFA) 14x14, and the Siemens Power Company 14x14 fuel assemblies were evaluated. The design basis fuel assembly used for the transfer canal rack criticality analyses was the Westinghouse original design 14x14 assembly since this was determined to have the highest reactivity for the proposed Holtec storage racks. For the nominal storage cell design, uncertainties due to manufacturing tolerances on boron loading, boral width, lattice spacing, stainless steel thickness, and fuel density and enrichment were included. In addition, a calculational uncertainty for burnup calculations and the effect of axial burnup distribution was included for burnup calculations. These uncertainties were appropriately determined at the 95/95 level, thus meeting the NRC acceptance criterion.

In order to store fuel with maximum initial enrichments up to 3.411 weight percent (w/o) U-235 in the transfer canal racks, the concept of burnup reactivity equivalencing was used. This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for use in PWR fuel storage analysis. A series of reactivity calculations is performed to generate a set of enrichment versus burnup ordered pairs which yield an equivalent k-eff less than 0.95 (approximately 0.945) for fuel stored in the storage racks. The results are illustrated in TS Figure 5.4.1, which indicate that fuel with an initial maximum nominal enrichment of 3.411 w/o U-235 must achieve a burnup of at least 30 gigawatt days per metric ton uranium (GWD/MTU) to be allowed storage in the proposed spent fuel racks in the transfer canal. Likewise, fuel initially enriched to 2.0 w/o U-235 or less must have accumulated a minimum burnup of 9.24 GWD/MTU.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events, such as the inadvertent placement of a fresh (unirradiated) fuel assembly into a location restricted to a burned assembly as per TS Figure 5.4.1, which could lead to an increase in reactivity. This event is highly unlikely because the transfer canal racks will be filled with old fuel assemblies following rack installation and the racks are not in close proximity to new fuel or in a transfer path for fuel. However, for such events, credit may be taken for the presence of soluble boron in the transfer canal water which is assured by KNPP TS 3.8.a.5, since the NRC staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). TS 3.8.a.5 requires that the minimum soluble boron concentration in the spent fuel pool and transfer canal be maintained at 2100 ppm and is confirmed by monthly surveillance measurements. The reduction in k-eff due to the boron more than offsets the reactivity addition caused by credible accidents. In fact, Holtec has determined that a soluble boron concentration of only 240 ppm would be sufficient to maintain k-eff less than 0.945 even if a fresh 5.0 w/o assembly were inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.4.1.

The following TS changes have been proposed and are acceptable based on the above safety evaluation:

- TS 5.4.b increases the total spent fuel pool storage capacity to 1205 assemblies.
- TS 5.4.c adds the minimum required fuel assembly burnup as a function of nominal initial enrichment for assemblies being stored in the spent fuel transfer canal.
- Figure 5.4-1 is added which specifies the minimum required fuel assembly burnup as a function of nominal initial enrichment for the fuel assemblies being stored in the transfer canal.

Based on the review described above, the NRC staff finds the criticality aspects of the proposed storage capacity expansion in the KNPP spent fuel pools transfer canal are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. Our review has also determined that the proposed changes to TS 5.4, "Fuel Storage," are acceptable.

2.4 Structural Integrity and Adequacy

This evaluation summarizes the results of the staff's review of the procedures and results of the structural analyses performed by the licensee to demonstrate the structural integrity of the four new high density spent fuel racks and the continued structural adequacy of the entire fuel handling area including the new wall in the transfer canal (TC). The design of the new partition wall was conducted under the postulated design loads (Appendix D of SRP Section 3.8.4) for normal, seismic, and accident conditions that are bounding for those load combinations mandated by the ACI Code 349-85 (Refs. 1 and 2).

2.4.A Spent Fuel Pool Storage Racks

SFP racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE) under all applicable loading conditions. The licensee's consultant, Holtec International, performed the design, fabrication, and safety analysis of the new high density SFP storage racks. The new spent fuel racks are made of ASME SA240-304L stainless steel. The non-stainless steel material utilized in the rack is the neutron absorber material called Boral which is composed of boron carbide and Alloy -1100 aluminum (Ref. 1, Section 3.1.1). The overall design of the new racks at KNPP is similar to Holtec racks that NRC has approved for service at many other nuclear power plants. The key design criteria are based on the US NRC memorandum titled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978, as modified by amendment dated January 18, 1979 (Ref. 3).

The key design criteria of the new Kewaunee SFP racks are described in Section 2.1 of Reference 1. Briefly, the following criteria are applicable from the structural safety point of view: (1) all rack modules are required to be free-standing; (2) all free-standing rack modules are required to be kinematically stable with a safety factor of at least 1.5 and 1.1 against overturning for load combinations including operating basis earthquake (OBE) and SSE seismic events, respectively; (3) all primary stresses in the rack modules must satisfy the limits postulated in Section III, Subsection NF of the 1989 ASME Boiler and Pressure Vessel Code; (4) the spatial average bulk pool temperature is required to remain under 140 °F following a normal refueling discharge (assuming a single active failure), and under 150 °F following a full-core discharge; and (5) the ability of the reinforced concrete structure of the SFP to withstand the effects of the load combinations set forth in the plant Updated Safety Analysis Report (USAR) must be demonstrated.

At the time of the original rack installation in the SFP at KNPP, the seismic evaluation of the racks was performed using single-rack (SR) three-dimensional (3-D) simulations. However, for the current SFP expansion, whole pool multi-rack (WPMR) analysis was performed to simulate the dynamic behavior of the high density rack structures (Ref. 1). Holtec used a computer program, DYNARACK, for the dynamic analysis to demonstrate the structural adequacy of the spent fuel rack design under the earthquake loading conditions. The DYNARACK program (which can perform simultaneous simulation of all racks in the pool for the WPMR analysis) has been accepted by the NRC in previous re-rack analyses for several nuclear power plants. The DYNARACK program utilizes a nonlinear analytical model consisting of inertial mass elements, spring elements, gap elements and friction elements to simulate the three-dimensional dynamic behavior of the rack and the fuel assemblies including the frictional and hydrodynamic effects (Ref. 1). The DYNARACK computer code simulates the friction, impact, and other nonlinear dynamic events accurately. The Code models the beam characteristics of the rack including shear, flexibility, and torsion effects appropriately, by modeling each rack as a three-dimensional structure having the support pedestals and the fuel assemblies in proper locations.

The potential rattling between the fuel and storage cells is simulated by permitting the impact at any of the four facing walls followed by rebound and impact at the opposite wall. Further, the rack pedestals can lift off, or slide, to satisfy the instantaneous dynamic equilibrium of the system throughout the seismic event. The rack structure can undergo overturning, bending, twist, and other dynamic motion modes as dictated by the interaction between the seismic inertia, impact, friction, and fluid coupling forces. The DYNARACK Code calculates the nodal forces and displacements at the nodes, and then obtains the detailed stress field in the rack elements from the calculated nodal forces.

The lateral motion of the rack due to earthquake ground motion is resisted by the pedestal-to-pool slab interface friction, and is amplified or retarded by the fluid coupling forces produced by the close position of the rack to other structures. The seismic analyses of the racks were performed utilizing the direct integration time-history method. One set of three artificial time-histories (two horizontal and one vertical acceleration time-histories) was generated in accordance with the provisions of SRP 3.7.1. A preferred criterion for the time-history generation given in SRP 3.7.1 calls for both the response spectrum and the power spectral density (PSD) corresponding to the generated acceleration time-history to envelope their target (design-basis) counterparts with only finite enveloping infractions. The licensee did not generate time histories for the OBE event, but evaluated the racks conservatively using the stresses produced by the SSE event and comparing them against the normal condition allowable values (Ref. 1, Section 6.4). In response to an NRC staff question, the licensee stated that the target (design basis) response spectra were the original licensing basis in-structure response spectra (ISRS) described in the KNPP USAR (Ref. 2). The licensee generated the time-histories to satisfy the preferred criterion stated above. This procedure is acceptable to the staff.

The licensee considered the applicable loads and their combinations in the seismic analysis of the new rack modules, and performed six parametric simulations for WPMR analyses for the SSE only (Ref. 1, Section 6.7). The parameters, which were varied in the six computer runs, consisted of the rack/pool interface coefficient of friction and the extent of storage locations occupied by spent fuel (ranging from half loaded to fully loaded). The results of these analyses (discussed in Ref. 1, Section 6.8) show the maximum rack displacement to be 0.317 inches (for a fully loaded WPMR analysis under the SSE condition). In response to an NRC staff question, the licensee stated that, for this case, a rack overturning evaluation indicated the factor of safety against overturning to be 45, which is much higher than the prescribed limit of 1.1 for the SSE condition (Ref. 2). This shows that there are large safety margins against overturning of the racks and the structural integrity and stability of the racks and fuel assemblies are maintained.

From the results of the parametric evaluations, the licensee computed the maximum values of pedestal vertical forces, pedestal friction forces (i.e., horizontal loads), pedestal thread shear stresses, rack displacements and rack stress factors (Ref. 1, Sections 6.8 and 6.9). Using these data, the licensee performed the rack impact evaluation, as well as the stress limit evaluation of the rack structure satisfying the ASME Code, Section III, Subsection NF, for normal and upset conditions (Level A or Level B), and Section F-1334 (ASME Section III, Appendix F) for Level D condition. The calculated results show that there are no impacts between racks or between racks and the surrounding TC walls (Ref. 1, Section 6.8.4). However, the parametric simulation showed that there was an instantaneous impact between fuel assemblies and fuel cell walls, with a maximum fuel/cell wall impact load of 230 lbf which is much less than the limiting impact load of 21,483 lbs. The nominal safety factor against fuel failure due to fuel-to-storage cell rattling force is greater than 90 (Ref. 1, Section 6.8.4.1).

The licensee calculated the weld stresses of the rack at the connections (e.g., baseplate-to-cell welds, and baseplate-to-pedestal welds, cell-to-cell welds) under the SSE and OBE loading conditions, and found that all the calculated weld stresses are well below the corresponding allowable stresses specified in the ASME Code Section III, Subsection NF, indicating that the weld connection design of the rack is adequate (Ref. 1, Section 6.9.4).

In summary, the licensee's parametric study described above showed that (1) all stresses are well below their corresponding "NF" limits, (2) there are no rack-to-wall or rack-to-rack impacts, and that (3) the rack overturning is not a concern (Ref. 1). Therefore, the NRC staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and are, therefore, acceptable.

2.4.B Spent Fuel Pool Structure

The spent fuel at KNPP is stored in two storage pools and a TC located in a seismic Category I, reinforced concrete (RC) structure (Ref. 1, Section 8.2). The pools are completely lined with a stainless steel liner. The top of the stainless steel liner is anchored to the pool slab. To demonstrate the continued structural adequacy of the SFP structure, the licensee analyzed the fuel handling area (FHA) using the finite element computer program, ANSYS, and combined the results for individual load components using factored load combinations per SRP 3.8.4. Tables 8.7 and 8.8 in Reference 1 summarize the safety factors for the RC members of the SFP upper structure and the basement shear walls, respectively, that are affected by the construction of the new TC wall and installation of the new high-density racks. The safety factors for the accident load condition of the upper structure range from 1.03 to 6.05 for bending, 1.04 to 2.90 for shear, and 1.08 to 7.99 for axial force (Ref. 1). The corresponding safety factors for the operating load condition for the upper structure are higher than those for the accident condition. The safety factors for the basement shear walls are considerably higher than those for the upper structure members stated above. In response to an NRC staff question related to the design of the new dividing wall in the TC, the licensee stated (Ref. 2) that the design loads considered for this wall were: dead loads, seismic loads, and normal operating and accident temperature loads. The safety factors for the new TC wall for the normal operating load conditions ranged from 1.58 to 30.26, and those for the extreme environmental (accident) load conditions ranged from 1.08 to 11.14 (Ref. 2). These safety factors representing the ratio between the ultimate capacities of the cross-section and the computed internal forces are greater than 1.0, thus demonstrating the structural integrity of the SFP under the increased loads due to the additional racks, and are acceptable. The licensee also demonstrated (Ref. 1, Table 5.8.1) that the maximum bulk pool temperature will not exceed the allowable value of 140 °F following a normal partial discharge, and 150 °F following a full-core discharge (per ACI Code 349).

The NRC staff has reviewed the licensee's analytical procedures and the summary of the results, and concluded that the licensee's structural analysis demonstrates the structural integrity of the SFP structure under full fuel loading and SSE loading conditions. Therefore, the SFP design is acceptable.

2.4.C Fuel Handling Accidents

The following fuel handling accident cases were evaluated by the licensee (Ref. 1): (1) one case for the drop of a fuel assembly (with its handling tool) impacting the top of a rack ("shallow drop" scenario), and (2) two cases for the drop of a fuel assembly (with its handling tool) falling through an empty storage cell and impacting the rack baseplate ("deep drop" scenarios). The "shallow drop" event (analyzed by finite element method) produces localized plastic deformation of the top of the impacted region, but the maximum depth of this plastic deformation is limited to 11 inches, which is below the design limit of 18.5 inches (Ref. 1, Section 7.5.1).

The "deep drop" event scenario, located above the support leg, produces a maximum stress of 25.32 ksi in the liner which is less than the failure limit of 66.2 ksi for the stainless steel liner, thus resulting in no damage to the liner (Ref. 1, Section 7.5.2 and Ref. 2). In response to a NRC staff question, the licensee stated that the maximum compressive stress of 15.75 ksi in the concrete slab (as shown in Fig. 7.54 in the KNPP USAR) is less than the failure limit of 20.22 ksi for the concrete (Ref. 2).

The "deep drop" scenario in which the fuel assembly impacts the baseplate away from the support pedestal produces some deformation of the baseplate and localized severing of the baseplate/cell welds (Ref. 1, Section 7.5.2). However, the licensee's analysis indicates that this "deep drop" event lowers the fuel assembly surface by a maximum of 1.068 inches, which is less than the distance of 6 inches from the baseplate to the liner. In response to a NRC staff question regarding the localized severing of the baseplate/cell wall welds, the licensee stated that its finite element analysis indicated that the effect of the baseplate deformation is confined to the impacted cell and cells that are directly adjacent. Therefore, the NRC staff agrees with the licensee's conclusion that such localized severing of the baseplate/cell wall welds will not lead to adverse hydraulic and critical consequences and will not degrade the overall structural integrity of the rack module (Ref. 2).

The NRC staff reviewed the licensee's fuel drop analysis results in References 1 and 2, and concurs with its findings that the postulated fuel drop accident events produce only localized damage within the design limits for the racks.

2.4.D Summary

Based on the review and evaluation of the licensee's November 18, 1999, submittal (Ref. 1), and its August 7, 2000, responses (Ref. 2) to the staff's request for additional information, the NRC staff concludes that the structural analyses of the SFP structure and the storage rack modules are in compliance with the acceptance criteria set forth in the USAR and are consistent with the current licensing basis.

2.5 Occupational Radiation Exposure

The NRC staff has reviewed the licensee's plan for the creation of an additional storage area and the installation of new racks in the Kewaunee Nuclear Power Plant SFP with respect to occupational radiation exposure. As stated above, the new storage capacity increase will be accomplished by installing four high density storage racks in a newly created pool resulting from the partitioning of the existing fuel transfer canal with a newly built concrete wall. A number of facilities have performed similar operations in the past. The licensee has estimated that the total occupational exposure for rack installation operations will be less than 1.3 person-rem. This represents a very small fraction (about 0.02 percent) of the total estimated person-rem burden from occupational exposure at the plant during its lifetime.

The estimated dose (0.7 to 1.3 person-rem) to increase the SFP storage capacity at Kewaunee is much lower than the doses (6 to 12 person-rem) required to perform similar modifications in the past at other nuclear power plants in the U.S. The dose at Kewaunee will be lower because the operation will not involve the removal of existing racks, and the rack installation will be performed in the transfer canal, which will have been drained, so that no underwater diving operations in the pool area where fuel assemblies are being stored will be necessary. The occupational exposure projected for the Kewaunee rack installation is based on current actual radiation levels at the SFP deck elevation as well as dose data collected for work performed under the radiation work permit associated with the most recent fuel transfer canal decontamination.

The licensee has calculated the gamma-ray dose rates in areas adjacent to and below the transfer canal, from fuel stored in the canal (oldest fuel currently stored in the main pool, discharged in 1984 or before). The dose rates at the outside surfaces of the north and east walls of the transfer canal are less than 0.0001 mrem/hr and the dose rate through the bottom of the transfer canal is less than 0.000001 mrem/hr. The dose rate at the southern surface of the new partition wall that divides the canal, is 7.88 mrem/hr. This area is typically flooded and therefore unoccupied. Personnel access to this area will be for infrequent maintenance of the fuel transfer equipment. Consequently, the increase in storage capacity in the Kewaunee SFP will not necessitate any radiation zoning changes to any of the surrounding areas.

The increment in the annual onsite occupational dose resulting from the proposed increase in stored fuel assemblies represents a negligible burden due to the fact that the assemblies that will be stored in the newly created storage area were discharged from the reactor core in 1984 or earlier and thus have undergone significant radioactive decay. The very small increase in radiation exposure (estimated by the NRC staff to be less than 0.5 percent to the total annual occupational radiation exposure at the Kewaunee facility) will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable (ALARA) and within the limits of 10 CFR Part 20.

The operating procedures planned for the Kewaunee rack installation are similar to procedures that were implemented successfully for other projects of fuel rack installation by HOLTEC International in the U.S. These procedures provide detailed requirements to assure equipment, personnel, and plant safety, and assure that ALARA practices are followed. The licensee's amendment request supporting documentation lists nine procedures that will be used to implement the rack installation phase of the project. One of these procedures is the "ALARA Procedure". Consistent with both HOLTEC International's and Kewaunee's ALARA programs, this procedure provides details to minimize the total person-rem received during the rack installation project, by accounting for time, distance, and shielding. Additionally, pre-job briefings will be performed in order to mitigate the potential for overexposures.

Work, personnel traffic, and the movement of equipment will be monitored and controlled to assure that exposures are maintained ALARA. Installation of the new fuel racks in the transfer canal will be implemented in accordance with the existing radiation protection program at

Kewaunee. Personnel involved in this activity will wear protective clothing, activities will be governed by the use of Radiation Work Permits, and personnel monitoring equipment will be issued to each individual (TLD badges and electronic dosimeters will be used by personnel working on the project). Radiation protection personnel will be providing constant coverage, including dose monitoring, for the majority of the work.

Each member of the project will be properly trained and will be provided appropriate education and understanding of critical job evolutions. Additionally, daily pre-job briefings will be employed to acquaint each team member with the scope of work to be performed and the proper means of executing such tasks. The use of divers is not planned for any activity associated with this project.

On the basis of our review of the Kewaunee proposal, the NRC staff concludes that the spent fuel rack installation operations at Kewaunee can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The NRC staff finds that the projected dose for the project of less than 1.3 person-rem is significantly lower than projected dose for similar operations authorized in the past at other facilities and is, therefore, acceptable. In addition, the NRC staff considers that the increment in the annual onsite occupational dose resulting from the proposed increase in stored fuel assemblies represents a negligible burden and will not affect the licensee's ability to maintain individual occupational doses as low as is reasonably achievable and within the limits of 10 CFR Part 20.

The NRC staff evaluation supports the conclusion that the proposed modification to the Kewaunee SFP is acceptable, because the increase in occupational exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

2.6 Solid Radioactive Waste

The NRC staff has reviewed the licensee's plan for the creation of an additional fuel storage area and the installation of new fuel racks in the Kewaunee Nuclear Power Plant SFP with respect to radioactive waste.

Radioactive waste generated from the rack installation effort may include vacuum filter bags, miscellaneous tooling, and protective clothing. Vacuum filter bags will be removed and stored as appropriate in a suitable container in order to maintain low dose rates. All vacuum cleaners will be equipped with high efficiency particulate absorber (HEPA) filters. In addition, trained personnel will be used to change the filter bags and monitor the radiation levels of the vacuum cleaners during use. The filters will be disposed of as normal low level radioactive waste. Contaminated tooling will be properly stored per Radiation Protection direction throughout the project. At project completion, an effort will be made to decontaminate tooling to the most practical extent possible.

With regard to resins generation, a very small amount of additional resins may be generated by the pool clean-up system on a one-time basis following installation of the new fuel racks. However, the licensee does not expect the resin changeout frequency of the SFP purification system to be permanently increased as a result of the storage of additional spent fuel assemblies in the SFP.

Overall, however, increasing the storage capacity of the SFP will not result in a significant change in the generation of solid radwaste at the Kewaunee Nuclear Power Plant (it is expected that the total volume of low level radioactive waste generated due to this project will be less than 50 cubic feet).

2.7 Fuel Handling Accident

Section 14.2.1 of the Kewaunee Nuclear Power Plant Updated Final Safety Analysis Report (UFSAR) presents the licensee's evaluation of the possible consequences of a FHA. The proposed expansion of the SFP will not affect any of the assumptions or inputs used in evaluating the dose consequence of the FHA.

The analysis applies to both the FHA in the reactor building and the FHA in the spent fuel pool and assumes breakage of all rods in the highest rated spent fuel assembly 100 hours after reactor shutdown, with no credit for holdup or filters in either the reactor building or the fuel-handling building. The resulting offsite dose consequences for the design basis fuel-handling accident are well within 10 CFR Part 100 guidelines.

The proposed modification increases the storage capacity but does not change the frequency or method for handling fuel assemblies. The NRC staff finds that the scenario for the postulated accident in the fuel-handling building does not change due to the use of high-density storage racks in the proposed fuel transfer canal storage area. No change is being made to the handling of the spent fuel or the types of fuel stored in the Kewaunee SFP, nor to the number of fuel assemblies being moved at any one time. Therefore, the inputs and assumptions for the dose consequences analysis do not change, and the current FHA dose analysis in the KNPP USAR remains bounding.

The current Kewaunee design basis FHA dose analysis, as described in the plant's USAR, remains bounding for the installation of high-density spent fuel racks in the fuel transfer canal storage area at Kewaunee Nuclear Power Plant. The staff finds the licensee's calculated radiological consequences of an FHA in the spent fuel pool, as shown in USAR Section 14.2.1, are well within 10 CFR Part 100 guidelines. Therefore, the staff finds the proposed expansion of spent fuel storage capacity at Kewaunee to be acceptable with regard to potential offsite radiological consequences of an FHA in the spent fuel pool.

The NRC staff evaluation supports the conclusion that the proposed modification to the Kewaunee SFP is acceptable, because the installation and use of the new fuel racks does not alter the consequences of an FHA for the SFP.

2.8 Summary

KNPP has proposed revisions to the Technical Specifications to permit the use of the north canal to be used to store spent fuel. New spent fuel storage racks will be installed in this area to provide storage for an additional 215 spent fuel assemblies. The proposed revisions to the Technical Specifications discussed in proposed amendment include:

(1) TS 3.8.a.3, "Refueling Operations,"

(2) TS 5.4.b, "Fuel Storage-Capacity," and

(3) TS 5.4.c, "Fuel Storage-Canal Rack Storage" and associated Figure TS 5.4-1.

Based upon the evaluation and results covering the areas of the control of heavy loads and thermal-hydraulics, the NRC staff concludes that the proposed revisions to KNPP Technical Specifications TS 3.8.a.3 "Refueling Operations" and TS 5.4.b "Fuel Storage-Capacity" comply with all applicable regulatory documents (i.e, NUREG-0612 and -0800, GL 85-11, NRC Bulletin 96-02, Regulatory Guide 1.13 and Office Technical Positions). This will allow for the safe handling and continued safe storage of spent fuel. These proposed revisions will allow the licensee to increase the spent fuel storage capacity and not result in plant damage during the installation and fuel storage periods.

Based on the review described above, the NRC staff finds the criticality aspects of the proposed storage capacity expansion in the KNPP spent fuel pools transfer canal are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. Our review has also determined that the proposed changes to TS 5.4, "Fuel Storage," are acceptable.

Based on the review and evaluation of the licensee's November 18, 1999, submittal (Ref. 1), and its August 7, 2000, responses (Ref. 2) to the staff's request for additional information, the NRC staff concludes that the structural analyses of the SFP structure and the storage rack modules are in compliance with the acceptance criteria set forth in the USAR and are consistent with the current licensing basis.

The NRC staff evaluation supports the conclusion that the proposed modification to the Kewaunee SFP is acceptable because the increase in occupational exposure to individuals due to the storage of additional fuel in the SFP would be negligible and that the installation and use of the new fuel racks does not alter the consequences of a fuel handling accident for the SFP.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to Title 10 of the Code of Federal Regulations (CFR), Part 51, Section 32 (10 CFR 51.32), an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* (65 FR 76672).

5.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from WPSC to US NRC, " Proposed Amendment 167 to the Kewaunee Nuclear Power plant Technical Specifications - Spent Fuel Storage", November 18, 1999.

2. Letter from Mark L. Marchi (WPSC) to US NRC, "Kewaunee Nuclear Power Plant Proposed Technical Specification Amendment Request PA #167 Spent Fuel Storage - Response to Additional Information", August 7, 2000.

3. USNRC "Office of Technology Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as modified by amendment dated January 18, 1979.

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PACKAGE DIVIDER

IWP-003