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

February 28, 2013

Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT REGARDING REPLACEMENT OF SPENT FUEL POOL REGION I STORAGE RACKS RE: (TAC NO. ME8074)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 250 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment consists of changes to the PNP Technical Specifications (TSs) in response to your application dated February 28, 2012, as supplemented by letters dated September 6, November 7, November 29, 2012, February 21, and February 25, 2013.

The amendment revises the PNP TSs to support the replacement of the Region I main spent fuel (SFP) storage racks and the storage racks in the north tilt pit (NTP) portion of the SFP, with new neutron absorber Metamic-equipped racks. The replacement of the SFP storage racks will allow recovery of the currently unusable storage locations in the SFP.

Document transmitted herewith contains Proprietary Information. When separated from Enclosure 3, this document is decontrolled.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to the Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a redacted publicly available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 250 to DPR-20
2. Non-Proprietary Safety Evaluation
3. Proprietary Safety Evaluation

cc w/ encl. 1 and 2: Distribution via ListServ

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

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 250
License No. DPR-20

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee), dated February 28, 2012, as supplemented by letter(s) dated September 6, November 7, November 29, 2012, February 21, and February 25, 2013 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-20 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 250, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: February 28, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 250

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Replace the following page of the Renewed Facility Operating License No. DPR-20 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

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

REMOVE	INSERT
Page 3.7.16-1	Page 3.7.16-1
Page 4.0-1	Page 4.0-1
Page 4.0-4	Page 4.0-4
Page 4.0-5	Page 4.0-5

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENP to possess and use, and (b) ENO to possess, use and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) ENO, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ENO, 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 the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; 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) ENO is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 250, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated 09/01/78, 03/19/80, 02/10/81, 05/26/83, 07/12/85, 01/29/86, 12/03/87, and 05/19/89 and subject to the following provisions:

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Storage

LCO 3.7.16 Storage in the spent fuel pool shall be as follows:

a. Each fuel assembly and non-fissile bearing component stored in a Region I Carborundum equipped storage rack shall be within the limitations in Specification 4.3.1.1 and, as applicable, within the requirements of the maximum nominal planar average U-235 enrichment and burnup of Tables 3.7.16-2, 3.7.16-3, 3.7.16-4 or 3.7.16-5,

b. Fuel assemblies in a Region I Metamic equipped storage rack shall be within the limitations in Specification 4.3.1.2, and

c. The combination of maximum nominal planar average U-235 enrichment, burnup, and decay time of each fuel assembly stored in Region II shall be within the requirements of Table 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly or non-fissile bearing component is stored in the spent fuel pool or the north tilt pit.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Initiate action to restore the noncomplying fuel assembly or non-fissile bearing component within requirements.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means each fuel assembly or non-fissile bearing component meets fuel storage requirements.	Prior to storing the fuel assembly or non-fissile bearing component in the spent fuel pool

4.0 DESIGN FEATURES

4.1 Site Location

The Palisades Nuclear Plant is located on property owned by Entergy Nuclear Palisades, LLC on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor core shall contain 204 fuel assemblies. Each assembly shall consist of a matrix of zircaloy-4 or M5 clad fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution. Poison may be placed in the fuel bundles for long-term reactivity control.

4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rods. Four of these control rods may consist of part-length absorbers. The control material shall be silver-indium-cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The Region I (See Figure B 3.7.16-1) Carborundum equipped fuel storage racks incorporating Regions 1A, 1B, 1C, 1D, and 1E are designed and shall be maintained with:

a. New or irradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.54 weight percent;

4.3 Fuel Storage

4.3.1 Criticality (continued)

3. Control blades may be stored in both fueled and unfueled locations in Regions 1D and 1E, with no limitation on the number.

4.3.1.2 The Region I (See Figure B 3.7.16-1) Metamic equipped fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. A nominal 10.25 inch center to center distance between fuel assemblies;

e. New or irradiated fuel assemblies;

f. Two empty rows of storage locations shall exist between the fuel assemblies in a Carborundum equipped rack and the fuel assemblies in an adjacent Metamic equipped rack; and

g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2.

4.3.1.3 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having maximum nominal planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

4.3 Fuel Storage

4.3.1 Criticality (continued)

e. New or irradiated fuel assemblies which meet the maximum nominal planar average U-235 enrichment, burnup, and decay time requirements of Table 3.7.16-1.

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

a. Twenty four unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. Keff ≤ 0.95 when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 250 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-20

ENTERGY NUCLEAR OPERATIONS, INC.

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated February 28, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12061A288, ML12061A289, and ML12061A290, respectfully), Entergy Nuclear Operations, Inc.(the licensee), proposed a change which would revise the technical specifications (TSs) for the Palisades Nuclear Plant (PNP) to support the replacement of the Region I main spent fuel pool (SFP) storage racks and the storage racks in the north tilt pit (NTP) portion of the SFP. The licensee supplemented its February 28, 2012, submittal, by letters dated September 6, 2012 (ADAMS Accession No. ML122541048), November 7, 2012 (ADAMS Accession No. ML12313A343), and November 29, 2012 (ADAMS Accession No. ML12335A071), in response to NRC staff's request for additional information (RAI).

The present SFP Region I fuel storage racks neutron absorbing material Carborundum® is degraded and is not credited as a neutron absorber. The purpose of this change is to allow the replacement of the degraded Carborundum®-equipped racks in Region I with new neutron absorber Metamic™-equipped storage racks. The replacement of the SFP storage racks will allow recovery of the currently unusable storage locations in the SFP. When the replacement is completed, the replacement Region I storage racks will have the same number of storage cell locations as those of the existing storage racks. As such, the replacement storage capacity will not be altered by replacement of the Region I storage racks. Once the installation has been completed, the Region I portion of the SFP will contain only Metamic-equipped racks. Therefore, no change to the current plant radiation zoning design is expected. The present TS for Region I restrict the use of some storage cell locations due to the degradation of the Carborundum®-equipped storage racks. The replacement Metamic™-equipped storage racks will allow storage in all the available cell locations within Region I of the SFP.

A draft copy of the proprietary SE was sent to the licensee on February 15, 2013, via e-mail (ADAMS Accession No. ML13056A501), followed up by letter on February 15, 2013, to perform proprietary review of the SE. Licensee provided their response on February 21, 2013 (ADAMS Accession No. ML13053A078). During a subsequent review of the proprietary information, licensee determined that certain information previously identified as proprietary in the February 21, 2013, response is in fact nonproprietary. Therefore, as a result licensee withdrew the February 21, 2013, via letter dated February 25, 2013 (ADAMS Accession No. ML13056A603).

Enclosure 2

The supplemental letters dated September 6, November 7, November 29, 2012, February 21, and February 25, 2013, provided additional information that clarified the application, but did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 5, 2012 (77 FR 33246).

2.0 REGULATORY EVALUATION

2.1 Reactor Systems Review

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, Criterion 62, requires that:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

10 CFR 50.36 (c)(4) requires that:

> Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of this sections.

10 CFR Part 50, Section 68(b)(1) requires that:

> Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safety subcritical under the most adverse moderation conditions feasible by unborated water.

10 CFR Part 50, Section 68(b)(4) requires, in part, that:

> If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The NRC staff issued an internal memorandum on August 19, 1998, containing guidance for performing the review of SFP criticality analysis. This memorandum is known as the 'Kopp Letter,' after the author. The Kopp Letter provides guidance on salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized-water reactor, and to borated and unborated conditions. The staff used the Kopp Letter as guidance for the review of the current analysis.

On September 29, 2011, the NRC staff issued the Interim Staff Guidance (ISG) DSS-ISG-2010-1 (ADAMS Accession No. ML110620086). The purpose of the ISG is to provide updated review guidance to the NRC staff to address the increased complexity of recent SFP license applications. The NRC staff used ISG DSS-ISG-2010-1 for the review of the current application.

2.2 Mechanical and Civil Engineering Review

The review performed by the NRC staff covers the structural integrity of the systems, structures and components (SSCs) affected by the proposed license amendment request (LAR) to support replacement of the Region I main SFP racks and the NTP fuel storage racks with new racks equipped with the neutron absorber Metamic. The present SFP Region I fuel storage racks neutron absorbing material Carborundum® is degraded and not credited as a neutron absorber. The replacement of the SFP storage racks will allow recovery of the currently unusable storage locations in the SFP. This review focused on verifying that the licensee has provided reasonable assurance of the structural integrity of the SSCs affected by the proposed replacement at PNP under normal and abnormal loading conditions, including postulated accidents and natural phenomena such as earthquakes.

The NRC staff's acceptance criteria in the areas of civil and mechanical engineering are based on the regulatory requirements described in the Standard Review Plan (SRP or NUREG-0800), Section 3.8.4, "Other Seismic Category I Structures." The regulatory requirements described in SRP Section 3.8.4 include the following: Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, and 10 CFR Part 50, Appendix A, General Design Criteria (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; and GDC 4 as it relates to safety-related structures being protected against dynamic effects, such as the loads imposed on structures by postulated missiles. In addition to complying with the aforementioned regulations, continued compliance with the PNP design basis requirements must also be demonstrated, following the proposed replacement. Chapter 3 of the PNP updated final safety analysis report (UFSAR) provides the design basis requirements relative to the design of SSCs at PNP. Chapter 9 of the PNP UFSAR includes the design bases of the spent fuel storage SSCs at PNP.

The NRC staff's acceptance criteria specific to the design of spent fuel racks can be found in Appendix D of SRP Section 3.8.4, "Guidance on Spent Fuel Pool Racks." Acceptance criteria which are coupled to this SRP section are also found in SRP Section 3.7.1, "Seismic Design Parameters," and SRP Section 3.8.5, "Foundations." Additional regulatory guidance regarding the review and acceptance criteria for SFP storage racks is documented in Enclosure 1 to the NRC's letter to all licensees dated April 14, 1978,."Office of Technology (OT) Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (OT Position Paper)," as revised by letter dated January 18, 1979. These two letters were subsequently numbered NRC Generic Letters (GLs) 1978-11, "Review and Acceptance of Spent Fuel Storage and Handling Applications," and 1979-04, "Modifications to NRC Guidance on 'Review and Acceptance of Spent Fuel Storage and Handling Applications,'" respectively.

Chapter 3 of the PNP UFSAR indicates that the SFP liner, and associated structures, including the current SFP racks at PNP were designed against the criteria found in the SRP Section 3.8.4. In response to the staff RAI regarding the use of SRP Section 3.8.4 in support of the proposed LAR, the licensee indicated that Revision 2 of SRP Section 3.8.4 had been utilized. However, the licensee indicated that the design and analysis of the replacement SFP racks at PNP is not impacted by the change of the SRP Section 3.8.4, Revision 3. The NRC staff reviewed the licensee's response and found the structural acceptance criteria for the SFP structures remain the same when comparing Revision 2 and Revision 3 (the most recent

revision) of SRP Section 3.8.4 (including the Appendix D portion of SRP Section 3.8.4). The staff considers the licensee's use of Revision 2 of SRP Section 3.8.4 acceptable, with respect to the proposed LAR at PNP, based on the fact that the structural acceptance criteria related to spent fuel storage SSCs are unaffected by the use of the Revision 2 of this SRP section.

The NRC has issued similar license amendments for SFP re-racking requests for Cooper Nuclear Station on September 6, 2007 (ADAMS Accession No. ML072130023), Arkansas Nuclear One, Unit 2, on September 28, 2007 (ADAMS Accession No. ML072620412), and the Beaver Valley Power Station, Unit 2, on April 29, 2011 (ADAMS Accession No. ML110890844).

2.3 Balance of Plant Review

The SFP is described in the PNP final safety analysis report (FSAR), Section 9.11, "Fuel Handling and Storage Systems." The SFP is located in the auxiliary building and connected to two adjacent fuel tilt pits by canals. The south fuel tilt pit is used for normal fuel transfer activities during refueling and the north fuel tilt pit has been modified for spent fuel storage. The SFP and north fuel tilt pit are divided into two regions for spent fuel storage. Region I spent fuel storage racks contain the neutron absorber Carborundum™. Due to concerns about the degradation of this material, credit for the neutron absorber was removed from the criticality calculations for Region I spent fuel storage racks. In order to ensure the criticality requirements of 10 CFR 50.68 were met, credit was taken for soluble boron, the storage pattern in Region I was controlled, and specific storage cells must be left empty.

Installation and removal of spent fuel storage racks are heavy-load lifts and will be accomplished using the fuel building crane, described in FSAR Section 9.11.4, "Fuel Handling System." The fuel building crane was approved for use as a single-failure-proof crane in accordance with the guidance of NUREG-0612 for loads up to 110 tons.

Decay heat from spent fuel stored in the SFP is removed by the SFP cooling system (SFPCS), described in FSAR Section 9.4. The SFPCS is designed to maintain the SFP temperature below 150 degrees Fahrenheit (°F) at all times. Makeup water for the SFP is available from the primary system makeup storage tank, the recycled boric acid storage tank, and the safety injection and refueling water tank.

The licensee has proposed changes to the TS that will allow the replacement of all Region I storage racks with new racks containing the neutron absorber Metamic™. Replacement of the Region I storage racks will allow the licensee to reclaim the storage cells that are currently prohibited by the criticality analysis.

This amendment request makes no change to the total number of spent fuel assemblies that can be stored in the SFP.

Criterion 61 of Appendix A to 10 CFR 50, "Fuel Storage and Handling and Radioactivity Control," requires, in part, that fuel storage and handling systems shall be designed to assure adequate safety and provide residual heat removal (RHR) capability with "reliability and testability that reflects the importance to safety of decay heat removal." PNP, Unit 1, FSAR, Section 5.1.7.2, discusses the plant's conformance to this criterion. The NRC staff evaluated the proposed changes to ensure that replacement of the Region I storage racks would not impact the ability to provide adequate RHR as described in FSAR, Sections 5.1.7.2 and 9.4.

"NRC OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, provides guidance for the content of license amendments that make changes to the storage of spent fuel. The guidance suggests that amendments include thermal hydraulic analyses to determine the increase in temperature of the SFP, the heat up rate following a loss of cooling, and to demonstrate that the storage racks will remain water filled. The proposed amendment states that the thermal hydraulic analyses meet the guidance of the NRC OT Position Paper.

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," contains guidance to assure the safe handling of heavy loads. FSAR Section 9.11.4 describes the conformance of the fuel building crane to this guidance. The NRC staff evaluated the removal and installation of spent fuel storage racks using the guidance of NUREG-0612 to ensure that heavy loads would be safely handled in and around the SFP.

2.4 Health Physics and Human Performance Review

The regulatory requirements and guidance which the NRC staff considered in its review of the LAR are as follows:

2.4.1 10 CFR 50.120, "Training and qualification of nuclear power plant personnel"

2.4.2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition":

Chapter 13 addresses "Conduct of Operation", specific sub-chapters considered in this review were Chapters 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training", and 13.5.2.1, "Operating and Emergency Operating Procedures".

Chapter 18 provides review guidance for "Human Factors Engineering."

2.4.3 NUREG-1764, "Guidance for the Review of Changes to Human Actions."

2.4.4 GL 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability."

2.4.5 NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2;

2.4.6 NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2;

2.4.7 IN 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times,"

2.5 Steam Generator Tube Integrity and Chemical Engineering Review

The following is the regulatory basis for the use of Metamic™ as a neutron absorber in the SFP:

> General Design Criteria 62, "Preventing of criticality in fuel storage and handling," states:
>
> *Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.*

The Palisades Nuclear Plant operating license precedes the publication of the Atomic Energy Commission (AEC) "General Design Criteria for Nuclear Power Plants" (10 CFR 50, Appendix A, February 20, 1971). As such, the Final Safety Analysis Report (FSAR) lists criterion reflecting the design intent for this nuclear power plant in consideration of the General Design Criteria (GDC) for Nuclear Power Plants. The FSAR Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," states:

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

SRP, Section 9.1.2, "Spent Fuel Storage," states:

The staff's review should ensure the compatibility and chemical stability of the materials wetted by the water in the SFP and, if applicable, in the new fuel vault, and evaluate potential mechanisms that alter the dispersion of any strong fixed neutron absorbers:

A. Compatibility and chemical stability of the materials in the components wetted by water in the spent fuel pool and in the new fuel vault. If the possibility for corrosion mechanisms is detected, the existing programs for preventing or minimizing corrosion are reviewed for their applicability to control corrosion.

B. The reactivity of fuel in the spent fuel pool is controlled by plates or inserts attached to spent fuel racks containing neutron poison dispersed in a matrix. In some environments, the matrix may degrade and release the neutron poison, resulting in some reduction of neutron absorbing properties of the panels. The licensee should have a program for monitoring the effectiveness of the neutron poison present in the neutron absorbing panels.

2.6 Accident Dose Review

The NRC staff evaluated the radiological consequences of the postulated design basis accidents against the dose criteria specified in 10 CFR Section 50.67, "Accident source term," and using the guidance described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The requirements of 10 CFR 50.67 state that the applicable dose criteria are 5 rem total effective dose equivalent (TEDE) in the control room, 25 rem TEDE at the exclusion area boundary and 25 rem TEDE at the outer boundary of the low population zone. RG 1.183 provides guidance to licensees on acceptable application of alternative source term submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST. The NRC staff also considered relevant information in the PNP FSAR and TSs.

3.0 TECHNICAL EVALUATION

3.1 Reactor Systems Branch Review

3.1.1 Selection of Bounding Fuel Assembly

The PNP uses 15x15 pressurized-water reactor fuel. The core was initially manufactured by Combustion Engineering, and subsequent reloads were manufactured by AREVA. Two main

fuel designs have been used in the core, which are fuel assemblies before "Batch R" and fuel assemblies from Batch R and beyond. In order to find the bounding assembly, the criticality analysis used one assembly from each design, which contains an initial nominal planar average enrichment of 5.0 wt% 235U, to bound the possible enrichments and different fuel types in the storage racks. To determine the design basis assembly, MCNP5-1.51 calculations are performed with a single cell MCNP5-1.51 Region I model, with reflective boundary conditions down the center of the flux trap. Table 4.5.1 of HI-2115004 shows the fuel assembly specifications for both assembly designs. Table 4.6.1 of HI-2115004 show the results of the analysis conducted with pure water, 850 ppm borated water, and 1720 ppm borated water. The fuel assembly used to represent Batch R and beyond is the bounding fuel assembly and is used as the design basis fuel assembly.

The method used to determine the limiting assembly follows the guidance set forth in DSS-ISG-2010-01.

3.1.2 Composition of Metamic

Metamic was developed in the mid-1990's by the Reynolds Metals Company for spent fuel reactivity control in dry and wet storage applications, and its performance lifespan is designed for 60 years. It is a metal matrix composite, consisting of a matrix of aluminum reinforced with ASTM C-750 boron carbide. Metamic is characterized by extremely fine aluminum and boron carbide powder. The average boron carbide particle size is between 10 and 40 microns. The panels will be on all four sides of the fuel cell by die forming sheathings. The sheathings are attached by welds to the cell. The absorber width is 6 and 7/8 inches and the absorber length is 134 inches. Metamic is also porosity free. In all cases, an uncertainty of 5 percent on the minimum areal density of the boron carbide amount in the Metamic (which is $0.02944 g/cm^2$) is included in the uncertainty analysis. The NRC finds the treatment of Metamic in the uncertainty calculations to be acceptable.

3.1.3 Criticality Analysis

The licensee's criticality analysis of the Region I SFP storage racks credits Metamic fixed neutron absorbers, as well as, soluble boron for normal and accident conditions in accordance with 10 CFR 50.68(b)(4). The licensee's criticality analysis control of the Region I SFP storage racks does not credit burnup of the fuel nor integral or non-integral absorbers such as gadolinium.

3.1.4 Spent Fuel Pool Water Temperature

Section 4.2.3.2 of HI-2115004 discuss the reactivity effect of water temperature in the SFP. The pool has a normal operating temperature of 68 -150 °F. Per the guidance issued in the Kopp Letter memorandum, the criticality analysis should be performed at the most reactive temperature and density. To determine the water temperature and density which result in the maximum reactivity, MCNP5-1.51 calculations were run using the bounding SFP temperature values in three cases for Region I, and in both pure water and 1720 ppm borated water conditions. Water temperature and density in Region II were also analyzed in order to establish interface interactions between Region I and Region II. In both regions, calculations were performed across a temperature range between 39.2 °F and 150 °F, and across a density range between 0.9803 and 1 g/cc of water.

The results of the SFP temperature and density calculations are shown in Table 4.6.2(a) of HI-2115004. For Region I, the most reactive water temperature and density is a temperature of 39.2 °F at a density of 1 g/cc with a k-eff of 0.9475. The results of the Region II SFP temperature and density calculations are shown in table 4.6.2(b) of HI-2115004 and show that the most reactive water temperature and density is a temperature of 150 °F at a density of 0.9803 g/cc with a k-effective of 0.9724. The NRC accepts the licensee's treatment of the water temperature.

3.1.5 Interface requirements

Three different interfaces were discussed and analyzed in: Region I to Region I gap, the effect of Region I to the reactivity of Region II, and the effect of Region II on the reactivity of Region I. In all three interfaces, the interface reactivity is less than the design basis reactivity. The results for all three cases were added as a bias and an uncertainty.

In the first case (Region I to Region I gap), the license used an 8x8 array of Region I storage cells with the minimum rack to rack water gap width. The 8x8 array is the size of one Region I storage module. Reflective boundary conditions were used down the center of the water gap. The temperature and density used in the model was the bounding temperature and density for the Region I racks.

In the second case (Region I to Region II), the license modeled three different scenarios. These scenarios were also evaluated assuming a SFP filled with pure water. The results are presented in Table 4.6.11.

In the third case (Region II to Region I), the licensee first determined how many Region II racks should be included in the MCNP5-1.51 model, the results of which are in Table 4.6.12(a), Table 4.6.12(b), and Table 4.6.13. The most reactive array in Region II, the 4x19 array, was chosen for the calculation to determine the effect of Region II on Region I. For all three cases, the interface reactivity is found to be less than the design basis reactivity.

3.1.6 Normal Conditions

Fuel movement, inspection, and reconstitution are considered normal operation. The bounding case is the fuel elevator region which can hold two fuel assemblies, one in the elevator and one in the inspection station. Three cases were evaluated to determine the effect of the two fuel assemblies in pure water. The first case is the design basis fuel assembly described above. The second case uses an 8x16 array of Region I storage cells, with a 2x6 array cut out in order to account for the fuel elevator region. This array accounts for the two 8x8 modules in the SFP that are affected by the fuel elevator, and does not model a fuel assembly in the elevator region. The third case is the same model as the second case but models the fuel assemblies in the elevator region. The bounding temperature and density described above is used in the calculations. The results of the calculations are presented in Table 4.6.10 of HI-2115004. In all cases, the fuel assemblies in the elevator region were bounded by the design basis case of 0.9475.

The manufacturing tolerances of the storage racks and fuel assemblies contribute to reactivity. Determination of the maximum k-effective should consider either (1) a worst-case combination with mechanical and material conditions set to maximize k-effective or (2) a sensitivity study of

the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant tolerance variations in the material and mechanical specifications of the racks.

The licensee's analysis evaluated the following uncertainty components: fuel pellet density, fuel pellet outer diameter, fuel rod pitch, fuel cladding inner diameter, fuel cladding outer diameter, instrumentation tube inner diameter, instrumentation tube outer diameter, guide bar width, storage cell inner diameter, storage cell wall thickness, eccentric fuel positioning, storage cell pitch, sheathing thickness, and poison pocket thickness. The analysis assumed a bounding pellet density of 96 percent theoretical density. As stated above, the minimum Metamic panel areal density with a 5 percent uncertainty is used in the reactivity calculations.

To determine the delta k associated with a specific manufacturing tolerance, the licensee used MCNP5-1.51 to calculate the k-effective for both pure water and borated cases. All tolerance perturbations were applied in the direction that increases reactivity relative to the nominal condition. Based on the considerations discussed above, the NRC staff finds the applicant's treatment of manufacturing tolerances acceptable.

3.1.7 Abnormal and Accident Conditions

Section 4.2.8 of HI-2115004 provides information on abnormal and accident conditions that are considered in the analysis. The licensee has considered the following abnormal conditions:

- Increased water temperature
- Horizontal dropped assembly
- Vertical dropped assembly
- Storage cell distortion
- Mislocated fuel assembly
- Boron dilution
- Rack movement

The licensee determined that the limiting abnormal condition was the mislocated fuel assembly at the inspection station in the elevator region. The licensee identified the worst-case locations for a mislocated fuel assembly to be in the elevator region, and in the NTP between Region I and Region II racks. The evaluation for these cases are contained in Tables 4.6.17(a), 4.6.17(b), 4.6.18(a), and 4.6.18(b) of HI-2115004.

The licensee considered two cases when analyzing temperature effects on the SFP, the lower bound temperature where the water temperature is at 39.2 °F and 1 g/cc and the upper bound temperature where the water temperature is at 255 °F and 0.8459 g/cc. In the second case, the model assumes the water is at 10 percent void. Table 4.6.14 in HI-2115004 shows that increased water temperature does not result in an increase in reactivity.

In the case of a horizontal and vertical dropped assembly, the licensee considered three cases each using bounding temperature and density for the Region I racks. Table 4.6.15 of HI-2115004 show that the effect of this accident on reactivity is bounded by the mislocated fuel assembly accident.

Storage cell distortation or altered geometry as a result of a dropped fuel assembly or by fuel handling uplift forces is possible. The damage to the top of a Metamic panel would be less

than 1.5 inch, and there would be no plastic deformation in the racks due to handling uplift forces. These results are evaluated in Chapter 7 of HI-2115004. These accidents are bounded by the mislocated fuel assembly accident.

Rack movement due to seismic activity was analyzed, and the result is provided in Table 4.6.19 of HI-2115004. The effect of this accident on reactivity is bounded by the mislocated fuel assembly accident.

The regulatory requirement is that the k-effective of the SFP racks, loaded with fuel of the maximum fuel assembly reactivity, must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water. The TSs require the normal fuel pool boron concentration to be at least 1720 part per million (ppm), which is an 870 ppm margin to the 0.95 limit for PNP. The deboration event was analyzed and was shown that a large volume of water and 9.8 hours from time of dilution were necessary to dilute the SFP from its present TS limit to 850 ppm. The analysis demonstrated that the fuel storage racks will be subcritical under all accident conditions with 850 ppm soluble boron.

Based on the above, the NRC staff finds that the information supporting the criticality analysis is acceptable.

3.1.8 Criticality Code Validation

The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainty are determined for use in the criticality calculation. ISG DSS-ISG-2010-1 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology."

NUREG/CR-6698 states that:

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A sufficient number of experiments with varying experimental parameters should be selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," as guidance for review of the code validation methodology presented in the application. NUREG/CR-66908 outline the basic elements of validation, including identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the area of applicability.

The analysis methodology uses the three-dimensional Monte Carlo code MCNP5-1.51 for the criticality analyses. MCNP5-1.51 calculations use continuous energy cross-section data exclusively based on ENDF/B-VII. All calculations are performed with a minimum of 12,000 histories per cycle, a minimum of 200 skipped cycles before averaging, and a minimum of 200 cycles that are accumulated. The calculations included a total of 532 critical experiments with fresh UO2 fuel, fresh MOX fuel, and fuel with simulated actinide composition of spent fuel (HTC experiments). The license considered which experiments would apply for the current analysis and ran a total of 365 critical experiments calculations for both borated and unborated

cases. The results for the applicable bias and bias uncertainties are documented in Tables 4.2.1(b), 4.2.1(c), and 4.2.1(d) of HI-2115004.

The licensee identified the applicable operating conditions for the validation, and compared the spectral parameters between the benchmarks and the spent fuel pool conditions to demonstrate that the selected benchmarks are applicable. The licensee performed a trend analysis and identified an additional bias for those parameters with a statistically significant trend as shown in Table 4.2.1(b).

Based on the above, the NRC staff finds that the information supporting the code validation is acceptable.

3.1.9 Summary of the Reactor Systems Review

The licensee credits soluble boron. Hence, the applicable regulatory requirement is taken from 10 CFR 50.68(b)(4), as stated in Section 2 of this safety evaluation (SE). The NRC staff evaluated the submittal against the criteria for both unborated and borated conditions. The NRC staff reviewed the analysis to ensure that the assumptions and analytical technique used are adequately substantiated to conclude at a 95 percent probability, 95 percent confidence level, that the regulatory requirement will be met.

The licensee has demonstrated through its submittal that its methodologies used in its criticality analysis follows the guidelines set forth in DSS-ISG-2010-001 and the appropriate NUREGs. After reviewing the licensee's original submittal and subsequent supplemental information, the NRC staff could determine reasonable assurance that proposed TS changes would be acceptable.

3.2 Mechanical and Civil Engineering Review

The licensee stated that PNP SFP contains 12 fuel storage racks, with space for an additional storage rack. Seven of these racks are in Region 1 and are equipped with the degraded Carborundum® neutron absorber plates. Six of the seven fuel storage racks in Region I are in the main SFP and one is in the NTP. The licensee stated that the storage capacity of the Region I fuel storage racks is 422 fuel assemblies. The storage capacity will not be altered by replacement of the Region I storage racks. The licensee further stated that the replacement SFP fuel storage racks meet all the criteria of the applicable codes and standards stipulated by the NRC guidance provided in the OT Position Paper.

The existing SFP at PNP includes SFP storage racks in two water-filled sections of the pool inside the plant auxiliary building. The first section is the main SFP, which is a reinforced concrete, rectangular pool with a stainless steel liner. This main pool contains storage racks, six of which contain degraded Carborundum® that will be replaced with new racks having the same number of cells per rack and essentially the same external dimensions. The second section of the pool is the NTP, which is also a rectangular, reinforced-concrete pool with a stainless steel liner. The two pool sections are hydraulically connected by a partial-height slot. The NTP contains three storage racks, of which one contains degraded Carborundum®. These racks will be replaced with a new rack having the same number of cells.

3.2.1 Replacement Rack Structural Evaluation

The scope of the replacement project at PNP includes the replacement of all of the existing SFP storage racks located in the Region I main SFP, and the NTP portion of the SFP, which contain degraded neutron absorber material. The present SFP Region I fuel storage racks neutron absorbing material Carborundum® is degraded and not credited as a neutron absorber. The replacement of the SFP storage racks will allow recovery of the currently unusable storage locations in the SFP. The staff's evaluation of the analyses performed to demonstrate the structural adequacy of the replacement rack structures included a review of the licensee's analysis methodology, the loading combinations used to support the structural evaluation, buckling evaluations, overturning evaluations, and additional areas outlined within the NRC's acceptance criteria relative to this review area.

3.2.1.1 Replacement Rack Construction

The replacement racks will be freestanding and self-supporting. The principal construction materials for the racks will be SA240-304L stainless steel sheet and plate stock, and SA564-630 precipitation hardened stainless steel bar for the adjustable support pedestals. To protect the liner, the replacement rack support pedestals are separated from the liner by thick bearing pads that serve to spread out any bearing impact loads from a seismic event. Whole-pool multi-rack (WPMR) seismic analyses were performed for the array of replacement racks in the main SFP. A single-rack analysis was performed for the replacement Region I rack in the NTP. The analyses were based on the simulation of the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) in accordance with SRP Section 3.7.1 requirements.

3.2.1.2 Acceptance Criteria

The NRC guidance regarding the structural acceptance criteria for which a SFP rack design must adhere to are outlined in Appendix D to SRP Section 3.8.4. In accordance with these criteria, the construction materials utilized in SFP racks should conform to the provisions for Class 3 supports found in Subsection NF of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code, Section III, which contains the requirements for the material, design, fabrication, installation, and examination of supports. The kinematic acceptance criteria specified within each of the above references provide requirements for minimum factors of safety against gross sliding and overturning of a rack structure during a seismic event. The loading combinations and their respective stress limits are specific to Level A, Level B, Level D, and a fourth service limit which includes loads due to postulated accidents; accident loading conditions are considered in Section 3.2.2 of this SE. The NRC staff notes that Footnote 3 to Table 1 in SRP Section 3.8.4, Appendix D, specifies additional requirements relative to the use of Subsection NF of the ASME B&PV Code, Section III, in this application. This footnote indicates that three provisions of RG 1.124, "Service Limits and Loading Conditions for Class I Linear Type Supports," must be considered when utilizing this subsection of the ASME B&PV Code. The staff confirmed that the design of the replacement racks at PNP did adhere to the additional stipulations discussed in RG 1.124, with respect to the use of Subsection NF.

For buckling loads, Appendix D to SRP Section 3.8.4, indicates that the SFP rack design must adhere to the provisions within Appendix XVII of the ASME B&PV Code, Section III, Division 1. For linear supports, the buckling criteria within the ASME B&PV Code have been relocated to Subsection NF (NF- 3321.1(b)). This provision of the ASME B&PV Code is identical to the

buckling requirements which were found in Appendix XVII within previous editions and addenda of the ASME B&PV Code.

The SRP and the OT position paper also specify additional considerations which must be given to the design and analysis procedures utilized to evaluate the structural adequacy of the SFP rack structures. As indicated above, the replacement racks at PNP are designed to be freestanding structures whose pedestals rest on bearing pads. The SRP notes that the seismic evaluation of free-standing SFP racks must account for a number of variables due to the complex combination of motions and impacts resulting from the deep submergence of a freestanding structure which contains spent fuel bundles. Seismic and impact loads considered in the evaluation of the SFP racks are also described within both guidance documents, with a requirement regarding the simultaneous application of seismic excitations along the three orthogonal directions during the seismic analysis. With respect to the potential for increased thermal gradients resulting from replacement, the SRP requires the evaluation of the differential heating effect between a SFP rack cell which contains spent fuel and one empty SFP rack cell.

3.2.1.3 Analysis Methodology

The NRC staff guidance associated with the analysis methods of SFP rack designs requires the consideration of a number of variables which can influence the seismic and structural evaluations of the rack design. Section 5.0 of Appendix D, to SRP Section 3.8.4, recognizes that the seismic analysis of free-standing SFP racks involves a complex combination of motions including sliding, rocking and twisting of the rack structure resulting from the motions induced within the SFP due to a seismic event. These motions are also coupled with potential impacts between the fuel assemblies and the fuel cell walls, rack-to rack impacts, and rack-to-wall impacts resulting from a seismic event. As such, the SRP notes that the seismic and structural analyses of these types of racks are typically performed using nonlinear, dynamic time-history analysis methods.

The NRC staff issued one RAI regarding the additional loads which will be generated by the impact of the fuel assemblies during a postulated seismic excitation due to the gaps between fuel assemblies and the walls of the guide tubes. In its September 6, 2012, RAI response, the licensee indicated that the time-history analyses performed for the PNP SFP rack design demonstrate that the maximum computed fuel impact loads are well below the failure limit of the fuel bundles. The licensee also indicated that Section 6.7.2 of Attachment 5 to the LAR provides information which demonstrates that the peak fuel-to-cell wall impact load is equivalent to a 2.21-g impact deceleration, which is significantly less than the 60-g design basis deceleration limit that has been approved by the NRC for Holtec's HI-STAR 100 spent fuel transportation cask. In addition to the licensee's RAI response, the staff also considered the information contained in NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," Appendix C, "Fuel Response to Loads." As outlined in Appendix C of NUREG-1864, the most limiting load for spent fuel assemblies is the vertical impact load, due to the potential for buckling of the fuel rods, and not the fuel-to-cell wall impacts considered in this case. Therefore, based on the review of the information delineated above, the staff considers the licensee's response acceptable given that it is unlikely that the consequent loads on the fuel assemblies due to a design basis earthquake at PNP will cause any fuel damage when stored in the replacement SFP racks.

The licensee employed the use of a proprietary computer program, DYNARACK, in order to perform WPMR time-history analyses of the replacement rack structures which will be utilized at

PNP. DYNARACK has been used for similar SFP replacement projects at many operating nuclear facilities. Table 6.4.1 of Attachment 5 to the LAR submittal provides a partial listing of previous re-racking projects which have utilized DYNARACK.

DYNARACK is capable of performing a time-history analysis on multiple racks within a SFP, with the additional capability of allowing for variably loaded SFP racks. The primary feature which allows DYNARACK to model these WPMR simulations is the characterization of SFP racks as a series of lumped mass modules with a number of mathematically defined features used to model the non-linear behavior of the SFP rack during a seismic event. Among other variables, this behavior is primarily a consequence of the deep submergence of the racks and the rattling caused by the fuel assemblies within the racks.

The details regarding the time-history analyses performed for the PNP replacement effort are described in Section 6.4 of Attachment 5 to the LAR submittal. This section provides a general overview of the procedure for the seismic analysis of the PNP SFP rack structures as it relates to the use of DYNARACK. In response to the staff RAI regarding whether the time-history analyses performed using DYNARACK resulted in any apparent rack-to-rack impacts at the top of the rack structures, the licensee indicated in its September 6, 2012, correspondence that the time-history analyses performed using DYNARACK did not result in any rack-to-rack impacts at the top of the rack structures. The licensee also stated that the maximum predicted displacement at the top of the replacement SFP racks is 0.3028-inch compared to the minimum clearance gap to either a wall or an adjacent SFP rack of 0.625-inch. For the new SFP rack in the NTP, the maximum predicted displacement at the top of the rack is 0.6952-inch. The staff considers this acceptable based on the fact that there is adequate clearance in the Region 1 (0.3028-inche versus 0.625-inch) and the NTP (0.6952-inch versus 5.04-inch).

The licensee provided additional specificity regarding the mathematical model of the rack structure within Section 6.4.2, "Essential of the Dynamic Model" of Attachment 5 to the LAR. It is noted that the rack itself is modeled as a 12 degree-of-freedom (DOF) structure, enabling the rack to translate in all three orthogonal directions and rotate freely about each direction at the top and bottom of the structure (i.e., there exists six DOFs at the top and bottom of the mathematical model of the rack structure). Additionally, the licensee noted that the fuel assemblies within the rack structure are modeled as five lumped masses at different elevations. Each of these lumped masses contains two translational DOFs, with the vertical DOF of each lumped mass being coupled with the vertical motion of the base of the rack. As such, each rack model contains 22 DOFs; 12 describing the motion of the rack and ten describing the motion of the fuel assemblies.

The licensee stated in Section 6.5.2, "Synthetic Time-Histories" of Attachment 5 to the LAR that the synthetic time-histories accelerations in three orthogonal directions (N-S, E-W and vertical) are generated in accordance with the provisions of Section 3.7.1 of the SRP and the design basis requirements of PNP. The licensee also stated in Section 6.6, "Dynamic Simulations for the Palisades SFP," of Attachment 5 to the LAR that two separate WPMR models were developed for the SFP and the NTP, and a total of eleven runs with fully loaded racks and random coefficients of friction were performed. The licensee further stated that simulations consider SSE excitation and are required to satisfy the stress and kinematic criteria in Section 6.2 of Attachment 5 to the LAR.

3.2.1.3.1 NRC Staff Evaluation

The NRC staff reviewed the information provided in the LAR regarding the use of DYNARACK to support the proposed re-rack of the PNP SFP. The NRC staff issued RAIs to the licensee regarding whether the DYNARACK model has been previously benchmarked against other numerical analysis methods and whether the PNP-specific acceleration time histories were used in this benchmark. In its September 6, 2012, RAI response, the licensee stated that LS-DYNA, a commercially-available dynamic finite element analysis (FEA) code, has previously been utilized to benchmark the results provided by DYNARACK simulations. This benchmark work is described in a previous RAI response related to the SFP re-rack LAR associated with the Beaver Valley Unit 2, nuclear plant (see letter dated May 21, 2010, ADAMS Accession No. ML101460057). The NRC staff previously reviewed the RAI response regarding the Beaver Valley, Unit 2, benchmarking and found the benchmarking acceptable in its SE dated April 29, 2011. This acceptance was based on the fact that the time-history results from the DYNARACK simulation demonstrated that the dynamic response using the lumped-mass model in the DYNARACK code are more conservative than those results generated using detailed FEA. The NRC staff considers the licensee's extension of the Beaver Valley, Unit 2, benchmarking to PNP acceptable given that it is reasonable to assume that a benchmarking of the PNP DYNARACK model against LS-DYNA will again yield conservative results. The NRC staff considers this reasonable due to the fact that a change in the parameters (from Beaver Valley Unit 2 to PNP) using each methodology is still likely to yield similar results, that being a conservative assessment using the DYNARACK model.

3.2.1.4 Structural Analysis Results

The results of the analyses performed to demonstrate compliance with the aforementioned acceptance criteria are documented in Section 6.7 of Attachment 5 to the LAR, the limiting impact forces, displacements, stress factors and pedestal loads relative to the structural evaluation of the replacement rack models were extracted from the various DYNARACK runs to enable a complete evaluation of the rack's structural integrity for service levels which include seismic loads. The stress factors used to evaluate the structural adequacy of the rack against the ASME B&PV Code criteria are described in Section 6.2 of Attachment 5 to the LAR. These factors represent a ratio between the actual stresses developed in a rack structural member, based on the DYNARACK time history analyses, to the allowable stresses for the same member for the prescribed service levels (i.e., Levels A, B, and D). By maintaining these stress factors to a value of less than 1.0, compliance with the applicable ASME B&PV Code provisions is confirmed. As indicated in Section 6.7 of Attachment 5 to the LAR, DYNARACK provides numerical results for the rack pedestal normal and lateral forces resulting from the time history analyses. In turn, these forces are utilized to determine the loading conditions on the most limiting locations (i.e., near the base of the rack) and subsequently, to develop the corresponding stress factors. The highest stress factor reported from the DYNARACK time history analyses was 0.140. Based on this information, the licensee concluded that the structural acceptance criteria were satisfied given that all of the stress factors remained below a value of 1.0.

The calculated stresses from the SSE simulations are compared against the allowable stresses corresponding to ASME Service Level D. No time-history simulations are performed for OBE loading. In lieu of performing OBE simulations, the minimum calculated safety factors for the SSE simulations must exceed 1.8 for weld and base metal shear and 2.0 for tension,

compression, bending, and combinations thereof. This approach is justified, and insures that OBE load combinations are not controlling, because:

1. OBE is equal to ½ of SSE, per Section 5.7.1.2 of the PNP FSAR

2. The allowable stresses for ASME Level D are equal to two times the allowable stresses for ASME Level A (excluding shear stress);
3. The allowable shear stress for ASME Level D is equal to 1.8 times the allowable shear stress for ASME Level A

4. The same damping value applies to OBE and SSE per Table 5.7-2 of the PNP FSAR

The multi-rack model prepared in the manner of the preceding sections is subjected to each of the five sets of the three dimensional time history accelerations. The maximum (in time and space) response of the rack modules under each of the five time-history sets is summarized in tables below.

Rack N7 in the NTP is modeled as a freestanding autonomous module with the interactive effect of fluid coupling from the surrounding structures appropriate for deeply submerged components. The response of the rack under each of the five time-history sets is also summarized in tables below:

Table 1

Maximum Values of Stress Factors and Impact Loads

Pool	Run No.	Max. Stress Factor (defined in Sec 6.2)	Max. Vertical Load on Single Pedestal (lbf)	Max. Shear Load on Single pedestal (lbf)	Max. Fuel to Cell Wall Impact (lbf)
Spent Fuel Pool	1	0.121	83,300	24,500	675
	2	0.119	90,600	25,300	698
	3	0.117	86,200	24,200	669
	4	0.122	76,200	28,000	760
	5	0.140	86,000	31,800	720
	11	0.118	84,100	22,800	709
Tilt Pit	6	0.095	72,600	25,300	592
	7	0.099	70,600	25,600	552
	8	0.092	70,100	25,800	632
	9	0.092	69,600	27,100	522
	10	0.092	81,700	22,800	700

Table 2

Maximum Values of lateral Displacements

Location on rack		**Maximum rack displacement relative to floor (inch)**	**Run No.**
Spent Fuel Pool	Base Plate	0.0164	5
	Top of Rack	0.3028	5
North Tilt Pit	Base Plate	0.0404	10
	Top of Rack	0.6952	10

3.2.1.4.1 Rack Cell Temperature Gradient

Appendix D of SRP Section 3.8.4 and the OT Position Paper require the evaluation of the rack structure for a specific, limiting thermal condition, whereby the structure is subjected to a bounding temperature gradient resulting from an empty cell being located adjacent to a cell loaded with spent fuel (i.e., an isolated hot cell). This differential heating effect results in thermal stresses being induced in the cell-to-cell welded joints which connect the adjacent cells. The results of the licensee's evaluation of the thermal gradients on cell-to-cell welds, resulting from an isolated hot cell, are presented in Section 6.8.2, "Analysis of thermal effects," in Attachment 5 to the LAR. The licensee stated that a temperature gradient of 50 °F was utilized in the evaluation and it is a conservative value based on the fact that the thermal-hydraulic analysis presented in Section 6.6.2 of Attachment 5 to the LAR revealed that the limiting cell temperature differs from the bulk pool temperature by a maximum value of 44 °F. Based on this thermal gradient, the evaluation results demonstrated that the maximum shear stress developed in the weld was within the stress limit prescribed by the applicable ASME B&PV Code provision.

The licensee also indicated that the maximum shear stress in an isolated hot cell, due to thermal gradient, is 14,030 psi. However, the thermal stresses do not require evaluation under Subsection NF per NF- 3121.11. But, for conservatism, the calculated stress is compared against the primary plus secondary stress limit per Table NF-3523(b)-I of the ASME Code. For SA-240-304L material, the limit is controlled by 2Sy, which equals 42,800 psi based on the material yield strength given in Table 6.5.1 of Attachment 5 to the LAR. Therefore, there is a safety factor of 3.05 against cell wall shear failure due to secondary thermal stresses from cell wall growth under the worst case hot cell condition. This provides a safety factor greater than two when compared against the acceptance limit from the OT position paper for load combinations involving thermal loads, which demonstrates compliance with the applicable design code provisions.

3.2.1.4.2 Rack Displacements

With respect to the evaluation of postulated overturning of a rack structure, the licensee reported that the maximum displacements are limited to 0.6952 inch at the top and 0.0404 inch at the base plate elevation of the rack array. The above bounding values of the computed rack displacement indicate that the modules' movements are insignificant and that there will be no rack-to-rack impacts in the cellular region. Likewise, no impact between the peripheral racks and proximate pool structures is indicated. The maximum lateral movement of the pedestal, limited to less than 0.1 inch, as indicated by Table 2 of this SE, provides assurance that the rack pedestals will not slide off their bearing pads. Therefore, overturning was of no concern and the acceptance criteria specified in Section 3.8.5 of the SRP were satisfied.

3.2.1.4.3 Cell Wall Buckling Evaluation

The licensee determined the allowable local buckling stresses in the fuel cell walls (from vertical loading) based on classical plate buckling analysis. Using this analysis, the licensee determined that the critical buckling stress on the lower portion of the cell walls is equal to 22,204 psi, which is greater than the yield strength (21,400 psi) for the rack cell wall material. The ASME B&PV Code buckling provisions require the maximum compressive stress developed in the member to be no more than two-thirds the critical buckling stress, which yields an allowable compressive stress of 14,788 psi for the cell wall. The licensee stated that the maximum compressive stress in the outer most cell is 3,578 psi. Therefore, the buckling criteria are satisfied for this rack design.

3.2.1.4.4 Fatigue Evaluation

Subsection NF of the ASME B&PV Code does not require a fatigue evaluation for Class 3 linear-type supports. However, for conservatism, the licensee performed fatigue evaluations for the following high stress concentration areas: (i) the threaded connection between the inner and outer support pedestals, and (ii) the weld connection between the rack cell walls and the baseplate. Based on the maximum stress at the weld location of 10,896 psi and the maximum compressive stress in the pedestal of 5,659 psi (based on maximum pedestal load 110,000 lbf), the resulting amplified alternating stress intensity is 19,066 psi. Table I.9-1 of the ASME Code shows that the endurance limit (corresponding to 1E+06 cycles) is 28.3 ksi. As such, the licensee concluded that there is a low likelihood of fatigue failure from oscillating stresses in the rack structure when the structure is subjected to loads from multiple earthquakes.

3.2.1.4.5 Weld Stress Evaluation

Section 6.7.7 of Attachment 5 to the LAR details the structural evaluations performed for the critical welds on the BVPS-2 replacement rack structures (i.e., welds susceptible to failure due to seismic loading). As indicated in Section 6.7.7 of Attachment 5 to the LAR, the limiting baseplate-to-rack cell welds, baseplate-to-pedestal welds, and cell-to-cell welds were evaluated against the aforementioned ASME B&PV Code provisions for the applicable service limits:

a. Baseplate-to-Rack Cell Welds

The rack's cellular structure is connected to the base plate through fillet welds that are typically 6 inches long. The maximum values of the tensile stresses in the connecting welds and the adjacent base metal is computed using the maximum values of the stress factors from the DYNARACK simulations. The table below shows the stresses in the weld and base metal along with the associated factors of safety (allowable is 1.0 minimum).

	Stress (psi)	**Allowable Stress (psi)**	**Safety factor**
Weld	10,896	35,694	3.28
Base Metal	6,552	15,408	2.35

b. Baseplate-to-Pedestal Welds

The shear load on any pedestal for all runs is less than 35,000 lbf per Table 6.6.1 of Attachment 5 to the LAR. This bounding shear load is used as an input to evaluate the

integrity of the pedestal-to-baseplate weld. Conservatively, this shear force is applied in two directions (x and y) simultaneously. The maximum pedestal compressive force is determined to be bounded by 110,000 lbf. This force is also applied to the finite element model. The table below shows that the stresses in weld and base metal are acceptable and that the safety factors are greater than 1.0.

	Stress (psi)	Allowable Stress (psi)	Safety factor
Weld	9,497	35,694	3.76
Base Metal	6,716	15,408	2.29

c. Cell-to-cell welds

Cell-to-cell joints consist of a series of connecting welds along the cell height. Stresses in storage cell-to-cell welds develop due to fuel assembly impacts with the cell wall. These weld stresses are calculated by assuming that fuel assemblies in adjacent cells are moving out of phase with one another so that impact loads in two adjacent cells are in opposite directions thus maximizing the stress in the connecting longitudinal welds. Both the weld and the base metal shear stress results using the strength-of-materials solution process are summarized below. The table below shows the safety force in cell-to-cell welds along with the associated factors of safety (allowable is 1.0 minimum).

Analysis Type	Stress (psi)	Allowable Stress (psi)	Safety factor
Weld	2,813	35,694	12.69
Base metal shear	1,989	15,408	7.75

The results demonstrate that each of the limiting weld locations have margins which are sufficient and acceptable, when compared to the ASME B&PV Code stress limits.

3.2.1.4.6 NRC Staff Evaluation

The NRC staff has reviewed the results of the licensee's structural evaluation of the replacement rack structures and finds the evaluation acceptable. This acceptance is based on a number of factors, which are outlined below. Based on the licensee's analysis of the rack structure, which subjected the structure to bounding loading combinations under the required service limits, the results of the structural analysis of the rack were shown to be in accordance with the pertinent provisions of Subsection NF of the ASME B&PV Code, Section III. As such, the licensee demonstrated that the rack meets the applicable acceptance criteria related to the stress limits and load combinations specified in Appendix D of SRP Section 3.8.4

With respect to a limiting impact between a fuel assembly and a rack cell due to seismically induced rattling, the licensee adequately demonstrated that the impact would not cause plastic deformation of the cell wall. The elastic behavior of the cell wall during this impact is acceptable given that geometry of the cell wall is maintained. The staff finds the licensee's buckling analyses performed to demonstrate acceptable margin against buckling at the limiting portion within the rack (i.e., the bottom of the structure), acceptable. This acceptability is based on the licensee's quantitative demonstration that the ASME B&PV Code requirements related to buckling are satisfied. The staff considers the overturning analysis results of the replacement SFP racks during a seismic event acceptable because the licensee adequately demonstrated that the acceptance criteria stipulated in Section 3.8.5 of SRP were satisfied. The staff also notes that while the fatigue evaluations performed by the licensee were not required by the code

of record used to qualify the replacement racks, the evaluations performed by the licensee demonstrated that the cyclic stresses produced by seismic loads will not exceed the endurance limit for the material. Therefore, the NRC staff concludes that there is reasonable assurance that fatigue failure of the highly stressed regions of the replacement rack structures is unlikely. Based on the review outlined above, the staff has determined that reasonable assurance has been provided which demonstrates that the replacement rack structures at PNP will maintain adequate structural margin under normal and abnormal loading conditions.

3.2.2 Mechanical Accidents

In accordance with Section IV(1)(b) of the OT Position Paper and Appendix D to SRP Section 3.8.4, SFP racks must be designed to withstand the effects of postulated fuel handling accidents. Specifically, SFP rack designs must demonstrate functional integrity following the occurrence of these postulated fuel handling accidents. The postulated fuel handling accidents evaluated in support of the proposed PNP replacement rack include a straight drop on the top of a rack (i.e., the shallow drop event), a straight drop through an individual cell all the way to the bottom of the rack (i.e., the deep drop event) and an inclined drop of a fuel bundle on the top of a rack. The licensee also evaluated the effects on the functional integrity of the replacement rack due to the postulated gate drop event, and the effects on the rack due to the uplift force resulting from a stuck fuel assembly.

The NRC staff notes that Section 7.2 of Attachment 5 to the LAR submittal, states that a rack drop event (i.e., drop of heaviest fabricated spent fuel rack to the bottom of the SFP) is postulated if the rack handling equipment is not designed, analyzed, fabricated, tested, and certified to the criteria outlined in NUREG-0612 and NUREG-0554 for single-failure-proof cranes. To satisfy the single-failure criteria stipulated by NUREG-0612 and NUREG-0554, Section 9.11.4.3 of the PNP FSAR states that PNP Facility Change FC-976 (2003) modified the main hoist of the Fuel Building Crane to increase the crane capacity to 110 tons. Based on the fact that a heavy load drop, such as the drop of an empty spent fuel rack, is not a credible accident scenario when the load is transferred with a single failure-proof crane, a rack drop analysis was not required to be performed as part of this LAR.

3.2.2.1 Acceptance Criteria

Appendix D of SRP Section 3.8.4 specifies that SFP rack designs must demonstrate functional integrity following postulated accidental drops of the heaviest loads from maximum possible heights. With respect to postulated accidents in which the limiting components may be those associated with the SFP structure, and not the racks themselves, the OT Position Paper and Appendix D of SRP Section 3.8.4 requires the functional capability of the SFP to be maintained following such an accident. As such, any postulated accident which can cause gross structural damage to the SFP liner and concrete must not result in a loss of SFP water inventory. Additionally, both guidance documents specify that ductility ratios used to absorb the kinetic energy associated with the postulated accidents must be quantified. The ductility ratio requirements applicable to reinforced concrete missile barriers are outlined in Section 5.5 of the PNP UFSAR.

3.2.2.2 Shallow-Drop Event

Chapter 7 of Attachment 5 to the LAR provides the details of the mechanical accident analyses performed in support of the proposed LAR, including the shallow-drop event discussed here

within. The licensee stated that the model used to demonstrate the structural integrity of the replacement rack structure under postulated event conditions, the pertinent parameters used in the event simulations (Table 7.4.3 and Figure 7.4.1) shows the plastic strain and the extent of deformation in the crush zone resulting from the shadow drop event. It is observed that the plastic deformation of the cell wall is above the neutron absorber region. Therefore, this deformation does not present any criticality concern or unacceptable consequences.

3.2.2.2.1 Shallow-Drop Event Analysis Methodology

The analyses performed for the shallow-drop event scenario at PNP are based on computer simulations using the LS-DYNA FEA computer code. Previous SFP replacement rack LARs have made extensive use of the LS-DYNA computer code for mechanical event analyses and the NRC has documented its review and approval of these analyses in previous SEs. As previously indicated, the pertinent simulation data used in the shallow drop event analysis is included in Table 7.4.2 of Attachment 5 to the LAR. These critical simulation parameters include the impact weight (i.e., the combined weight of one PNP fuel bundle and fuel handling tool) and the impact velocity, which was computed based on the assumption that the impactor would be free falling through the water. The analysis of the shallow drop scenario assumes that the impactor free falls through the SFP water until the impactor strikes the top of the replacement rack structure on a peripheral rack wall.

3.2.2.2.2 Staff Evaluation

The NRC staff finds the analysis acceptable based on the fact that the assessment assessments performed to demonstrate the functional integrity of the PNP replacement racks subject to shallow drop impact loads, the staff's review concludes that the licensee's assessment is acceptable, based on the results of the analytical evaluations, which yielded plastic deformation (≥0.01 in/in) of the cell wall is confined in the crush zone in the rack cell walls that does not extend down into the neutron absorber zone, with an adequate margin of safety accidents are analyzed and found to produce localized structural damage in the impacted region well within the design limits for the racks. Therefore, it is determined that the damage from the postulated shallow drop event will not result in criticality concern or unacceptable consequences.

3.2.2.3 Deep-Drop Event

The description of the mechanical events considered in the evaluation of the functional integrity of the replacement SFP racks proposed for use at PNP is included in Section 7.2 of Attachment 5 to the LAR. The licensee notes that two types of deep-drop events were evaluated; (1) a drop into an interior rack cell away from the support pedestals, and (2) a drop into a rack cell located above a support pedestal. The most limiting scenario between these two deep-drop event variations is the former case, in which the fuel bundle impacts the base plate of the replacement rack structure away from the pedestal, which is most flexible due to the lack of pedestal support underneath the center of the rack. This is illustrated by Figure 7.2.2 and Figure 7.2.3 of Attachment 5 to the LAR. The licensee stated the acceptance criteria for this postulated accident is such that rack module baseplate must be sufficiently robust to prevent piercing of the baseplate by the dropped fuel assembly and deformations large enough to cause secondary impact with the SFP liner. Additionally, large baseplate deformations must be

avoided to prevent fuel assemblies in the impacted region from protruding excessively towards the liner, leading to a potential increase in the reactivity beyond that permitted for an accident event.

In performing the analyses, the licensee employed LS-DYNA to evaluate the deep-drop event scenarios. By determining the initial conditions resulting from a 24-inch fuel bundle drop height, the licensee simulated the structural effects resulting from the fuel bundle traversing the entire length of the cell, in contrast to the shallow drop event whereby the fuel bundle impacted the rack at the top edges. The results of the most limiting deep drop event scenario, with respect to the functional integrity of the rack, are summarized in Section 7.4.2 of Attachment 5 to the LAR and presented graphically in Figures 7.4.2, 7.4.3, and 7.4.4 of the same enclosure. These results demonstrate that while local plastic deformation of the base plate does occur due to weld separation, the SFP liner remains unaffected. The licensee notes that the base plate deformation due to the deep drop event postulation is considered in the criticality analyses.

3.2.2.3.1 NRC Staff Evaluation

The NRC staff finds the deep-drop event assessment performed by the licensee in support of the proposed replacement rack of the PNP SFP acceptable, primarily based on the fact that the licensee demonstrated that only local deformation, which included weld separations, occurs due to the deep-drop event. This deformation remained constant when the licensee incorporated the use of the true stress-strain material model in the revised deep-drop event analysis. The licensee was able to demonstrate that the deep-drop event does not affect the leak tightness of the SFP liner, given that the deformation of the rack base plate is limited to a depth which is above the SFP liner.

3.2.2.4 Inclined-Drop Event

The OT Position Paper requires the consideration of a fuel handling accident whereby an inclined fuel bundle is postulated to impact a SFP rack. Following this impact, the functional integrity of the SFP rack must be demonstrated. The LAR submitted in support of the proposed replacement racks of the PNP SFP did not include information regarding the postulated impact of an inclined fuel assembly on the replacement racks at PNP. In its November 7, 2012, response to an NRC staff RAI regarding the absence of this information, the licensee stated that this event was not explicitly analyzed because it is bounded by a straight (vertical) fuel assembly drop on the top of a rack, with respect to the potential damage to the rack structure. The licensee also stated that, based on the rack's geometry and physical construction, the maximum depth of damage occurs when a fuel assembly falls in a vertical orientation and impacts the top edge of a rack cell wall. This is because all of the dropped fuel assembly's kinetic energy is focused on a single storage cell location. In the event of an inclined fuel assembly drop, only a portion of the fuel assembly's kinetic energy is delivered to the initial impact point. Once the inclined fuel assembly comes in contact with the top of the rack, the fuel assembly will rotate downward to a horizontal position, at which point a secondary impact will occur across many storage cell locations. The larger impact area results in less depth of damage to the rack. Thus, a straight (vertical) drop of a fuel assembly bounds an inclined fuel assembly drop from a damage standpoint.

The NRC staff finds the licensee's justification for not performing an explicit evaluation of an inclined fuel assembly impact acceptable because only a portion of the fuel assembly's kinetic energy is delivered to the initial impact point in the event of an inclined fuel assembly drop. In

the event of a straight, vertical drop of a fuel bundle, all of the kinetic energy is absorbed by the initial impact point. As such, the NRC staff notes that the inclined drop is bounded by the straight (vertical) fuel assembly drop in terms of depth of damage to the rack. The NRC staff considers this acceptable. It should be noted that the staff's acceptance of the above reasoning is specific to PNP and, therefore, is not a generic acceptance and the necessity of performing an inclined-drop event analysis. Given that consideration of this accident is specified in the OT position paper, reviews of other inclined drop accidents will continue to be reviewed on a case-by-case basis.

3.2.2.5 Gate-Drop Event

In addition to evaluating the effects of a postulated shallow drop of a fuel bundle on the PNP replacement racks, the licensee also evaluated the effects of an additional shallow-drop event. The second shallow-drop event, described in Section 7.2 of Attachment 5 to the LAR, postulates a drop of the transfer canal gate onto the top of the replacement rack structure. This gate is used to cordon off the fuel transfer canal between the reactor containment building and the SFP in the fuel handling building. As with the other events considered in support of the evaluation of the PNP replacement racks, the pertinent evaluation are presented in Section 7.4 of Attachment 5 to the LAR and evaluation parameters for this accident are located in Table 7.4.1 of Attachment 5 to the LAR. The staff finds the licensee's assessment of the gate-drop event acceptable based on the dimensions of the gate; the limiting gate drop event resulting in an impact to only a periphery cell wall would lead to immediate gate rotation because the gate width is more than four times the size of the rack cell. However, the impact energy of the gate is only about 1.17 times that of the fuel assembly in the shallow-drop event. Therefore, it can be concluded that the gate-drop event is bounded by the shallow-drop event in terms of the plastic strains sustained in the crush zone.

3.2.2.6 Stuck-Fuel (Uplift Force) Event

Section 7.2 of Attachment 5 to the LAR includes a description of the event analysis performed to demonstrate the satisfactory structural response of the PNP replacement SFP racks due to a stuck fuel assembly. This accident assumes that a bounding load is applied to the racks resulting from the inability to remove a fuel assembly from one cell of the replacement racks. The results of the uplift force evaluation presented in Section 7.4 of Attachment 5 to the LAR indicate that the rack is able to withstand the vertical uplift force of 5,000 lbf. The maximum stress in the rack cell as a result of trying to pull the stuck fuel assembly out is only 1,500 psi, which is well below the material yield strength (28 ksi for the SA240-304L).

The NRC staff finds the licensee's assessment of the replacement rack's structural integrity during a postulated stuck fuel assembly event acceptable, based on the following factors. The licensee utilized a bounding value for the uplift force which could be expected due to a stuck fuel assembly. The staff finds this as an acceptable and bounding value, based on the fact that the evaluation performed by the licensee using a value of 5,000 lbs uplift force bounds the conditions which could be expected at PNP. The PNP SFP handling machine hoist and grapple have a load capacity of 3,200 lbs. The hoist and grapple raise and lower loads within the stationary mast on the SFP handling machine. Additionally, an auxiliary hoist, with a load capacity of 2,000 lbs, is used for fuel handling which is less than 5,000 lbs. The maximum stress induced in the rack cell as a result of trying to pull the stuck fuel assembly out with an uplift force of 5,000 lbf is well below the material yield strength (28 ksi for the SA240-304L). Using this value, the licensee demonstrated that the replacement racks will not be expected to

plastically deform, given that the maximum stress induced in the replacement rack is well below the limit at which departure from elastic behavior would be expected. The staff finds this acceptable based on the fact that there is significant margin between the stress induced in the replacement rack and the stress limit at which the functional capability of the replacement rack would be called into question; this satisfies the mechanical accident acceptance criterion previously outlined.

3.2.3 Spent Fuel Pool Structural Evaluation

The SFP is a cast-in-place; steel lined, and reinforced concrete structure which provides space for the storage of fuel assemblies. To ensure that an adequate inventory of the pool water exists in the PNP SFP at all times, it is essential that the engineered barriers against uncontrolled loss of water from the pool remain effective under all operational conditions. The barriers installed in the PNP SFP to ensure water retention consist of the reinforced concrete structure and the pool liner. The former is designated as a safety significant load bearing structure that must meet the load combinations of the applicable ACI code prescribed in the plant's FSAR. The latter is the pool liner made of seam welded sections of austenitic stainless steel that provides a non-structural barrier that is nevertheless important in keeping the corrosive pool water from affecting the reinforced concrete. To ensure that both water retention systems in the PNP SFP will maintain their physical integrity under the loadings that act on them under all service conditions to which they are subjected during their design life. The pertinent dimensions of the SFP, including the NTP, are documented in Table 8.2.1 of Attachment 5 to the LAR submittal. As the licensee indicated in its LAR submittal, the replacement of the spent fuel racks in Region I of the SFP and the NTP will not increase the total number of available spent fuel storage cells at PNP. In Section 8.2 of Attachment 5 to the LAR submittal, the licensee documents its assessments of the various loads imposed on the SFP slabs and SFP walls. For the slabs, the licensee notes that the applicable loads include the deadweight of the water, spent fuel racks and fuel and seismic inertia loads. While the deadweight of the fuel and water will not change as a result of the re-rack, the licensee notes that the seismic inertia and rack weight loads will be reduced with the installation of the new racks.

For the SFP walls, the licensee indicates that applicable loads on these structures include the lateral pressure acting on the walls from the pool water, hydraulic pressure resulting from sloshing during a seismic event and mechanical loads induced by the seismic restraints currently utilized for spent fuel storage at PNP. The mechanical restraints will be removed concurrent with the installation of the new racks at PNP, which eliminates the mechanical restraint loads. Additionally, the pressure and sloshing loads will remain unchanged due to the fact that the SFP water inventory will remain unchanged as a result of the re-rack.

As summarized in Section 8.3 of Attachment 5 to the LAR submittal, the licensee also performed an evaluation of the effects of the proposed re-rack on the SFP liner. The results are summarized in the tables below:

Table 3

Key Input Data for SFP Liner Evaluation

Item	Value
Liner thickness, in	0.1875
Liner plate dimensions, in	48 x 96
Width of bearing pad, in	12
Hydrostatic pressure on SFP liner, psi	17.3 (note 1)
Bounding normal operating temperature, °F	150
Maximum shear load on SFP liner from WPMR analysis, lbf	<35,000

Note 1: Conservatively based on 40 feet of water.

Table 4

Results of SFP Liner Evaluation

Item	Value
Maximum computed shear stress in the liner, psi	7,991
Allowable shear stress for liner material at 150°F, psi	12,076
Factor of safety against tearing	1.51
Maximum computed alternating stress intensity, psi	8,000
Number of stress cycles associated with 1 SSE and 20 OBE events	7,140
Maximum allowable alternating stress intensity at 1 million stress cycles, psi	28,200
Cumulative damage factor	7,140 / 1,000,000 = 0.00714

The NRC staff finds the licensee's evaluation of the SFP structure, including the SFP and NTP slabs, walls, and stainless steel liner, acceptable. This acceptance is based primarily on the fact that the licensee was able to demonstrate that the loads which affect the structural integrity of these SFP structures will not be affected, or will be reduced, by the implementation of the proposed re-rack of the PNP SFP and NTP. The NRC staff considers this a reasonable assertion, given that the licensee does not intend to increase the capacity of the SFP but rather only intends to replace the rack structures themselves in a one-for-one approach. While the old and new racks are different, this difference serves to reduce the overall load imposed by the spent fuel racks on the SFP structures. The licensee was able to demonstrate this through its WPMR simulations to develop revised seismic loads using the new racks. Additionally, the licensee also notes that the mechanical restraint loads currently imposed on the SFP walls will be eliminated with the installation of the new racks, given that the new racks are free-standing and do not require mechanical restraints. Based on the fact that the acceptance criteria related to the structural integrity of the SFP structures (i.e., design code requirements) will continue to be satisfied following installation of the new racks, the NRC staff concludes that the licensee has provided reasonable assurance that the structural integrity of the SFP will be maintained following installation of the replacement racks at PNP.

3.2.4 Summary of Mechanical and Civil Engineering Review

Based on its review as described above, the NRC staff concludes that the proposed LAR regarding the replacement fuel storage racks in Region I of the main SFP and the north tilt pit portion of the SFP is acceptable. This acceptance is outlined above and is based on the licensee's compliance with the acceptance criteria related to the structural adequacy of the replacement racks themselves and other SSCs affected by the proposed LAR, including the SFP and the SFP liner under normal and abnormal loading conditions. Furthermore, based on its review described above, the NRC staff has concluded that the regulatory requirements described in SE Section 2.0 have been satisfied for the existing and replacement SSCs at PNP associated with the proposed replacement. Therefore, there is reasonable assurance that the structural integrity of the affected SSCs will be maintained following the implementation of the proposed LAR.

3.3 Balance of Plant NRC Staff Review

As stated in Section 5.1.7.2 of the PNP FSAR, the SFPCS is the normal means of decay heat removal from the SFP. Section 9.4 further states that the SFPCS maintains the water temperature in the SFP below 150 °F for the partial or full core offload scenario. This is achieved by limiting the maximum heat load present in the SFP. No changes to the SFPCS or the licensed SFP spent fuel storage capacity were included in the proposed amendment. Following the replacement of the Region I spent fuel storage racks the SFPCS must continue to perform as described in the FSAR.

To demonstrate the continued capability to provide adequate decay heat removal, the licensee provided a detailed thermal hydraulic evaluation in Attachment 5 to the LAR "Holtec Report HI-2115004." In Chapter 5, the Holtec Report outlines the methodology, assumptions, and results of the thermal hydraulic analyses. The NRC staff reviewed the report to ensure the methods used and information required met the guidance of the NRC OT Position Paper. The NRC staff also evaluated the methodology described with respect to the precedents identified in the amendment request.

The NRC OT Position Paper guidance calls for use of conservative methods when calculating the SFP water temperature, heat up rate, and the local water temperature. The Holtec Report describes how the assumptions used in the calculation of bulk water temperature and time-to-boil conservatively maximize the water temperature.

The SFPCS was the only credited path for heat removal from the pool, ignoring the effects of heat lost from the surface of the SFP or through the SFP structure. Loss of SFP cooling was timed to begin coincident with the highest SFP water temperature. The calculation of maximum local water temperature imposes additional hydraulic resistance to conservatively maximize the local heating. The NRC staff found that the methodology described in the Holtec Report is conservative in determining the impact of the new spent fuel storage racks on the thermal hydraulic performance of the SFP. The NRC staff found that the methodology reflects assumptions and calculations similar to the identified precedent amendment for Beaver Valley Power Station, Unit 2, which was approved by the NRC.

The Holtec Report concluded that the maximum bulk water temperature in the case of a full core offload would be 150 °F. This result takes into account a 158-hour minimum in-core hold time before spent fuel assemblies are transferred to the SFP. The minimum time-to-boil, assuming

loss of SFP cooling coincident with the highest pool temperature was determined to be 1.8 hours. In the event of a loss of the SFPCS this would be the minimum available time to initiate corrective actions. With consideration for the available sources of makeup to the SFP, the NRC staff concluded that this represents sufficient time to respond before the SFP temperature reached boiling. The maximum local water temperature was determined to be 168 °F. This is lower than the saturation temperature at the top of the spent fuel storage racks. Therefore, the NRC staff concluded that boiling within the racks will not occur using the new spent fuel storage racks.

Chapter 10 of the Holtec Report describes the installation of the new spent fuel storage racks and the removal of the current spent fuel storage racks. The Region I spent fuel storage racks will be removed from the pool and replaced, one at a time, using the fuel building crane, which is described in FSAR Section 9.11.4.3. The fuel building crane is approved for use as a single-failure-proof crane for loads up to 110 tons. All heavy load handling operations will be performed in compliance with the guidance of NUREG-0612.

NUREG-0612, Section 5.1.1, provides general guidance for the safe handling of heavy loads in nuclear power plants, and Section 5.1.2 provides additional guidance for heavy load handling in the vicinity of the SFP. The Holtec Report describes the procedures that will be used for heavy load handling operations and the required training for crane operators. Safe load paths have been pre-established and will be followed for the movement of spent fuel storage racks. The fuel building crane is single-failure-proof and all lifting devices are designed to the standards identified in NUREG-0612. The NRC staff reviewed the information provided in the proposed amendment and the Holtec Report and found that the planned heavy load handling operations complied with the guidance of NUREG-0612.

3.3.1 Summary of Balance of Plant Review

The proposed amendment supports the replacement of the SFP Region I storage racks. The NRC staff reviewed the thermal hydraulic analysis of the SFPCS and the heavy load handling described in the Holtec Report and amendment request. Installation of the new Region I spent fuel storage racks did not result in a significant change to the performance of the SFP cooling system. Therefore, the NRC staff concluded that the SFPCS will continue to perform as described in the PNP FSAR and that the system provides adequate decay heat removal for the quantity of spent fuel that will be stored in the SFP. The storage racks will be lifted using the single-failure-proof fuel building crane and all heavy load handing operations will be performed in conformance with NUREG-0612. Therefore, the NRC staff concluded that performance of heavy load lifts associated with the replacement of SFP Region I storage racks will be performed with adequate safety.

3.4 Health Physics and Human Performance Review

3.4.1 Health Physics Review

3.4.1.1 Occupational Radiation Exposure

The NRC staff has reviewed the licensee's plan for the replacement of the existing Region I storage racks at PNP with respect to occupational radiation exposure. The license estimates that the proposed SPF storage rack installation can be performed for 3.85 person-rem. The estimate includes the fuel handling, fuel rack removal and installation, and radioactive waste

processing of the existing contaminated racks. There are no plans to use divers for this fuel pool rack replacement operation.

The fuel rack removal and installation will use procedures prepared by the licensee's contractor (Holtec) with full consideration of as low as is reasonably achievable (ALARA) principles. In accordance with the licensee's radiation protection procedures, workers performing the replacement activities will be given daily pre-job briefings to acquaint each team member with the scope of work to be performed and the proper means of performing those tasks. Remote tooling such as lift fixtures, underwater welding, and a support leveling devise have been developed to execute numerous activities from the SFP surface where dose rates are relatively low. During the course of fuel storage rack project, primary shielding will be provided by the water in the SFP. The licensee states that if necessary, additional shielding may be used to meet ALARA principles.

The PNP SFP area radiation and airborne monitoring instrumentation will be augmented with wireless remote monitors as needed during the re-rack operations. In addition, remote monitors may be positioned to determine the conditions of high dose rate components, fuel pool cleanup system filters, or underwater debris baskets.

As materials are being removed from the SFP, particular attention will be paid to controlling the removal of any irradiated particles from the SFP (e.g., pressure washing and Health Physics surveys of racks being removed from the SFP). All the rack removal and installation activities in the SFP floor area will take place within a defined foreign material exclusion zone.

During all operations except actual rack handling, existing plant equipment (e.g., SEP cooling filtration and skimmers, and supplemental "tri-nuke" pool vacuum equipment) will be utilized to remove materials dispersed to the SFP water. At all times, SFP water activity is monitored to track and control the results of pressure washing and rack handling operations. During rack handling, SFP cooling must be off to reduce water movement, facilitate safe rack manipulation, and minimize the impact on pool clarity from the potential release of materials from existing degraded fuel racks. In cases of degraded SFP clarity from handling operations, the plant's SFP clean up systems will be operated to restore SFP water quality prior to subsequent rack movements. In addition, after each removal sequence (for example, removal of 3 racks) the SFP floor will be vacuumed to remove radioactive sediments and other bulk materials.

On the basis of our review of the PNP license amendment, the staff concluded that the proposed changes to the SPF can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The NRC staff finds that the projected total dose for the project of 3.85 person-rem is in the range of doses for similar modifications at other plants and it is therefore acceptable.

3.4.1.2 Radioactive Waste

The existing contaminated fuel storage racks will be the main source of radioactive waste for the proposed modification. They will be washed to remove as much contamination as possible and surveyed by Health Physics. The removed racks shall be moved to a designated area for packaging and preparation for shipment. The contaminated racks and the SFP clean up system filters will be processed and shipped offsite in accordance with the existing licensee radioactive waste procedures.

3.4.1.3 Summary of Health Physics Review

On the basis of our review of the PNP license amendment, the NRC staff concludes that the proposed increase in spent fuel storage capacity at PNP can be performed in a manner that will ensure that doses to the workers will be maintained ALARA and the generation of additional solid radioactive wastes minimized. Therefore, the staff finds the proposed increase in spent fuel storage capacity at PNP to be acceptable.

3.4.2 Human Performance Review

3.4.2.1 Description of Operator Action(s) and Assessed Safety Significance

There are no changes to Operator Actions as a result of this LAR.

3.4.2.2 Operating Experience Review

Operating experience at PNP includes recent past and current loading pattern restrictions in Region I of the SFP and loading restrictions in Region II. The loading restrictions in Region II would remain in place with approval of the LAR, and the loading pattern restrictions in Region I would remain in place while the current Region I Caborundum equipped storage racks remain in the SFP. The NRC staff finds the licensee's application of operating experience to be acceptable.

3.4.2.3 Functional Requirements Analysis and Function Allocation

Because there are no changes to operator action, a full, functional, requirement analysis and function allocation was not necessary.

3.4.2.4 Staffing

Staffing and qualification are not affected by the proposed LAR. No new or additional staff are required, nor are there any new or additional qualifications required to perform the action sequence within the time constraints established.

3.4.2.5 Probabilistic Risk and Human Reliability Analyses

No probabilistic risk analysis was conducted and therefore no risk insights were obtained.

3.4.2.6 Human-System Interface Design

In their response to the staff's request for additional information (RAI), (dated September 6, 2012), Entergy stated that there will be no changes to physical interfaces, including monitoring instruments, boron concentration, or crane operation as a result of this LAR.

3.4.2.7 Procedure Design

The changes being made to the fuel movement procedures include:

a. Steps to ensure two empty storage rows are maintained between the fuel assemblies in the Carborundum equipped storage rack and an adjacent Metamic equipped rack, and

b. Upon completion of the removal of the Region I Carborundum equipped storage racks from the SFP, procedure instructions associated with the current loading pattern restrictions of the Region 1 Carborundum equipped racks will be removed. Regarding these procedure changes, Entergy states that fuel movement plans are developed by experienced and qualified personnel. Engineering Manual procedure EM-04-29, "Guidelines for preparing Fuel Move Plans," is the governing procedure for preparing for fuel moves. This procedure requires an independent review by another qualified person and ensures both the preparer of the fuel move, and the reviewer verify that the fuel move plan would result in approved storage patterns in accordance with technical specifications. The NRC staff finds the proposed procedure changes acceptable based on the licensee's use of Engineering Manual Procedure EM-04-29 which ensures verification of the fuel move plans.

3.4.2.8 Training Program and Simulator Design

In the RAI response, dated September 6, 2012, the licensee states that there are no specific changes to SFP loading training or qualifications as a result of this LAR. During installation of the new Metamic equipped fuel storage racks, current restrictions will remain for loading fuel in the Carborundum equipped fuel storage racks. The licensee's current training and qualifications appropriately address the current loading restrictions. The unrestrictive loading requirements for the new Metamic equipped fuel storage racks would not require any specific training of qualification changes.

3.4.2.9 Summary of Human Performance Review

Based on the evidence provided by, as well as the appropriate administrative controls being applied to procedures, training, and human-system interface design, and the application of industry and in-house operating experience, the NRC staff concludes that the proposed LAR is acceptable from the human performance point of view.

3.5 Steam Generator Tube Integrity and Chemical Engineering Review

3.5.1 Metamic™ Surveillance Program

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 Metamic™ known to perform effectively as a neutron absorber and Al 6061 is a marine-qualified alloy known for its resistance to corrosion.

The licensee proposes a surveillance program which consists primarily of monitoring the condition of representative coupon samples in order to assess the performance of the Metamic™ panels mounted on the spent fuel storage racks. The coupons to be used will be manufactured from the same material lot as the panels used in the storage racks.

The licensee proposes to perform periodic dimensional and visual checks to confirm the physical properties of the coupons, and to perform neutron attenuation testing to confirm the neutron absorption capabilities of the coupons. The objective of these examinations is to obtain the data necessary to assess the capability of the Metamic™ panels in the racks to continue to perform their intended function.

3.5.2 Program Description

The Metamic™ surveillance program includes 10 Metamic™ test coupons mounted on a coupon tree, placed in a designated cell in Region I, and surrounded by spent fuel. The coupons will be approximately 6" x 8"x 0.106", with a [[ ]] minimum B-10 areal density of [[ ]] 0.02944 g B-10/cm^2 [[ ]]. The nominal B-10 areal density loading corresponds to the nominal B_4C loading in Metamic™, and the nominal coupon thickness. The minimum B-10 areal density loading corresponds to the analysis.

The coupons will be positioned axially within the central eight feet (approximate) of the active fuel zone where the licensee stated that the gamma flux is expected to be reasonably uniform. The coupons will be mounted on the tree by way of threaded rods, which protrude from the tree and pass through holes in the coupons, washers and wing nuts. The coupons will not be sheathed. One coupon will be removed from the array and examined after 2, 5, 9, 13, 19, 26, 35, 44, 52, and 60 years of service life starting from the installation of the storage racks. During examination, certain physical quantities of the coupons will be measured from which the stability and integrity of the Metamic™ in the storage cells may be inferred.

To ensure that the coupons have experienced a slightly higher radiation dose than the panels in the racks, the licensee stated that the coupon tree will be surrounded by freshly discharged fuel assemblies after each of the first four offloads following installation. At the time of the first fuel offload, the four storage cells surrounding the tree shall be loaded with fresh, discharged fuel assemblies from among those which are not scheduled to be returned to the core. The licensee stated that an effort will be made to surround the coupon tree with freshly discharged fuel during each of the next three refueling outages by replacing the fuel assemblies initially placed in the four cells surrounding the tree with freshly discharged assemblies. The tree will remain in its original cell.

The licensee stated that the evaluation of the coupons removed will provide information on the effects of the radiation, thermal, and chemical environment of the pool and by inference, comparable information on the panels in the racks. Over the duration of the coupon testing program, the licensee stated that the coupons will have accumulated more radiation dose than the expected lifetime dose for normal storage cells. Additionally, coupons that have not been destructively analyzed may optionally be returned to the storage pool and remounted on the tree to be available for subsequent investigation of defects, should any be found.

3.5.3 Monitoring Changes in the Physical Properties and Testing of Coupons

The Metamic™ surveillance program is intended to monitor the changes in physical properties of the neutron absorber by performing the following measurements, on the preplanned scheduled noted above, on each coupon removed:

1. Visual observation and photography:
   Visual or photographic evidence of unusual surface pitting, blistering, corrosion or edge deterioration.

2. Neutron attenuation testing.

3. Dimensional measurements:
   a. Length
   b. Width
   c. Thickness

4. Weight and specific gravity:
   Unaccountable weight loss in excess of the measurement accuracy.

The licensee's acceptance criteria for neutron attenuation measurements to verify the continued presence of the boron, and thickness measurements to monitor potential swelling, are as follows:

- A decrease of no more than 5 percent Boron-10 content, as determined by neutron attenuation, is acceptable.
- An increase in thickness at any point should not exceed 10 percent of the initial thickness at that point.

The licensee stated that the remaining measurement parameters serve a supporting role. They should be examined for early indications of the potential onset of degradation that would suggest a need for further attention and a possible change in measurement schedule. These measurement parameters include: (1) visual or photographic evidence of unusual surface pitting, corrosion or edge deterioration, or (2) unaccountable weight loss in excess of the measurement accuracy.

Prior to installing the coupons in the SFP, each coupon will be pre-characterized. At a minimum, the coupons will be pre-characterized for weight, dimensions, and B-10 loading, which will be used as a baseline when determining if future measurements meet the acceptance criteria for neutron attenuation and thickness. If there are changes in excess of the above two acceptance criteria, the licensee is required to investigate and perform an engineering evaluation to confirm the indicated change(s). If the property changes are confirmed, then the licensee will perform an engineering evaluation to determine if further testing or corrective action is necessary.

After testing is completed, coupons that have not been destructively analyzed may be returned to the storage pool and remounted on the tree for future evaluation, if needed. The licensee stated that removed coupons will be reinserted to the SFP within three months of their removal from the pool. Coupons that are removed for testing will be cleaned and then wiped dry to remove the surface water prior to testing. The coupons will not be vacuum dried.

3.5.4 NRC Staff Evaluation

The NRC staff has reasonable assurance that the proposed testing frequency, methods, and acceptance criteria will identify any material property changes in the Metamic™ before significant degradation occurs. Furthermore, the testing frequency, methods, and acceptance criteria proposed by the licensee are in line with what the staff has approved for other plants using Metamic™ as a neutron absorber in the SFP. Given the relatively slow nature of material degradation seen in other neutron absorbing materials used in the SFP, favorable results from industry testing concerning the performance of Metamic™ in a SFP environment, and no operational experience concerning the degradation of the Metamic™ currently in use as a neutron absorbing material in other SFPs, the staff has reasonable assurance that the

licensee's surveillance program will allow enough time for the licensee to take corrective actions prior to any degradation challenging the minimum areal density of the Metamic^{TM} panels. Therefore, the staff finds that licensee's proposed surveillance program acceptable.

3.5.5 Summary of Review of Steam Generator Tube Integrity and Chemical Engineering Review

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

3.6 Accident Dose Review

The licensee evaluated the consequences of the dropped spent fuel assembly in the SFP for the proposed change by analyzing a potential impact on the replacement racks. The licensee asserts that the structural damage to the fuel building, pool liner, and fuel assembly resulting from a dropped fuel assembly striking the pool floor or another assembly located in the racks is primarily dependent on the mass of the falling object and drop height. The licensee declares that since these two parameters are not changed by the proposed TSs revision, the postulated structural damage to these items remains unchanged. As stated in its RAI response letter dated September 6, 2012, the licensee concludes that the consequences of the dropped spent fuel assembly in the SFP remains bounded by the design basis evaluation for the postulated cask drop accident, because the mass of the falling object and drop height are not changed.

The design-basis spent fuel cask drop analysis was found to be acceptable to the NRC staff by License Amendment No. 226 to Facility Operating License No. DPR-20 for PNP (ADAMS Accession No. ML072470676). In that amendment, the licensee determined that the evaluation of the radiological consequences for a postulated load drop of a loaded multi-assembly sealed basket transfer cask (MTC) from the main hoist bounds the radiological consequences from postulated load drops from the auxiliary hoist. Based on this information, the damage resulting from a dropping a singular spent fuel assembly in the SFP does not exceed that of 97-ton MTC design-basis drop. Therefore, the NRC staff finds the consequences of the dropped spent fuel assembly in the SFP to be bounded by the design-basis SFCD analysis, which is described in PNP FSAR Section 14.11, "Postulated Cask Drop Accidents." The NRC staff finds, with reasonable assurance that the licensee's estimates of the exclusion area boundary, low population zone, and control room dose will continue to comply with the dose criterion provided in 10 CFR 50.67, and, as well as, the accident specific dose guidelines specified in SRP 15.0.1. Therefore, the proposed change is acceptable with regard to the radiological consequences of postulated design basis accidents. The proposed amount of fuel damage assumed to occur within a fuel assembly located in the replacement storage racks was evaluated by the NRC staff and found to be acceptable.

3.6.1 Summary of Accident Dose Review

As described above, the NRC staff reviewed the justification and assumptions used by the licensee to assess the radiological impacts of replacement of the Region I SFP storage racks

with new neutron absorber Metamic-equipped racks at PNP. The staff finds that the licensee used methods consistent with regulatory requirements and guidance identified in Section 2.0 above. The staff finds with reasonable assurance that the licensee's estimates of the exclusion area boundary, low-population zone, and control room doses will continue to comply with the 10 CFR Section 50.67 criteria. Therefore, the proposed change is acceptable with regard to the radiological consequences of postulated design basis accidents.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the 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 (77 FR 33246, dated June 5, 2012). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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) there is reasonable assurance that 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.

7.0 REFERENCES

(1) Title 10 of the *Code of Federal Regulations*, Part 50, Section 68, "Criticality Accident Requirements."

(2) L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," Nuclear Regulatory Commission, Rockville, MD, August 1998.

(3) K. Wood, DSS-ISG-2010-1, "Draft Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools." ADAMS Accession No. ML102220567, Nuclear Regulatory Commission, Rockville, MD, August 2010.

(4) Letter from Anthony J. Vitale, Entergy Nuclear Operations, Inc. (PNP 2012-005) to NRC Document Control Desk, "License Amendment Request-Replacement of Spent

Fuel Pool Region I Storage Racks," dated February 28, 2012 (ADAMS Accession Nos. ML12061A288 through ML12061A290).

(5) Letter from Anthony J. Vitale, Entergy Nuclear Operations, Inc. (PNP 2012-071), to NRC Document Control Desk, "Response to Request for Additional Information-License Amendment-Replacement of Spent Fuel Pool Region I Storage Racks," dated September 6, 2012 (ADAMS Accession No. ML122541048).

(6) Letter from Anthony J. Vitale, Entergy Nuclear Operations, Inc. (PNP 2012-096), to NRC Document Control Desk, "Response to Follow-up Request for Additional Information - License Amendment Request-Replacement of Spent Fuel Pool Region I Storage Racks" dated November 7, 2012 (ADAMS Accession No. ML12313A343).

(7) Letter from Anthony J. Vitale, Entergy Nuclear Operations, Inc. (PNP 2012-096), to NRC Document Control Desk, "Response to Third Request for Additional Information - License Amendment Request-Replacement of Spent Fuel Pool Region I Storage Racks (ME8074)" dated November 7, 2012 (ADAMS Accession No. ML12313A343).

(8) Letter from B. K. Grimes, Nuclear Regulatory Commission, Position Paper: "Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

(9) U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition - Design of Structures, Components, Equipment, and Systems – Other Seismic Category I Structures," NUREG-0800, Section 3.8.4, Revision 3, March 2007.

(10) U.S. Nuclear Regulatory Commission, "A Pilot Probabilistic Risk Assessment of a DryCask Storage System at a Nuclear Power Plant," NUREG-1864, March 2007.

(11) U.S. Nuclear Regulatory Commission, "Safety Evaluation Report – Docket No. 72-1014 – Holtec International HI-STORM 100 Cask System –Certificate of Compliance No. 1014 – Amendment 7" (ADAMS Accession No. ML093620075).

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The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to the Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a redacted publicly available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 250 to DPR-20
2. Non-Proprietary Safety Evaluation
3. Proprietary Safety Evaluation

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