September 6, 2007

Mr. Stewart B. Minahan
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: ONSITE SPENT FUEL STORAGE EXPANSION (TAC NO. MD3349)

Dear Mr. Minahan:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 227 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 17, 2006, as supplemented by letters dated February 7, April 17, May 4, and July 26, 2007.

The amendment revises TS 4.3.1.1.c, “Criticality,” by adding a new nominal center-to-center distance between fuel assemblies for two new storage racks, and revises TS 4.3.3, “Capacity,” by increasing the capacity of the spent fuel storage pool from 2366 assemblies to 2651 assemblies.

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

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 227 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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BBalsam, DLR/REB
GHill, OIS (2)

ADAMS Accession Nos.: Pkg ML072130018, Amdt./License ML072130023, TS Pgs ML072130026 *SE dated

OFFICE	LPL4/PM	LPL4/LA	IHPB/TL	EMCB/BC	SRXB/BC
NAME	FLyon	JBurkhardt	RPedersen*	KManoly*	GCranston*
DATE	8/6/07	8/6/07	6/21/07	6/6/07	8/17/07

OFFICE	CSGB/BC	SBPB/BC(A)	OGC	LPL4/BC
NAME	AHiser*	SJones*	APHodgdon	THiltz
DATE	6/15/07	7/27/07	8/29/07	8/30/07

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 227
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee), dated October 17, 2006, as supplemented by letters dated February 7, April 17, May 4, and July 26, 2007, 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 license 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 and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: September 6, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Facility Operating License No. DPR-46 and Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
3	3

Technical Specification

REMOVE	INSERT
4.0-2	4.0-2

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2381 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 227, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

(4) <u>Fire Protection</u>

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated October 17, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062990428), as supplemented by letters dated February 7, April 17, May 4, and July 26, 2007 (ADAMS Accession Nos. ML070440315, ML071240495, ML071310384, and ML072120350, respectively), Nebraska Public Power District (NPPD, the licensee), requested changes to the Technical Specifications (TSs) for Cooper Nuclear Station (CNS). The proposed changes would revise TS 4.3.1.1.c, “Criticality,” by adding a new nominal center-to-center distance between fuel assemblies for two new storage racks, and revise TS 4.3.3, “Capacity,” by increasing the capacity of the spent fuel storage pool from 2366 assemblies to 2651 assemblies.

The supplements dated February 7, April 17, May 4, and July 26, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 5, 2006 (71 FR 70561) and January 19, 2007 (72 FR 2560).

Specifically, the licensee proposes to revise TS 4.3.1.1.c, “Criticality,” to reflect the nominal center-to-center dimension between fuel assemblies in the new fuel racks. This TS currently addresses the nominal center-to-center dimension between fuel assemblies placed in the existing Boral-poisoned storage racks. These racks have a center-to-center dimension of 6 9/16 inches. This dimension in the proposed two new Metamic™-poisoned racks is 6.108 inches. The proposed revised TS reads:

> A nominal 6 9/16 inch center-to center distance between fuel assemblies placed in the Boral-poisoned storage racks. A nominal 6.108 inch center-to-center distance between fuel assemblies placed in the Metamic-poisoned storage racks.

The licensee proposes to revise TS 4.3.3, "Capacity," to reflect an increased storage capacity of the spent fuel pool (SFP). The current number of fuel assemblies authorized to be stored in the SFP is 2366. The proposed revised TS reads:

> The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2651 fuel assemblies.

1.1 Background

The CNS SFP currently contains 13 storage racks with a capacity of 2366 fuel assemblies. Currently, the SFP does not have sufficient capacity to accommodate a full core offload. This capability was lost when the spent fuel was discharged to the SFP following Cycle 22 in January 2005.

The licensee stated that it has evaluated spent fuel storage alternatives that are currently feasible for use at CNS. The licensee concluded that increasing the storage capacity of the SFP is the most cost-effective alternative to restore and maintain full core offload capability at CNS as an interim action until dry storage of spent fuel can be implemented.

Increasing the capacity of the SFP to 2651 is based on adding two racks into the SFP. The first rack (Rack A) is a 9 x 13 cell rack that will add 117 storage locations. Rack A rack will be placed into the SFP area north of the cask set-down area (CSA). The second rack (Rack B) is a 14 x 13 cell rack (non-rectangular array) that will add 168 storage locations as a contingency. The only available space in the SFP to place Rack B is the CSA. The CSA and adjacent open space north of the CSA contain portions of the seismic restraint system for the existing rack modules and cask restraint systems. NPPD intends to modify a beam in the vicinity of the CSA to create the space required for Racks A and B, and then to install Rack A. The licensee states that Rack B will be installed in the SFP only if there is a need to offload the entire core into the SFP.

The increased capacity will provide full core offload capability to the licensee until receipt of new fuel for Cycle 26 in summer 2009. For long-term resolution of SFP storage capability, the licensee states that it intends to build an Independent Spent Fuel Storage Installation.

2.0 <u>EVALUATION</u>

The Nuclear Regulatory Commission (NRC) staff divided its review of the licensee's proposed changes into the areas of (1) criticality considerations, (2) use of Metamic™ poison inserts, (3) seismic analysis and structural design, (4) thermal-hydraulic considerations and handling of heavy loads, and (5) health physics. The staff's review of each area is documented below.

2.1 Criticality Considerations

The NRC staff reviewed the proposed change for the purpose of assuring that its design and use continued to prevent criticality in new and spent fuel storage.

2.1.1 Regulatory Basis

The construction of CNS predated the 1971 issuance of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." CNS is designed to be in conformance with the intent of the Draft General Design Criteria (GDC) published in the *Federal Register* on July 11, 1967, except where the licensee made commitments to specific 1971 GDCs. The applicable GDC for criticality consideration is Draft GDC 66 - Prevention of Fuel Storage Criticality:

> Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

The licensee also states in its submittal that the new racks are designed using the guidance of the OT position paper (NRC's letter to the licensee dated April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as revised by letter dated January 18, 1979) and NUREG-0800 applicable to the spent fuel racks. The acceptance criteria of NUREG-0800, Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling Review Responsibilities," includes, in part, 10 CFR Part 50, Appendix A, GDC 62, "Prevention of Criticality in Fuel Storage and Handling," and 10 CFR 50.68, "Criticality accident requirements." The proposed changes do not impact the design or use of the existing storage racks. Since the new storage racks have been designed to prevent criticality in the racks consistent with GDC 62 and 10 CFR 50.68, the proposed changes are acceptable.

2.1.2 Technical Evaluation

The new racks will be separated from each other by a gap of approximately 23 inches. The smallest gap between the new racks and the walls of the SFP will be 10 1/16 inches. The smallest gap between the new racks and the nearest structural member will be 3 29/32 inches. There will be at least 27 inches between the new racks and the existing racks.

With the expanded capacity, the SFP cooling system will be required to remove an increased heat load while maintaining the pool water temperature at or below the design limit of 150 degrees Fahrenheit (°F) bulk-water temperature. The SFP thermal performance and criticality response have been reanalyzed by the licensee considering the increased storage capacity. The NRC staff reviewed the design and analyses performed by the licensee as provided in its submittal and concludes that the design of the new storage racks is consistent with the governing requirements of applicable codes, standards, and NRC guidance, as provided in NRC Generic Letter (GL) 78-11, "Review and Acceptance of Spent Fuel Storage and Handling Applications," as modified by NRC GL 79-04.

Primary nuclear criticality control in the new racks is provided by means of a fixed neutron absorber (Metamic™) integrated within the rack structure. The use of Metamic™ in wet storage pool applications was previously approved by the NRC for use at other nuclear power plants (e.g., Clinton Power Station, Unit 1 (ADAMS Accession No. ML053070598) and Arkansas Nuclear One, Unit 1 (ADAMS Accession No. ML070160040)). The staff's evaluation of the use of Metamic™ at CNS is documented below in section 2.2.

The new spent fuel storage racks were designed by the licensee to maintain the required subcriticality margin when fully loaded with unirradiated fuel assemblies of maximum allowed enrichment at a temperature corresponding to the highest reactivity. For reactivity control in the racks, neutron absorber panels will be used. The panels were sized to sufficiently shadow the active fuel height of fuel assemblies stored in the pool. The panels will be held in place and protected against damage by a stainless steel jacket welded to the cell walls. The panels will be mounted on the exterior or on the interior of the cells, wherever required to satisfy criticality analysis requirements.

As required by TS 4.3.1.1, the spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum exposure-dependent k-infinity [infinite neutron multiplication factor] of 1.29. Furthermore, the new racks were designed by the licensee to assure that the effective neutron multiplication factor (K_{eff}) is equal to or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity and pool-flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations and in mechanical tolerances, statistically combined, giving assurance that the true K_{eff} will be less than 0.95 with a 95 percent probability at a 95 percent confidence level. Reactivity effects of abnormal and accident conditions were also evaluated to assure that under credible abnormal or accident conditions, the reactivity will be maintained less than 0.95. The accidents and malfunctions evaluated included impact on criticality of water temperature and density effects; and impact on criticality of eccentric positioning of fuel assemblies within the rack. The minimum subcriticality margin (i.e., K_{eff} less than or equal to 0.95) will be maintained.

2.1.3 Conclusion

The proposed changes were evaluated by the NRC staff to determine whether applicable regulations and requirements continue to be met. The design and analyses performed by the licensee demonstrate that the new racks comply with the applicable codes and standards. The staff concludes that applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, and sufficient safety margins will be maintained. Since the proposed changes do not impact the design or use of the existing storage racks and the new storage racks have been designed to prevent criticality in the racks, the staff concludes that the proposed changes are acceptable.

2.2 Use of Metamic™ Poison Insert Assemblies

The licensee proposes a modification to the CNS SFP that will increase the capacity of the SFP from 2366 assemblies to 2651 assemblies by adding up to two new storage racks. Metamic™, a fixed neutron poison, will be integrated within the rack structure for nuclear criticality control. The NRC staff evaluated the portions of the submittal addressing behavior of the Metamic™ material used in the racks.

Metamic™ is a fully dense metal matrix composite material composed primarily of B_4C and aluminum alloy Al 6061. B_4C is the constituent in the Metamic™ known to perform effectively as a neutron absorber and Al 6061 is a marine-qualified alloy known for its resistance to corrosion. As noted above in Section 2.1, Metamic™ has previously been approved by the NRC for use in SFPs by other licensees. On the basis of its evaluation, the NRC staff concludes that

Metamic^{TM} is compatible with the environment of the SFP and is not expected to exhibit degradation which could impair the design function of the racks.

2.2.1 Metamic™ Coupon Sampling Program

In the licensee's submittal dated April 17, 2007, the licensee described its Metamic^{TM} coupon sampling program, which consists primarily of monitoring the physical properties of the absorber material by performing periodic dimensional and visual checks to confirm the physical properties. In addition, the program requires that neutron attenuation testing be performed at intervals of 4, 12, and 20 years to confirm the neutron absorption capabilities of the Metamic^{TM} material are being maintained. The licensee's Metamic^{TM} coupon sampling program is similar to that approved by the NRC staff for previous licensees using Metamic^{TM} in SFPs.

2.2.2 Program Description

The purpose of the licensee's Metamic^{TM} coupon sampling program is to ensure the physical and chemical properties of Metamic^{TM} behave in a similar manner as that described in a vendor topical report on simulated service performance of Metamic^{TM}. The coupon program will monitor how the Metamic^{TM} absorber material properties change over time under the radiation, chemical, and thermal environment found in the SFP. The licensee states that its coupon sampling program will be incorporated into CNS station procedures which will direct the performance of the sampling program.

The coupons will be installed on a coupon tree that holds eight coupons. Each coupon is nominally 6 inches long, 4 inches wide, and 0.075 inches thick. Coupon samples will contain 25 percent B_4C, which is consistent with the B_4C content used in the new spent fuel storage racks. The coupon tree will be placed in the SFP at a location that will ensure a representative dose to the coupons. Coupons will be examined on a 2-year basis for the first two operating intervals and thereafter at 4-year intervals over the service life of the new storage racks.

2.2.3 Monitoring Changes in the Physical Properties and Testing of Coupons

The coupon sampling program will require a coupon to be removed from the SFP for testing after 2, 4, 8, 12, 16, 20, 24, and 28 years of service. The licensee stated that when a coupon is removed in accordance with the sampling program, the following measurements will be performed:

1. Physical observation and photography:

    a. The coupons will be observed for physical indications on the surface to detect bubbling, blistering, cracking, or flaking or any other visual degradation.
    b. Photographs will be taken of both sides of the exposed coupon.

2. Dimensional measurements:

    a. Length
    b. Width
    c. Thickness

3. Mass

4. Neutron attenuation testing

    a. Neutron attenuation testing will be conducted to confirm the neutron absorption capabilities if there are physical changes outside of the allowable tolerances given below.
    b. Neutron attenuation testing will also be conducted regardless of the results of the physical testing after 4, 12, and 20 years of service.

The licensee's acceptance criteria for dimensional, weight, and density measurements are as follows:

- Any change in the length and width of ± 0.125 inches

- Any change in the thickness of ± 0.07 inches

- Any change mass of ± 5 percent

The NRC staff concludes that these are reasonable limits that will assure further evaluation before significant degradation occurs.

Prior to installing the coupons in the SFP, each coupon is pre-characterized. The physical characteristics discussed above are documented for each coupon. When a coupon is removed, measurements and physical observations will be recorded and evaluated for any physical or visual change when compared to the original data. If the measurements taken do not meet the established acceptance criteria, the licensee will perform an investigation which will include directly assessing the neutron absorption capabilities. If the neutron attenuation testing reveals degradation, the impact on K_{eff} would be evaluated. The intent of this evaluation would be to confirm that the value of K_{eff} for spent fuel storage in the SFP remains less than 0.95. After all testing is finished, the coupons will be returned to the coupon tree, to support long-term testing, as required.

The licensee stated that the results of the baseline inspection data and subsequent coupon sampling program results will be submitted to the NRC staff for review.

2.2.4 Conclusion

Based on its review of the licensee's submittal, the NRC staff concludes that the Metamic™ neutron absorber is compatible with the environment of the SFP. Also, the staff finds the proposed coupon sampling program, which includes visual, physical, and confirmatory tests, is capable of detecting potential degradation of the Metamic™ material that could impair its neutron absorption capability. Therefore, the staff concludes that the use of Metamic™ as a neutron absorber panel in the new spent fuel racks at CNS is acceptable.

2.3 Seismic Analysis and Structural Design Review

2.3.1 Regulatory Requirements

In its review, the NRC staff used the regulatory guidance documented in Enclosure 1 to the NRC's letter to the licensee dated April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (referred to as the OT Position paper), as revised by letter dated January 18, 1979 (Reference 3; these two letters were subsequently numbered NRC GLs 78-11 and 79-04, respectively), and NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.8.4, "Other Seismic Category I Structures," Appendix D, "Technical Position on Spent Fuel Racks," Revision 0, dated July 1981 (Reference 4).

As documented in Section II of SRP 3.8.4 (Reference 14), the NRC staff's acceptance criteria are based on 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1, as they relate to safety-related structures being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed; GDC 2 as it relates to the design of the safety-related structures being capable to withstand the most severe natural phenomena such as wind, tornadoes, floods, and earthquakes and the appropriate combination of all loads; GDC 4 as it relates to safety-related structures being capable of withstanding the dynamic effects of equipment failures including missiles and blowdown loads associated with the loss of coolant accidents; GDC 5 as it relates to sharing of structures important to safety unless it can be shown that such sharing will not significantly impair their validity to perform their safety functions; and 10 CFR Part 50, Appendix B, as it relates to the quality assurance criteria for nuclear power plants.

In its October 17, 2006, submittal (Reference 1), NPPD states that the proposed rack modifications to the CNS SFP are designed and analyzed in accordance with the NRC guidance of the OT Position paper (Reference 3), and SRP 3.8.4, Appendix D, Revision 0 (Reference 4). NPPD also states that material procurement for the new racks and the analysis, fabrication, and installation of the new racks conforms to the requirements of 10 CFR Part 50, Appendix B.

The NRC staff concludes that NPPD's methodology for the design of the new racks is in accordance with NRC staff recommendations and is, therefore, acceptable.

2.3.2 Design Criteria and Applicable Codes

Section IV(2), "Applicable Codes, Standards and Specifications," of the OT Position paper states that, "Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC [American Institute of Steel Construction] specification or Subsection NF requirements of Section III of the ASME B&PV Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] for Class 3 component supports." NPPD states that primary stresses in the rack modules are required to satisfy the stress limits documented in Section III, Subsection NF, and Appendix F of the ASME B&PV Code for

Class 3 linear structures (Reference 5) for the load combinations documented in the OT Position paper. For the CNS racks, NPPD defines the code jurisdictional boundary at the interface between the rack shear pads and the supporting platforms for the new racks. The code, therefore, defines the platforms to be "intervening parts" that should be engineered to enable the subject NF structures (i.e., the racks) to perform their intended functions, but does not mandate any specific stress limits for such components. NPPD states that the platforms for the new racks are also designed to NF limits. However, the NPPD states that, "Because the platforms are not an integral part of the rack, their stress analysis and structural qualification are not addressed in this licensing report."

In its April 17, 2007, supplement (Reference 8), NPPD stated that an analysis of Platforms A and B has been performed using the ANSYS commercial computer code. Calculated stresses in the platform components and welds meet ASME B&PV Code, Section III, Subsection NF, and Appendix F requirements for the Level A and Level D service loads. Platform A is shown on Holtec International Drawing 4732 (Reference 6), and Platform B is shown on Black & Veatch Drawing 142707-1BSA-S6002 (Reference 7), which were provided in NPPD's supplement dated April 17, 2007.

The NRC staff concurs that NPPD's analysis of Platforms A and B, as summarized in Item 3 of Attachment 2 to NPPD's supplement dated April 17, 2007, demonstrates that the platforms are structurally adequate for the imposed service loads. A list of the codes, standards, and NRC documents used by NPPD as guidance documents in the design of the SFP racks is documented in Section 2.2 of Reference 2.

The NRC staff finds NPPD's use of ASME B&PV Code, Section III, stress limits and the load combinations documented in the NRC's OT Position paper to design the new CNS SFP racks to be in accordance with NRC staff guidance and, therefore, to be acceptable.

2.3.3 Rack Geometry and Material

CNS Racks A and B each consist of a cellular structure and a baseplate with shear pads. Each rack is freestanding and self-supporting. The base of each rack bears on a platform (Platforms A and B). The racks are not mechanically connected to the platforms. The racks are primarily fabricated from SA240-Type 304 austenitic stainless steel sheet and plate stock. Metamic™ neutron absorber is the material used for reactivity control. The plan dimensions of Rack A are about 55 inches by 80 inches. The plan dimensions of Rack B are about 86 inches by 80 inches. The dry weight of Rack A is about 13,000 pounds. The dry weight of Rack B is about 18,500 pounds. The platforms are also fabricated from SA240-Type 304 austenitic stainless steel. The tops of Racks A and B are at the same elevation as the tops of the existing racks. The platforms elevate the bottoms of the rack baseplates to prevent interference with hardware connected to the SFP liner. The base of the cellular portion of each rack is welded to the top of the baseplate. The top of the baseplate also provides the bearing surface for the bottom fitting of each fuel assembly. Shear pads are welded to the underside of each baseplate at the corners of the baseplates.

Platform A is an open-lattice structure that is anchored to the SFP liner structure by existing swing bolts. The NPPD submittal does not document an analysis of the swing bolts for the loads Platform A transmits. In its April 17, 2007, supplement, NPPD states that an analysis has been performed for the swing bolts and swing-bolt anchorages and that these components meet ASME B&PV Code, Section III, Subsection NF, stress limits. The locations of the swing bolts are shown on Burns and Roe Drawing 4228 (Reference 9). Details of the welded connections between the swing bolts and SFP floor slab are shown on Burns and Roe Drawing 4230 (Reference 10). The drawings were provided in NPPD's supplement dated April 17, 2007.

NPPD concludes, and the NRC staff agrees, that the swing bolts and swing-bolt anchorages for Platform A are structurally adequate for the imposed service loads. A summary of NPPD's analysis is documented in Item 1 of Attachment 2 to NPPD's supplement dated April 17, 2007.

The NPPD submittal indicates that Platform B rests directly on the SFP liner and is not anchored to the SFP liner structure. NPPD notes that, "any significant membrane strains in the pool liner are prevented by the presence of the platforms. As a result, the maximum strain sustained by the liner during a seismic event is assumed to be less than the ultimate strain for the liner material (austenitic stainless steel, ultimate strain $\geq$0.38)." However, the NPPD submittal does not document that the SFP liner plate remains leak-tight for the bearing and friction loads Platform B transmits into the liner. In its April 17, 2007, supplement, NPPD provided Burns and Roe Drawing 4288 (Reference 11) to demonstrate that Platform B does not bear directly on the liner, but instead bears on an existing 7 foot-by-7 foot-by-1 inch-thick cask pad that is welded to the 1/4 inch SFP liner plate with continuous 1/4 inch fillet welds. Platform B is also partially restrained by a new 2 inch-thick by 2 1/2-wide by 6 foot 11 inch inside diameter circular cask ring that is welded to the cask pad with continuous 5/16 inch fillet welds. The new cask ring is also shown on the Burns and Roe Drawing 4228 (Reference 11). NPPD also provided Black & Veatch Drawing 142707-1BSA-S6002 (Reference 7), which shows the proposed installation of Platform B over the new cask ring. NPPD notes that Platform B does not bear vertically on the cask-restraint ring. The vertical loads transmitted through Platform B react directly into the 1 inch-thick cask pad. The cask restraint ring is designed to react the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) shear loads into the cask pad. NPPD states that an analysis of the 5/16" fillet weld between the cask ring and the cask pad demonstrates that the weld meets Level A and Level D allowable shear stresses in the weld throat. Platform B, therefore, does not bear directly on the SFP liner but instead bears on the intermediate 1 inch-thick cask pad welded to the liner. NPPD's summary of the stress analysis is documented in Item 2 of Attachment 2 to NPPD's supplement dated April 17, 2007.

2.3.4 Rack Structural Qualification

2.3.4.1 Placement of New Racks

NPPD states that the existing SFP racks are laterally restrained at their bases and at their tops, which prevents lateral movement and eliminates fluid coupling forces between the racks during a seismic event. The gaps between the new racks and the existing racks and walls of the SFP are large enough to prevent contact and to minimize fluid coupling forces during a seismic event.

Based on the restraint pattern of the existing racks and the spacing between the new racks and the existing racks and walls of the SFP, the NRC staff concurs with NPPD's conclusion that it is acceptable to analyze the new rack modules by the "single rack seismic analysis" procedure.

2.3.4.2 Applicable Load Combinations

As noted in Section 2.3 above, NPPD analyzed the new racks using the load combinations documented in the NRC staff's OT Position paper and Appendix D to SRP 3.8.4 using ASME B&PV Code, Section III, Subsection NF stress limits. Loads considered include: dead weight (D), including the dead weight of stored spent fuel and control elements; live load (L); the upward force on the racks caused by a postulated stuck fuel assembly (P_f); the impact force due to the accidental drop of the heaviest load from the maximum possible height (F_d); the operating basis earthquake (OBE, or E); the safe shutdown earthquake (SSE, or E'); differential thermal expansion loads under normal conditions (T_o); and differential thermal expansion loads under postulated abnormal conditions (T_a). These loads are separately combined as tabulated in Section 6.3 of the NPPD submittal for the normal, upset, and faulted conditions. NPPD conservatively bounds the upset condition load combination for the OBE by evaluating the faulted condition load combination for the SSE using normal condition stress limits. NPPD states that T_o and T_a are not applicable to the stress analysis of the new SFP racks since these thermal expansion loads produce stresses that are self-limiting and because the new racks are free to expand or contract. Also, no live load (L) is identified for the new racks.

NPPD states that mechanical loads P_f and F_d result in plastic strains that are not evaluated to ASME B&PV Code, Section III, Subsection NF, stress limits but are instead evaluated to determine the extent of local damage the new racks sustain under localized loads. The NRC staff's review of NPPD's evaluation of the mechanical loads postulated for the new racks is separately documented in Section 2.3.4.9 of this safety evaluation.

2.3.4.3 Synthetic Time Histories

Since the new fuel racks are non-linear structures due to their restraint mechanisms (friction/bearing) and free-to-rattle fuel bundles, NPPD generated synthetic acceleration time histories for the SSE in the north-to-south (N-S), east-to-west (E-W), and vertical directions in accordance with the requirements of SRP 3.7.1, “Seismic Design Parameters” (Reference 12). The maximum reactions obtained from the time-history solutions at the bases of the new racks were combined by the square root sum of the squares (SRSS) and used to design the welds connecting the rack cells to the tops of the rack baseplates. NPPD did not take credit for material (hysteresis) or fluid damping in the time history-generation algorithm. NPPD generated the acceleration time histories for the SFP slab in accordance with the SRP requirement that the response spectra generated from the acceleration time-histories envelop the design-basis response spectra. Table 6.4.1 of Enclosure 1 of NPPD's submittal tabulates the design-basis zero period accelerations (ZPA) in the N-S, E-W, and vertical directions for the OBE and SSE response spectra.

2.3.4.4 Analysis Methodology

NPPD used the DYNARACK proprietary software code to integrate the rack nonlinear equations of motion with the three orthogonal acceleration time histories as the forcing functions. NPPD states that the DYNARACK program has been used for nearly all rerack license amendment requests over the past 2 decades. The analytical basis of the DYNARACK program is documented in Reference 6.5.1 of Enclosure 1 of NPPD's submittal.

The NRC staff has previously reviewed and approved the use of the DYNARACK code for rack analysis (e.g., Clinton Power Station, Unit 1 (ADAMS Accession No. ML053070598)).

2.3.4.5 Rack Dynamic Model

NPPD has modeled the new rack as a 12 degree-of-freedom (DOF) structure with 6 DOF at the top of the rack and 6 DOF at the base of the rack. Bending and shear springs connect the lumped masses. Each fuel assembly is modeled as a slender rod pinned at the base and free at the top and is able to displace laterally (rattle) inside its storage cell within a specified gap. The mass of each fuel assembly is lumped at the top and bottom of the rack and at the rack quarter points. Beam springs connect the adjacent nodes. Compression-only gap elements account for potential impact between the fuel assembly masses and the walls of the fuel cells. Fluid coupling coefficients are based on the nominal gap between the fuel assemblies and cell walls to model fluid resistance to gap closure. The vertical (axial) motion of each fuel assembly is assumed rigid and equal to the vertical motion of the rack baseplate. The centroid of each fuel assembly can be offset with respect to the centroid of the rack structure at the same elevation to model a partially loaded rack. The fuel assemblies are assumed to move in phase within the rack during a seismic event to maximize dynamic loads. The rack model accounts for fluid coupling between the fuel assemblies and the rack and between the rack and adjacent walls. The derivation of the fluid coupling matrix is based on fluid mechanics principles that Holtec International verified by shake-table experiments in the late 1980s. Fluid damping and form drag are conservatively neglected. Since the top of the rack is more than 25 feet below the water surface of the SFP, sloshing of the water mass surrounding the rack is negligible and is neglected in the rack dynamic model. Friction springs and compression-only springs model the reactions between the bottoms of the rack shear pads and the top of the platform supporting the rack. Bounding values of 0.2 and 0.8 are used for the coefficient of friction. Table 6.5.2 of Enclosure 1 of the NPPD submittal tabulates the 22 translational and rotational DOFs for the rack model.

2.3.4.6 Acceptance Criteria

In addition to ASME B&PV Code, Section III, Subsection NF, stress limits, the new racks satisfy the kinematic acceptance criteria documented in Section 6 of Appendix D to SRP 3.8.4. Section 6 requires that factors of safety against sliding and overturning under a seismic event meet the requirements of SRP Section 3.8.5, “Foundations,” Subsection II.5 (Reference 13). Subsection 6(a) of Appendix D waives the requirement to meet the factor of safety against sliding if “it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between the adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by SRP Section 3.8.5, subsection II.5.”

As documented in the NPPD submittal, the new racks are designed to meet the factors of safety against tilting specified in SRP Section 3.8.5 (1.5 times the OBE or 1.1 times the SSE). In addition, the new racks are not permitted to impact adjacent SFP structures, including the existing racks and structural restraints. Finally, the rack shear pads are not permitted to slide past the edges of the platform supporting the rack under a seismic event.

NPPD stated that it imposed a separate impact criterion on the rack fuel assemblies. Based on studies conducted by Lawrence Livermore National Laboratory, as documented in Reference 6.6.2 of Enclosure 1 of the NPPD submittal, the fuel assemblies are required to exhibit accelerations less than 63 g (acceleration of gravity) due to rattling under a seismic event.

2.3.4.7 Input Data

Table 6.7.1 of Enclosure 1 of the NPPD submittal tabulates the primary input data for the seismic analysis of Racks A and B, including the height of the rack above the top of the baseplate, shear pad thickness, and storage cell square dimensions. Table 6.7.1 also specifies 4 percent damping for the OBE and 5 percent damping for the SSE as input data for the rack seismic analysis. Section (3) of Appendix D to SRP Section 3.8.4 (Reference 4) states that,

> For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in SRP Section 3.7. The ground response spectra and damping values should correspond to Regulatory Guides 1.60 and 1.61, respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guides 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable."

Table 1 of NRC Regulatory Guide 1.61 specifies 2 percent damping for OBE and 4 percent damping for SSE for welded steel structures. In its April 17, 2007, supplement, NPPD stated that,

> The CNS design and licensing basis information found in USAR [updated safety analysis report] Section XII-2.3.5.2.5 indicates that "Steel Frame Structures" are to be analyzed using a damping value of 2.0 percent and "Welded Assemblies" are to be analyzed using a damping value of 1.0 percent when conducting dynamic analyses using seismic response spectra methodology. The selection of the CNS design and licensing basis ground response spectra for the seismic design analyses of safety-related Structures, Systems, and Components (SSCs) was completed prior to the October 1973 issuance of NRC Regulatory Guides 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." The CNS-specific ground response spectra does not completely envelope [sic] the Regulatory Guide 1.60 spectra, which precludes the direct use of the higher (less conservative) damping values permitted by Regulatory Guide 1.61 for the analysis of welded steel structures.
>
> The subject CNS OBE floor response spectra (at 4 percent damping) and SSE floor response spectra (at 5 percent damping), were not directly utilized to conduct dynamic response spectra-type analyses of the proposed new storage racks. The subject floor response spectra for the 976'-0" elevation of the Reactor Building were utilized to create

artificial acceleration time histories of the dynamic input motion applicable to the location of the proposed fuel storage racks. The synthetic “conversion” of the subject floor response spectra information to artificial time-history input motion was confirmed to be accurate by ensuring that “output” floor response spectra, created from these artificial time-history input motions, would adequately and appropriately envelope the CNS OBE and SSE floor response spectra originally provided to the analysts. These “verified” artificial time histories were then used as input data to conduct the non-linear dynamic analyses of the new storage racks, which are base-supported on the storage pool floor, elevation 962'-3".

Time-history dynamic input motion information is not dependent on an assumed damping level in the structure being dynamically loaded, as the “input” information is in the format of acceleration versus time, rather than a format of acceleration versus structural response (frequency or period). As such, the damping level of the “input” floor response spectra would not be critical to the dynamic analyses of the proposed new storage racks.

The numeric values of the structural damping values assumed in the rack structures were confirmed to be as listed in Table 6.7.1 of the NPPD report (4 percent for the OBE dynamic analyses, and 5 percent for the SSE dynamic analyses). These internal damping values are not in accordance with the CNS USAR, nor are they consistent with Regulatory Guide 1.61. The appropriate structural damping value for use in conducting each of the dynamic analyses for CNS is 1 percent. As such, the assumed structural damping values utilized in the rack dynamic analyses are potentially non-conservative.

As the lowest fundamental mode of horizontal structural response in the proposed new storage racks was determined by analysis to be approximately 7 Hz [hertz, cycles per second], and because the input dynamic response was applicable to an elevation higher than the pool floor (976'-0" versus 962'-3"), the effect of this non-conservative assumption is not significant.

The potential increase in seismic response is estimated as follows:

Seismic Level	7 Hz Response at 1% Damping, 958'-3"/ 976'-0" Level	7 Hz Response at 4% Damping, 976'-0" Level	7 Hz Response at 5% Damping, 976'-0" Level	Potential Impact of 1% Damping on Response
OBE	0.37g	0.37g	N/A	Nil
SSE	0.60g	N/A	0.58g	0.02 g (3.3%) increase

The use of potentially non-conservative 4 percent damping in the rack structure for the OBE analyses has a negligible impact on the response when compared to the required 1 percent damping response. The use of potentially non-conservative 5 percent damping in the rack structure for the SSE analyses has a small impact (less than 3.5 percent) when compared to the required 1 percent damping response. This small difference is not considered to be significant.

The NRC staff concludes that NPPD's seismic analysis of the new racks remains valid despite NPPD's use of 4 percent damping for the OBE and 5 percent damping for the SSE instead of the design-basis damping value of 1 percent. The stress and kinematic margins of safety calculated for the new racks are documented in Section 6.8 of the NPPD submittal and summarized in Section 2.3.4.8 of this safety evaluation.

Section 6.7.2 of Enclosure 1 of the NPPD submittal documents the yield and ultimate strengths for SA240-304L material used in the analyses. The stress limits for this material are lower than the stress limits of the SA240-304 material used to fabricate the new racks.

2.3.4.8 Parametric Review

Table 6.7.3 of Enclosure 1 of the NPPD submittal lists a total of 26 different rack analyses (13 for Rack A and 13 for Rack B). Analysis variables include full or partial fuel loading, magnitude of coefficient of friction, and OBE or SSE seismic input. The results of these analyses are listed in Table 6.8.1 of Enclosure 1 of the NPPD submittal. Table 6.8.1 documents the maximum rack lateral displacement, the maximum stress factor (the ratio of the calculated and allowable stress), the maximum vertical load, the maximum shear load, and the maximum fuel-to-cell-wall impact. Based on the results of these analyses, NPPD concludes that (1) the new racks possess a large margin of safety against impact and an even larger margin of safety against overturning, (2) maximum stress factors for the faulted condition meet upset-condition stress limits with large margins of safety, and (3) the new racks will not slide past the edges of their supporting platforms. Section 6.8 of Enclosure 1 of the NPPD submittal tabulates the maximum calculated rack displacements and minimum clearances to demonstrate that the new racks do not impact the adjacent SFP walls or the seismic restraints of the existing racks.

With respect to the acceptance criterion for the rack fuel assemblies, the calculated maximum impact load corresponds to a deceleration of about 6 g, which is about one-tenth of the acceptance criterion of 63 g.

Based on its review of the information provided by the licensee, the NRC staff concludes that Racks A and B meet postulated stress and kinematic criteria for the imposed service loads.

2.3.4.9 Mechanical Accidents

Subsection IV.(1)(b) of the NRC staff's OT Position Paper states that, "Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack." Section (4) of Appendix D to SRP 3.8.4 states, in part, that, "The fuel pool racks, the fuel pool structure including the pool slab and fuel pool liner, should be evaluated for accident load combinations which include the impact of the spent fuel cask, the heaviest postulated load drop, and/or accidental drop of fuel assembly from maximum height. The acceptable limits (strain or stress limits) in this case will be reviewed on a case-by-case basis but in general the applicant is required to demonstrate that the functional capability and/or the structural integrity of each component is maintained." The fuel racks are, therefore, not required to meet faulted condition stress limits for a postulated drop. Instead, Table 1 of Appendix D requires that, "The functional capability of the fuel racks should be demonstrated" for the faulted condition load combination that contains F_d, the postulated drop.

NPPD evaluated the damage to the new racks, rack platforms, and the SFP liner and slab due to the impact of a fuel assembly for a postulated shallow-drop event and two deep-drop events. NPPD did not evaluate an inclined-drop event. NPPD considered the inclined-drop event to be bounded by the postulated shallow-drop event.

For the shallow-drop event, a fuel assembly and a portion of the fuel handling tool is assumed to drop vertically and impact the top of a rack cell and the fuel assembly stored in the cell. For rack function to be preserved, damage to the impacted cell walls must be limited to the portion of the cell above the top of the active fuel region (the neutron absorber) located 13 1/16 inches below the top of the cell. Since the impact resistance of a rack cell at the perimeter of the rack is less than the impact resistance of an interior cell, the bounding shallow-drop event is postulated to impact the outer wall of a rack cell located on the perimeter of the rack.

The first postulated deep drop assumes that a fuel assembly falls through an empty storage cell located in the rack interior and impacts the rack baseplate away from the baseplate shear pads.

For rack function to be preserved, the baseplate is required to remain intact. Since Platform A is a box structure fabricated without a cover plate, the rack baseplate is the sole structural barrier between the impacting fuel assembly and the liner below Platform A. Platform B is fabricated with a 1 inch cover plate and bears on an existing 1 inch cask pad welded to the liner. The Rack A geometry is, therefore, the bounding geometry for the first postulated deep drop. The second postulated deep drop assumes that a fuel assembly falls though an empty storage cell located above a baseplate shear pad. The rigid impact surface reacts the impact load through the rack shear pad and liner into the SFP floor slab. For SFP function to be preserved, the liner is required to remain leak-tight. The Rack A geometry is also the bounding geometry for the second postulated deep drop. For these postulated deep drops, the magnitude of the free-fall height used in the evaluation bounds the maximum elevation of a fuel assembly in transit. NPPD also evaluated the structural integrity of the rack cell walls for the uplift load caused by a postulated stuck fuel assembly.

NPPD used the computer code LS-DYNA to prepare the finite element models (FEMs) for the postulated events. The NRC staff has previously reviewed and approved the use of LS-DYNA for rack analysis (e.g., Clinton Power Station, Unit 1 (ADAMS Accession No. ML053070598) and Diablo Canyon Power Plant, Units 1 and 2 (ADAMS Accession No. ML052970272)). For the postulated drops, NPPD assumes that the fuel assemblies are rigid and impact the postulated targets with no loss of energy. The fuel assembly impact velocities are not reduced due to the effects of fluid drag. Minimum ASME Code material properties are used in the FEM analyses.

Table 7.5.1 of Enclosure 1 of NPPD's submittal summarizes the weights, drop heights and impact velocities used in the FEM analyses for the shallow- and deep-drop events. The FEM analysis for the shallow-drop event demonstrates that the maximum depth of plastic deformation due to the impact of the fuel assembly does not extend into the active fuel region of any stored fuel. The FEM analysis of the deep-drop event through an interior cell demonstrates that the impacting fuel assembly deforms the baseplate with local severing of the baseplate/cellwall welds. NPPD has determined that the lowered seating position of the fuel assembly due to the deformation of the baseplate is within acceptable limits. The FEM analysis of the deep-drop event above a baseplate shear pad produces a maximum stress in the liner beneath the shear pad that is about half of the liner-yield strength. NPPD's FEM analysis of the stuck-fuel event demonstrates that the structural components of the new racks maintain adequate margins of safety for the bounding uplift load.

Table 7.5.3 of Enclosure 1 of NPPD's submittal summarizes the results of the FEM analyses for the shallow-drop, deep-drop, and stuck-fuel events. Table 7.5.3 states that the calculated values of the evaluation parameters for the shallow-drop, deep-drop, and stuck-fuel events are no more than about half the allowable values, except for the deformation of the baseplate due to a deep-drop event through an interior cell. For this postulated deep drop, the calculated deformation of the baseplate is 2.93 inches versus an allowable deformation of 3 inches. However, the bottom of the deformed baseplate still remains about 10 inches above the SFP liner due to the combined height of the baseplate shear pads and the supporting platform.

Based on the results of NPPD's FEM analyses for the shallow-drop, deep-drop and stuck-fuel events, NPPD concluded, and the NRC staff concurs, that the new fuel racks maintain adequate margins of safety for the postulated mechanical accidents.

2.3.5 Fuel Pool Structural Integrity Evaluation

NPPD evaluated the SFP floor slab for the increased loads due to the addition of Racks A and B for the bounding service and factored load combinations tabulated in Section II.3 of SRP 3.8.4 (Reference 14). Loads combined include dead (D), live (L), normal operating thermal (T_o), seismic OBE (E), and seismic SSE (E').

To determine the magnitudes of the vertical seismic loads acting on the SFP floor slab, NPPD performed a preliminary modal analysis of the floor slab that demonstrates that the fundamental frequency of the floor slab in the vertical direction is 35.4 Hz, which is greater than the rigid-range frequency of 33 Hz. NPPD, therefore, used the design-basis OBE and SSE ZPA as seismic load factors to analyze the floor slab.

As documented in NPPD's submittal, NPPD performed the modal analysis of the floor slab assuming an uncracked section modulus for the floor slab cross-section. NPPD documented the basis for this assumption in Item 4(a) of Attachment 2 to NPPD's supplement dated April 17, 2007, which states, in part, that, "Cracked section properties are used only to evaluate thermal loads and to provide a realistic assessment of the redistributed internal forces and moments, as permitted by Section A.3.3 of American Concrete Institute (ACI) 349. The intent of the ACI Committee is further clarified in ACI 349R-85 (Commentary on Code Requirements for Nuclear Safety Related Concrete Structures), which states that the analysis may 'consider the structure uncracked for mechanical loads and only consider the effect of cracking on thermal loads.' Holtec has used this method of analysis numerous times to qualify reinforced concrete SFP structures, based on an established history of acceptance by the NRC."

The NRC staff, therefore, accepts NPPD's basis for the use of an uncracked section modulus to perform the modal analysis of the SFP floor slab.

NPPD documented incorporation of the mass of the SFP water in the modal analysis of the floor slab in Item 4(b) of Attachment 2 to NPPD's supplement dated April 17, 2007, which states, in part, that, "The calculated first mode frequency of 35.4 Hz for the SFP slab, reported in Holtec Report No. HI-2043224, is based on a 64-inch thick concrete slab (γ = 150 lb/ft^3) with simply supported boundary conditions and no additional fluid mass. While it is clearly conservative to assume simply supported boundary conditions, it is non-conservative to assume that none of the contained SFP water mass participates in the dynamic response of the SFP slab. To provide a more accurate estimate of the SFP floor fundamental frequency, a series of modal analyses have been performed assuming both clamped and simply supported boundary conditions and increased slab densities to account for half or all of the contained SFP water mass. The minimum result is 18.4 Hz, which represents a conservative lower bound estimate of the slab fundamental frequency since it assumes both simply supported boundary conditions and full participation of the SFP water mass. In reality, the SFP slab behaves more like a rectangular plate with clamped edges, and the mass participation of the SFP water is less than 100 percent since the water is not rigidly attached to the slab. Therefore, it is reasonable to conclude that the fundamental frequency of the slab is above 20 Hz. Since the vertical SSE response spectrum for the SFP floor, which is shown in Figure 3, has a constant acceleration above 20 Hz, the use of the zero period acceleration (ZPA) to compute the seismic amplification of the SFP slab and the contained SFP water mass is justified, and the minimum safety factors reported in Holtec Report No. HI-2043224 are indeed valid."

The NRC staff concurs that NPPD's revised modal analysis of the SFP floor slab to incorporate the SFP water mass confirms the use of the OBE and SSE ZPA as seismic load factors.

NPPD used the ANSYS commercial computer code to prepare a finite element model of the SFP floor slab for the bounding service and factored load combinations that combine dead (D), live (L), normal operating thermal (T_o), OBE (E), and SSE (E') ZPA loads. In Item 4(b) of Attachment 2 of NPPD's supplement dated April 17, 2007, NPPD noted that, "Finally, the static mass of the SFP water was inadvertently omitted from Table 8.5.1 of Holtec Report HI-2043224. The finite element analysis of the SFP slab conservatively considers a uniform acting pressure of 16.9 pounds per square inch (psi) over the entire SFP slab area. This represents a total hydrostatic load of 2.7 million pounds, which is significantly more than the contained water mass of 2,100 thousand pounds reported in CNS Updated Safety Analysis Report (USAR) Section XII-2.3.3.2.4. For the earthquake load, the hydrostatic load (2,700 thousand pounds) is amplified by the vertical ZPA values for OBE (0.0685 g) and SSE (0.137 g)." Table 8.5.1 of Enclosure 1 of NPPD's report tabulates the dead loads on the SFP floor slab due to the weights of the existing and new racks and fuel. NPPD uniformly distributed the total weight acting on the floor slab over the floor slab area. NPPD considered the combined weights of Rack B and the cask in the analysis, which is conservative. For the normal operating thermal load, NPPD evaluated a thermal gradient based on a bulk pool temperature of 160 ^{o}F for the top of the SFP floor slab and an ambient temperature of 85 ^{o}F for the bottom of the SFP floor slab. The results of NPPD's finite element analysis of the SFP floor slab are tabulated in Table 8.6.1 of Enclosure 1 of NPPD's submittal. The factors of safety tabulated in the table for the floor slab moments and shears at critical cross-sections are generally between 2.0 and 4.0.

NPPD concluded, and the NRC staff concurs, that the structural integrity of the SFP floor slab will remain adequate for the additional weights of the new racks.

Regarding any nonconformances related to material degradation issues in the SFP, NPPD noted in Item 5 of Attachment 2 to its supplement dated April 17, 2007, that, "No nonconformance related to material degradation issues in the concrete/rebar structural elements of the CNS SFP have been documented to date. No leakage from the CNS SFP has been identified to date. However, there were two significant nonconformance (events), not related to material degradation issues, which are relevant to the integrity of the CNS SFP. These events involved dropping a core shroud head bolt and dropping a control rod blade in the SFP. Neither of these two events resulted in any discernable damage to the 1/4-inch thick stainless steel liner plate. The core shroud head bolt did not come into contact with the liner plate. The area of contact/impact of the control rod blade with the liner plate was inspected through the use of an underwater camera. No damage was visible." Based on the information provided by the licensee, the NRC staff concludes that there are no substantive nonconformances related to material degradation issues in the SFP.

2.3.6 Heavy Loads Considerations

The CNS USAR, Section 4.6, "Control of Heavy Loads," documents NPPD's response to GL 80-113, "Control of Heavy Loads" (Reference 15). GL 80-113 requested that licensees of operating plants review controls for the handling of heavy loads in accordance with the recommendations documented in NUREG-0612 (Reference 16). Section 5.1.1 of NUREG-0612 recommends, in part, that, (1) safe load paths be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment; (2) procedures be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment; (3) crane operators be trained and qualified in accordance with Chapter 2-3 of American National Standards Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes"; (4) special lifting devices satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials"; (5) lifting devices not specially designed be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings"; (6) the [reactor building] crane be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, "Overhead and Gantry Cranes," with the exception that tests and inspections be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency; and (7) the crane be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, "Overhead and Gantry Cranes," and of CMAA-70, "Specifications for Electric Overhead Travelling Cranes."

NPPD states in Table 10.1.2 of its submittal that Rack A (and Rack B, if required) will be installed in compliance with the recommendations documented in NUREG-0612. NPPD states that the heaviest total lift will be less than 25,000 pounds, which is about one-eighth of the 100-ton (200,000 pounds) rating of the reactor building crane main hook. A remotely engaging lift rig that meets the applicable guidelines of NUREG-0612 will be used to lift the new rack. The new rack will be placed in the SFP after the support platform has been installed and leveled. The new rack will be moved along a pre-established safe path before being lowered into the cask pit and placed on its platform. The new rack will be leveled with shims if required. As-built gaps will be measured and adjusted as necessary to comply with design dimensions. Holtec International will install Rack A (and Rack B, if required) using applicable Holtec and CNS procedures. NPPD also noted that Holtec International has installed "over 1,000 racks in light water reactor pools around the world without a single mishap."

Based on the information provided by the licensee, the NRC staff concludes that NPPD's controls to place the new racks into the CNS SFP meet the requirements of NUREG-0612. Additional details of the NRC staff's review of the licensee's proposed handling of heavy loads are provided in section 2.4 below.

2.3.7 Conclusion

Based on the NRC staff's review of NPPD's submittal, as supplemented (References 2 and 8), the staff concludes that NPPD's analyses were performed in accordance with the regulatory guidance summarized above. The staff also concurs with NPPD's conclusions that:

- The new rack modules are designed in accordance with NRC staff recommendations.
- The NRC staff has previously approved NPPD's use of DYNARACK and other proprietary software for the analysis of the rack modules.
- The seismic analysis of the new rack modules remains valid despite NPPD's use of 4 percent damping for the OBE and 5 percent damping for the SSE instead of the design-basis damping value of 1 percent.
- The new rack modules meet postulated stress and kinematic criteria.
- The new rack modules maintain adequate margins of safety for the postulated mechanical accidents.
- The revised modal analysis of the SFP floor slab to incorporate the SFP water mass confirms the use of the OBE and SSE ZPAs as seismic load factors.
- The structural integrity of the SFP liner and floor slab will remain adequate for the additional weights of the new rack modules.
- Controls to place the new racks into the SFP meet the requirements of NUREG-0612.

Based on its review of the seismic analysis and structural design, the NRC staff concludes that the proposed addition of the two new storage racks to the SFP is acceptable.

2.3.8 References for Section 2.3

1. Letter from S. Minahan (NPPD) to NRC, "License Amendment Request to Revise Technical Specification - Onsite Spent Fuel Storage Expansion/Cooper Nuclear Station, Docket No. 50-298, DPR-46," dated October 17, 2006.

2. Enclosure 1 to Letter dated October 17, 2006, from S. Minahan (NPPD) to NRC, "Licensing Report on the Wet Fuel Storage Capacity Expansion at Cooper Nuclear Station/Cooper Nuclear Station/Docket No. 50-298, DPR-46/Proprietary Version".

3. Enclosure 1 to NRC Letter, Docket No. 50-289, dated April 14, 1978 entitled: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, with Addendum dated January 18, 1979 (NRC Generic Letters (GLs) 78-11 and 79-04, respectively).

4. Standard Review Plan (SRP) 3.8.4, "Other Seismic Category I Structures," Appendix D, "Technical Position on Spent Fuel Racks," Revision 0, dated July 1981.

5. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NF, and Appendix F, 1998 Edition.

6. Holtec International Drawing 4732, "Rack A Support Platform," Revision 4, Sheets 1-5.

7. Black & Veatch Drawing 142707-1BSA-S6002, "Platform B/Plan & Sections," Revision 2, dated April 25, 2006.

8. Letter from S. Minahan (NPPD) to NRC, "Response to Request for Additional Information Regarding License Amendment Request for Onsite Spent Fuel Storage Expansion/Cooper Nuclear Station, Docket No. 50-298, DPR-46," dated April 17, 2007.

9. Burns and Roe Drawing 4228, "Structural Reactor Building Fuel Storage Pool Plan & Elevations," Revision 11, dated January 23, 1971.

10. Burns and Roe Drawing 4230, "Structural Reactor Building Misc. Sects & Dets Sh. #1 ," Revision 14, dated August 11, 1971.

11. Burns and Roe Drawing 4288, "Structural Reactor Building / I. F. 300 Cask Support - Plan, Sect. & Det'l," Revision 1, dated January 23, 1971.

12. SRP 3.7.1, "Seismic Design Parameters," Revision 1, dated July 1981.

13. SRP 3.8.5, "Foundations," Revision 1, dated July 1981.

14. SRP 3.8.4, "Other Seismic Category I Structures," Revision 1, dated July 1981.

15. GL 80-113, "Control of Heavy Loads," dated December 22, 1980.

16. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980.

2.4 Thermal-Hydraulic Considerations and Handling of Heavy Loads.

2.4.1 Regulatory Guidance

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," 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 reactor core, safety-related equipment, and equipment needed for decay heat removal.

NUREG-0612 endorses a defense-in-depth approach for handling of heavy loads near spent fuel and safe shutdown systems. General guidelines for overhead handling systems that are used to handle heavy loads in the area of the reactor vessel and SFP are given in Section 5.1.1 of NUREG-0612.

Section 5.1.2 of NUREG-0612 provides additional guidelines for control of heavy loads in the spent fuel pool area of pressurized-water reactors. Recommended supplemental actions include either using a single-failure proof handling system or evaluate the effects of a drop against the criteria of Section 5.1 of NUREG-0612. Appendix A of NUREG-0612 includes guidelines for evaluating the effects of load drops.

Appendix A of 10 CFR Part 50, GDC 61, specifies, in part, that fuel storage systems shall be designed with residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat removal, and with the capability to prevent significant reduction in fuel storage coolant inventory under accident conditions.

2.4.2 Thermal Considerations

The proposed new racks will be separated from each other by a gap of approximately 23 inches. The smallest gap between the new racks and the walls of the SFP will be 10 1/16 inches. The smallest gap between the new racks and the nearest structural member will be 3 29/32 inches. There will be at least 27 inches between the new racks and the existing racks.

The fuel pool cooling (FPC) system consists of two parallel cooling pumps that circulate SFP water through two parallel heat exchangers. Cross-tie piping allows the output of either pump to be directed to either or both of the FPC heat exchangers. SFP water is circulated through the tubes and heat is transferred to component cooling water circulating through the shell side. During a worst-case single active-failure condition, a single FPC pump would supply water to both FPC heat exchangers.

There are two postulated refueling offloads defined: partial core offload and full core offload. In a partial core offload, between 160 and 250 fuel assemblies are discharged from the reactor into the SFP at the end of a normal operating cycle. A single FPC pump supplying both FPC heat exchangers operates to provide cooling during the partial core offload. In a full core offload, the entire core of 548 fuel assemblies is discharged from the reactor into the SFP at the end of a normal operating cycle. For the full core offload, both FPC pumps supplying both FPC heat exchangers operate to provide cooling prior to the start of transfer. Once fuel transfer starts, cooling is provided by one train of the residual heat removal system operating in FPC Assist mode.

With the addition of two new racks, the SFP cooling system will be required to remove an increased heat load while maintaining the pool water temperature at or below the design limit of 150 °F bulk-water temperature. The SFP thermal performance and criticality response were reanalyzed by the licensee considering the increased storage capacity. Prior to offloading the spent fuel, the licensee determines the minimum in-core hold time required to ensure that the pool water temperature will remain at or below the design limit of 150 °F bulk-water temperature. The licensee stated that Holtec International prepared a thermal analysis that bounds the proposed SFP expansion. The Holtec report includes an evaluation of 25 different scenarios. The result of the analyses demonstrates that by applying procedural controls and determining the required in-core hold time before core offloading, the licensee can ensure that the bulk temperature limits are not exceeded.

If there is a complete loss of forced cooling, the SFP bulk-water temperature will begin to rise and will eventually reach the boiling temperature. The Holtec report includes analyses that calculated the minimum time to boil and the maximum boil-off rate. The time-to-boil evaluation assumed that forced cooling was lost the moment the peak SFP bulk temperature was reached. The SFP time to boil and corresponding maximum boil-off rates were then determined. For the worst-case scenario, the calculated time to boil was determined to be 4.19 hours after a loss of forced cooling; at current conditions, the time to boil is 5 hours. The new time to boil of 4.19 hours still provides sufficient time for the operators to align and start the addition of makeup water or take any remedial actions required.

The corresponding maximum boil-off rate for this condition was determined to be about 68 gallons per minute. The required makeup can be provided by multiple seismically qualified makeup water sources, all of them capable of providing more than the minimum required makeup water flow, e.g., the Reactor Building Service Water, condensate storage tank, residual heat removal system cross-tie, and the suppression pool.

Based on its review of the information provided by the licensee, the NRC staff concludes that there is adequate cooling water flow to the SFP heat exchanges to remove the decay heat generated by the increased number of spent fuel assemblies in the pool during normal and abnormal offload conditions. The use of procedural controls will prevent SFP water bulk temperature to rise above the limit of 150 ^{o}F. The staff also finds that the licensee has sufficient time and capability, prior to the onset of boiling, to align makeup water to the pool, and provide makeup at a rate in excess of the boil-off rate, thus satisfying GDC 61 with respect to maintaining the fuel covered with water under accident conditions.

2.4.3 Handling of Heavy Loads

The Reactor Building (RB) crane is an electric motor-driven overhead crane with a 100-ton-rated capacity and is controlled from a traversing cab. The crane is controlled either in the “normal” or “restricted” modes. In the “restricted” mode, interlocks limit crane speed to 18.5 feet per minute and limit switches restrict the path of travel. The crane spans the east/west walls of the RB and has two hoisting systems, the main hoist and the auxiliary hoist. The main hoist (rated for 100-ton capacity) will be principally used for the installation of the racks. The auxiliary hoist (rated for 5-ton capacity) will be used for moving smaller items.

The RB crane has been designed to prevent dropping or losing control of the heaviest load to be handled. While the hoist system design is predicated upon a dual-load path, some items within the path cannot be made redundant. Where full redundant features are not feasible or are impractical or impossible, increased design safety factors are used.

In its Safety Evaluation Report dated February 28, 1977, the NRC staff concluded that the licensee’s RB crane met the requirements of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” for loads of 70 tons or less. The licensee has stated that the maximum load to be handled during the installation of the new racks is less than 15 tons. NUREG-0612 states that if a licensee is using a single-failure proof crane (or equivalent), the licensee is not required to evaluate the effects of a load-drop event. To ensure the proper handling of heavy loads, NUREG-0612 provides guidelines for a defense-in-depth approach. Following these guidelines, the licensee identified their defense-in-depth approach as follows:

a. Safe Load Paths and Procedures

Safe load paths will be defined for moving the new racks into the RB. The racks will be lifted by the main hook of the RB crane and enter the laydown/staging area through the equipment hatch. The rack will enter the building at a location close to the laydown/staging area adjacent to the Cask Pit. The staging area location also will not require any heavy loads to be lifted over the pools or any safety-related equipment.

b. Supervision of Lifts

Procedures used during the installation of the racks require supervision of heavy load lifts by a designated individual who is responsible for ensuring procedural compliance and safe lifting practices. Holtec personnel experienced in similar rack installations will supervise the initial installation of the racks.

c. Crane Operator Training

CNS staff involved in the use of the lifting and upending equipment will be given training by Holtec International using a videotape-aided instruction course that has been utilized by Holtec in previous rack installation operations.

d. Lifting Devices Design and Reliability

The RB crane can access the equipment hatch, the adjacent laydown area, and the Cask Pit. The RB crane has sufficient capacity to handle the heavy load lifts during the new rack installing process.

A remotely engaging lift rig, meeting applicable guidelines of NUREG-0612, will be used to lift the rack modules. The rack-lift rig consists of four independently loaded traction rods in a lift configuration. The individual lift rods have a safety factor of greater than 10. If one of the rods breaks, the load will still be supported by at least two rods, and this will have a safety factor of more than 5 against ultimate strength. The lift rigs comply with the duality feature called for in Section 5.1.6(3) of NUREG-0612.

e. Crane Maintenance

The RB crane is maintained functional per NPPD's preventive maintenance procedures.

Additionally, NUREG-0612 guidelines cite four major causes of load-handling accidents: operator errors, rigging failure, lack of adequate inspection, and inadequate procedures. The licensee included in its submittal the proposed measures specifically planned to deal with the major causes of load handling accidents. These measures are:

> Operator errors: Comprehensive training in compliance with ANSI B30.2 will be provided to the installation crew.
>
> Rigging failure: The lifting device designed for handling and installing the new racks has redundancies in the lift legs and lift eyes such that there are four independent load members in the new rack-lift rig. Failure of any one load bearing member would not result in dropping the load. The rig complies with all provisions of ANSI Standard N14.6-1993, including compliance with the primary stress criteria, load testing at 300 percent of maximum lift load, and dye-penetrant examination of critical welds. The design of the lift rig is similar to that approved by the NRC and used in the initial rack installation or rack replacement at other plants, including Hope Creek, Millstone Unit 1, Indian Point Unit 2, FitzPatrick, Three Mile Island Unit 1, Callaway, and Wolf Creek.
>
> Lack of adequate inspection: The designer of the racks has developed a set of inspection points that have been proven to eliminate any incidence of rework or erroneous installation in numerous prior rerack projects. Surveys and measurements are performed on the storage racks prior to and subsequent to placement into the pool to ensure that the as-built dimensions and installed locations are acceptable. Measurements of the platform level are performed to ensure that the racks will be level after installation with minimum manipulation during placement into the pool. Preoperational crane testing will verify proper function of crane interlocks prior to rack movement.
>
> Inadequate procedures: Procedures will be developed to address rack installation, including, but not limited to, mobilization, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA (as low as reasonably achievable) compliance. The procedures will reflect the procedures successfully implemented in previous projects.

Based on its review of the information provided by the licensee, the NRC staff finds the licensee has provided adequate assurance that their planned actions for the handling of heavy loads for the installation of the new storage racks are consistent with the defense-in-depth approach to safety described in NUREG-0612.

2.4.4 Conclusions

Based on the considerations discussed above in section 2.4, the NRC staff concludes that there is adequate cooling water flow to the SFP heat exchanges to remove the decay heat generated by the increased number of spent fuel assemblies in the pool during normal and abnormal offload conditions. The use of procedural controls will prevent SFP water-bulk temperature to rise above the limit of 150 °F. The staff also finds that the licensee has sufficient time and capability, prior to the onset of boiling, to align makeup water to the pool, and provide makeup at a rate in excess of the boil-off rate, thus satisfying GDC 61 with respect to maintaining the fuel covered with water under accident conditions. Additionally, based on the review of the licensee's submitted information on the handling of heavy loads associated with this amendment request, the staff finds the licensee has provided adequate assurance that their planned actions for the handling of heavy loads for the installation of the new storage racks are consistent with the "defense-in-depth" approach to safety described in NUREG-0612.

Therefore, the staff finds the amendment request acceptable in regards to the SFP thermal-hydraulics, and the handling of heavy loads.

2.5 Health Physics Review

The NRC staff reviewed the radiological impact of the proposed change to assure that its design and use were in accordance with ALARA principles to minimize radiological exposure, consistent with the requirements of 10 CFR Part 20.

2.5.1 Occupational Radiation Exposure

The NRC staff reviewed the licensee's plan for installation of the new storage racks with respect to occupational radiation exposure.

The licensee has stated that the work required to install the new racks will be to clean and vacuum the cask pit, remove underwater appurtenances, and install new racks. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations and consistent with other plants' experience with rack installations, the licensee estimates that the proposed fuel rack project can be performed for between 1.1 and 2.2 person-roentgen equivalent man (rem) collective occupational worker dose.

The licensee states that all of the operations involving the installation of the new fuel racks will be governed by procedures. These procedures were prepared with full consideration of ALARA principles, consistent with the requirements of 10 CFR Part 20. The Radiation Protection department will prepare a Radiation Work Permit (RWP) for the various jobs associated with the in-pool and out-of-pool operations. The RWP and supporting job procedures establish requirements for timely external radiation and airborne surveys, personal protective clothing and equipment, individual monitoring devices, and other access and work controls consistent with good radiation protection practices and 10 CFR Part 20 requirements. Each member of the project team will receive radiation protection training to ensure an understanding of critical evolutions.

For out-of-pool work activities, all workers will be provided with thermoluminescence dosimeters (TLD) and electronic alarm dosimeters. Additional personal monitoring devices (e.g., extremity badges) will be used, as appropriate. Periodic radiation surveys will be conducted for direct radiation levels and loose surface contamination levels, as appropriate and in accordance with the governing RWP. Previous historical experience during similar rack installations shows that radioactive airborne material levels in the above-pool work area should be negligible. However, air sampling will be performed, and continuous air monitors will be used when a job evolution has the potential for generating significant airborne radioactivity.

Diving operations in the SFP to prepare for placement of the additional racks were completed in August 2006. The licensee states that, at this time, there are no planned diving operations in the SFP. However, should the need arise for additional diving operations for the CNS spent fuel pool rack installation project, qualified underwater divers will be used. The sources of high radiation that may be in the SFP during diving operations for minor modification of the beam segments are the spent fuel assemblies stored in the existing racks, used control blades, and several filters from previous vacuuming operations stored in the northwest corner of the SFP. During diving operations, no spent fuel or other highly radioactive components shall be moved. To ensure that these divers do not gain access to high and very high radiation sources (e.g., spent fuel), all diving operations will be governed by procedures. These procedures will require a minimum separation of 10 feet to be maintained between the diver and any fuel spent fuel assembly, control equipment, or irradiated component, a “safe dive zone” will be established to ensure that the diver is protected from coming in contact with the fuel assemblies or components, highly visible physical boundaries are used in the areas of the SFP containing highly radioactive components, and a briefing is required prior to starting diving operations. Continuous monitoring of radiation levels in the dive zone and dose rates to the diver will be communicated to the diver to allow for constant pool-side radiation surveillance of all diver activities. Each diver will be provided with multiple TLDs and electronic dosimeters for whole body and extremity monitoring, with remote read-out capabilities for pool-side observation, monitoring, and control. The CNS diving control and survey procedures described above meet the intent of NRC Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants”, Appendix A, "Procedures For Diving Operations In High and Very High Radiation Areas." This Appendix was developed from the lessons learned from previous diver overexposures and mishaps, and summarizes good operating practices for divers acceptable to the NRC staff.

The licensee states that an underwater vacuum system will be used to supplement the installed spent fuel pool filtration system, so that radiation/contamination levels (including hot particles and debris) can be reduced before diving operations. The SFP floor dive area will be vacuum cleaned using long-handled tools from above the pool. Final radiation surveys and visual inspection (by underwater camera) will be performed prior to any diving activities. These hot particle/debris identification/control actions should effectively minimize the potential for unplanned diver exposures from these sources as well as to assist in the restoration of SFP clarity following installation of the new racks.

Prior to installation of the new racks, the drum platform will need to be removed. As the drum platform is removed from the cask pit area in the SFP, it will be rinsed as it breaks the surface of the SFP by spraying demineralized water during removal to minimize airborne concentrations. Once removed, the drum platform will be covered in plastic to minimize airborne contamination. The licensee states that, once properly packaged in approved shipping containers, the racks will be shipped in accordance with Department of Transportation and NRC regulations. To address the extremely high-dose rates due to filling the new racks completely with freshly discharged fuel, the licensee committed in its supplemental letter dated April 17, 2007, that, “Two rows of 5-year cooled fuel will be placed along the sides of the new racks facing the fuel pool walls to provide shielding from freshly discharged fuel assemblies. The procedure for controlling storage of spent fuel in the spent fuel pool will be revised to require the placement of two rows of 5-year cooled fuel.” With this commitment of placing 5-year old decayed fuel in the two outer rows along the sides of the new fuel racks facing the pool walls, the licensee has calculated the maximum dose rate on contact with the surface of the SFP wall to be less than 2 millirem per hour.

Based on the information provided by the licensee, the NRC staff concludes that the SFP rack installation can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds the projected dose for the project of about 1.1 to 2.2 person-rem to be reasonable and in the range of doses for similar SFP modifications at other plants and, therefore, acceptable.

2.5.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP purification system. The licensee predicts that on a one-time basis only a very small amount of additional resin will be generated from the new, increased capacity rack installation; therefore, the change-out frequency of the SFP purification system may increase slightly during the period of the new rack installation. Because the installation of the new racks will not significantly introduce a large volume of solid radioactive waste, the impact to solid radioactive waste from installation is minimal. The licensee does not expect that increasing the storage capacity of the SFP will result in a significant change in the long-term generation of solid radioactive waste at CNS. The NRC staff concurs with the licensee's assessment, and, therefore, finds the proposed addition of the new racks acceptable.

2.5.3 Gaseous Radioactive Wastes

The storage of additional spent fuel assemblies in the SFP is not expected to affect the release of radioactive gases from the SFP. Gaseous fission products such as Krypton-85 and Iodine-131 are produced by the fuel in the core during reactor operation. A small percentage of these fission gases are released to the reactor coolant from the small number of fuel assemblies that are expected to develop leaks during reactor operation. During refueling operations, some of these fission products enter the SFP and are subsequently released into the air. Since the frequency of refueling (and therefore the number of freshly offloaded spent fuel assemblies stored in the SFP at any one time) will not increase, there will be no increase in the amounts of these types of fission products released to the atmosphere as a result of the increased SFP fuel storage capacity.

The increased heat load on the SFP from the storage of additional spent fuel assemblies could potentially result in an increase in the SFP evaporation rate. However, this increased evaporation rate is not expected to result in any significant increase in the amount of gaseous tritium released from the pool. This has not been an operational problem with any previous rack installations at other facilities.

Therefore, the licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to significantly increase due to the expanded SFP storage capacity. This is consistent with the operating experiences to date with previous SFP expansions. Gaseous effluents from the spent fuel storage area are combined with other station exhausts, and monitored before release. Past SFP area contributions to the overall site gaseous releases have been insignificant, and should remain negligible with the increased capacity. The impact of any increases in site gaseous releases should be considered negligible, and the resultant doses to the public will remain a very small fraction of 10 CFR Part 20 and 10 CFR Part 50, Appendix I dose limits. The NRC staff concurs with the licensee's assessment, and, therefore, finds the proposed addition of the new racks acceptable.

2.5.4 Liquid Radioactive Wastes

The release of radioactive liquids will not be directly affected as a result of the SFP expansion. The SFP ion exchanger resins remove soluble radioactive materials from the SFP water. When the resins are changed out, the small amount of resin sluice water is processed by the radioactive waste system, before release to the environment. As stated above, the frequency of resin change-out may increase slightly during the installation of the new racks. However, the amount of liquid effluent released to the environment as a result of the proposed SFP expansion is expected to be negligible.

2.5.5 Radiological Impact Assessment

The licensee states that Radiation Protection personnel will monitor the doses to the workers during the SFP expansion operation, and all work will be in accordance with RWPs and implementing procedures. If needed, divers will be used for the SFP racking operations and the licensee will provide procedures specifying required survey, personal dosimetry, and other work requirements and controls that meet the intent of Regulatory Guide 8.38, Appendix A guidance. The total occupational dose to plant workers as a result of the SFP expansion operation is estimated to be between 1.1 and 2.2 person-rem. This dose estimate is reasonable, given the work scope proposed, and is consistent with comparable doses for similar SFP projects performed at other plants. The SFP expansion project will follow detailed procedures prepared with full consideration of ALARA principles, consistent with the requirements of 10 CFR Part 20. The estimated collective dose to perform the proposed SFP racking operation is a small fraction of the annual collective dose accrued at the facility.

On the basis of the NRC staff's review of the licensee's proposal, as documented above in Section 2.4, the staff concludes that the SFP expansion can be performed in a manner that will ensure that doses to workers will be maintained ALARA.

3.0 REGULATORY COMMITMENTS

In its application dated October 17, 2006, as supplemented by letters dated February 7, April 17, and May 4, 2007, the licensee made the following regulatory commitments:

1. NPPD will develop a procedure implementing the coupon sampling program, as discussed in Attachment 4 of [its supplemental letter dated April 17, 2007], prior to installation of the Metamic™-poisoned spent fuel storage rack.

2. NPPD will obtain baseline data taken on the unirradiated Metamic™ coupons and submit that data to the NRC, prior to installing the coupon tree with the Metamic™ coupons.

3. NPPD will remove a coupon and perform testing and surveillance on the coupon after 2, 4, 8, 12, 16, 20, 24, and 28 years following initial placement of irradiated

fuel in the SFP, and will submit the results to the NRC, beginning with Operating Cycle 25 (approximately May 2008), after the following periods: 2 years + 6 months, 4 years + 6 months, 8 years + 6 months, 12 years + 6 months, 16 years + 6 months, 20 years + 6 months, 24 years + 6 months, and 28 years + 6 months.

4. Two rows of 5-year cooled fuel will be placed along the sides of the new racks facing the fuel pool walls to provide shielding from freshly discharged fuel assemblies. The procedure for controlling storage of spent fuel in the spent fuel pool will be revised to require the placement of two rows of 5-year cooled fuel, [prior to] placement of the new racks in the spent fuel pool.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 70561 and 72 FR 2560, published December 5, 2006, and January 19, 2007, respectively).

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has previously been prepared and published in the *Federal Register* on September 5, 2007 (72 FR 50988).

Based on the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant impact upon the quality of the human environment.

6.0 CONCLUSION

Based on its review of the (1) criticality considerations, (2) use of Metamic™ poison inserts, (3) seismic analysis and structural design, (4) thermal-hydraulic considerations and handling of heavy loads, and (5) health physics considerations of the licensee's proposed changes, the NRC staff finds the proposed changes acceptable.

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

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