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

April 29, 2011

Mr. Paul A. Harden
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING THE SPENT FUEL POOL RERACK (TAC NO. ME1079)

Dear Mr. Harden:

The Commission has issued the enclosed Amendment No. 173 to Renewed Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit No. 2 (BVPS-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 9, 2009, as supplemented by letters dated June 15, 2009, January 18, 2010, March 18, 2010, May 3, 2010, May 21, 2010, June 1, 2010, August 9, 2010, October 7, 2010, October 18, 2010, January 5, 2011, February 18, 2011, March 18, 2011, and March 21, 2011.

The amendment modified TSs to support the replacement of existing Boraflex neutron absorber fuel storage racks in the BVPS-2 spent fuel pool with new high density, Metamic neutron absorber fuel storage racks, which will increase the total storage locations from 1,088 to 1,690.

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

Sincerely,

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosures:
1. Amendment No. 173 to NPF-73
2. Safety Evaluation

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FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 173
License No. NPF-73

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee), dated April 9, 2009, as supplemented by letters dated June 15, 2009, January 18, 2010, March 18, 2010, May 3, 2010, May 21, 2010, June 1, 2010, August 9, 2010, October 7, 2010, October 18, 2010, January 5, 2011, February 18, 2011, March 18, 2011, and March 21, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-73 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 173, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC 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 its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: April 29, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 173

RENEWED FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4	4

Replace the following pages of 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
3.7.14-1	3.7.14-1
3.7.14-2	3.7.14-2
3.7.14-3	3.7.14-3
----------	3.7.14-4
----------	3.7.14-5
----------	3.7.14-6
----------	3.7.14-7
4.0-2	4.0-2
4.0-3	4.0-3
----------	4.0-4

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the term or conditions of any lease agreements executed as part of these transactions; (ii) the BVPS Operating Agreement, (iii) the existing property insurance coverage for BVPS Unit 2, and (iv) any action by a lessor or others that may have adverse effect on the safe operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

FENOC is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 173, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Storage

LCO 3.7.14

The combination of initial enrichment and burnup of each fuel assembly stored in the spent fuel storage pool shall be within the limits specified in Table 3.7.14-1A (Unit 1); for Unit 2:

Table 3.7.14-1B or in accordance with Specification 4.3.1.1.e, for the fuel assemblies stored in a Boraflex rack,
and

Table 3.7.14-1C, Table 3.7.14-1D, Table 3.7.14-1E, and in accordance with Specification 4.3.1.1.e, for the fuel assemblies stored in a Metamic rack.

- NOTE -

For Unit 2 only, Technical Specification requirements applicable to the fuel storage pool are also applicable to the fuel cask area when a fuel assembly is in the fuel cask area during the installation phase of the Unit 2 reracking project.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	------------------------------ **- NOTE –** LCO 3.0.3 is not applicable. ------------------------------	
		Initiate action to move the noncomplying fuel assembly to a location that complies with Table 3.7.14-1A (Unit 1); LCO 3.7.14 (Unit 2).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.14-1A (Unit 1); LCO 3.7.14 (Unit 2).	Prior to storing the fuel assembly in the spent fuel storage pool

Table 3.7.14-1A (page 1 of 1)
(Unit 1 Spent Fuel Pool Storage)

Fuel Assembly Minimum Burnup versus U-235 Initial Enrichment for Storage in Spent Fuel Rack Regions 1, 2, and 3

Nominal Enrichment (w/o U-235)	**Region 3** Assembly Discharge Burnup (MWD/MTU)	**Region 2** Assembly Discharge Burnup (MWD/MTU)	**Region 1** Assembly Discharge Burnup (MWD/MTU)
2.0	0	2585	0
2.348	0	7911 (calculated)	0
2.5	1605	9551	0
3.0	6980	15784	0
3.5	11682	21643	0
4.0	16239	27260	0
4.5	20672	33710	0
5.0	25000	40000	0

NOTES:

Region 2: The data in the above Table may be interpreted linearly or may be calculated by the conservative equation below. This equation provides a linear fit to the design burnup limits.

Minimum Burnup, MWD/MTU = 12,100 * E% - 20,500

Where E = Enrichment ($E \leq 5\%$)

Region 3: The data in the above Table may be interpreted linearly or may be calculated by the conservative equation below. This equation provides a best fit to the design burnup limits.

Minimum Burnup, MWD/MTU = $-480 * (E\%)^2 + 12{,}900 * E\% - 27{,}400$

Where E = Enrichment ($E \leq 5\%$)

Table 3.7.14-1B (page 1 of 1)
(Unit 2 Spent Fuel Pool Storage - Boraflex Rack)

Fuel Assembly Minimum Burnup versus Initial Enrichment for the
"All-Cell" Storage Configuration

Initial Enrichment (w/o U-235)	Burnup (MWD/MTU)
1.856	0
3.000	13,049
4.000	23,792
5.000	34,404

NOTES:

Any fuel assembly may be loaded at the interface with another configuration.

The required minimum assembly burnup (in MWD/MTU) for an assembly of a given initial enrichment may be calculated using the equation below, where E% is the assembly initial enrichment in weight percent U-235.

Assembly Burnup = $78.116(E\%)^3 - 1002.647(E\%)^2 + 14871.032(E\%) - 24649.599$

Where E = Enrichment ($E \leq 5\%$)

Table 3.7.14-1C (page 1 of 1)
(Unit 2 Spent Fuel Pool Storage - Metamic Rack)

Fuel Assembly Minimum Burnup with Enriched Blankets versus U-235 Initial Enrichment for Storage in Unit 2 Spent Fuel Rack Regions 1, 2, and 3

	Region 3	**Region 2**	**Region 1**
Nominal Enrichment (w/o U-235)	Assembly Discharge Burnup (MWD/MTU)	Assembly Discharge Burnup (MWD/MTU)	Assembly Discharge Burnup (MWD/MTU)
2	640	11140	0
2.5	8020	19530	0
3	14990	27500	0
3.5	21570	35060	0
4	27760	42200	0
4.5	33550	48920	0
5	38940	55230	0

NOTES:

Region 2: The equation below can be used to determine intermediate burnup limits.

Minimum Burnup, MWD/MTU = $- 832.4(E\%)^2 + 20523(E\%) - 26578$

Where E = Enrichment ($E \leq 5\%$)

Region 3: The equation below can be used to determine intermediate burnup limits.

Minimum Burnup, MWD/MTU = $- 793(E\%)^2 + 18315(E\%) - 32814$

Where E = Enrichment ($E \leq 5\%$)

Table 3.7.14-1D (page 1 of 1)
(Unit 2 Spent Fuel Pool Storage - Metamic Rack)

Fuel Assembly Minimum Burnup with Natural Blankets versus U-235 Initial Enrichment for Storage in Unit 2 Spent Fuel Rack Regions 1, 2, and 3

	Region 3	**Region 2**	**Region 1**
Nominal Enrichment (w/o U-235)	Assembly Discharge Burnup (MWD/MTU)	Assembly Discharge Burnup (MWD/MTU)	Assembly Discharge Burnup (MWD/MTU)
2	650	10990	0
2.5	8060	19270	0
3	15060	27130	0
3.5	21660	34560	0
4	27850	41560	0
4.5	33630	48140	0
5	39010	54280	0

NOTES:

Region 2: The equation below can be used to determine intermediate burnup limits.

Minimum Burnup, MWD/MTU = $-855.3(E\%)^2 + 20418(E\%) - 26425$

Where E = Enrichment ($E \leq 5\%$)

Region 3: The equation below can be used to determine intermediate burnup limits.

Minimum Burnup, MWD/MTU = $-813.4(E\%)^2 + 18481(E\%) - 33063.4$

Where E = Enrichment ($E \leq 5\%$)

Table 3.7.14-1E (page 1 of 1)
(Unit 2 Spent Fuel Pool Storage - Metamic Rack)

Fuel Assembly Minimum Burnup with No Blankets versus U-235 Initial Enrichment for Storage in Unit 2 Spent Fuel Rack Regions 1, 2, and 3

	Region 3	**Region 2**	**Region 1**
Nominal Enrichment (w/o U-235)	Assembly Discharge Burnup (MWD/MTU)	Assembly Discharge Burnup (MWD/MTU)	Assembly Discharge Burnup (MWD/MTU)
2	1030	11190	0
2.5	8170	19460	0
3	15190	27290	0
3.5	22080	34690	0
4	28840	41650	0
4.5	35470	48170	0
5	41970	54260	0

NOTES:

Region 2: The equation below can be used to determine intermediate burnup limits.

Minimum Burnup, MWD/MTU = $- 873.1(E\%)^2 + 20467(E\%) - 26250$

Where E = Enrichment ($E \leq 5\%$)

Region 3: The equation below can be used to determine intermediate burnup limits.

Minimum Burnup, MWD/MTU = $- 257.4(E\%)^2 + 15449(E\%) - 28840$

Where E = Enrichment ($E \leq 5\%$)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

Unit 2
K_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR,

c. Unit 2
$K_{eff} \leq 0.95$ if fully flooded with water borated to 495 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR,

d. Unit 1
A nominal center to center distance between fuel assemblies placed in the fuel storage racks of 10.82 inch for Region 1, with 9.02 inch for Regions 2 and 3,

Unit 2
A minimum center to center distance between fuel assemblies placed in the fuel storage racks of 10.4375 inches (Boraflex rack), 9.03 inches (Metamic rack), and

e. Fuel assembly storage shall comply with the requirements of LCO 3.7.14, "Spent Fuel Pool Storage",

Unit 2
Boraflex Rack
New or partially spent fuel assemblies within the limits of Table 3.7.14-1B may be allowed unrestrictive storage in the fuel storage racks, and

New or partially spent fuel assemblies not within the limits of Table 3.7.14-1B will be stored in compliance with NRC approved WCAP-16518-P, "Beaver Valley Unit 2 Spent Fuel Rack Criticality Analysis," Revision 2, July 2007.

Unit 2
Metamic Rack
New or partially spent fuel assemblies within the limits of Table 3.7.14-1C, Table 3.7.14-1D, and Table 3.7.14-1E may be stored in the fuel storage racks, provided:

1. Region 1 storage cells are located on the periphery of each rack (outer row only) and are therefore separated from other Region 1 cells in adjacent racks by the 1.5 inch minimum gap between the racks. Region 1 cells are additionally separated

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

from other Region 1 cells within the same rack by Region 2 cells (including a Region 2 cell in the diagonal direction). Since Region 1 cells are qualified for the storage of fresh fuel, any fuel assembly (fresh or burned) meeting the maximum enrichment requirement may be stored in a Region 1 location,

2. Region 2 cells are located on the rack periphery (outer row) interspaced with (separating) Region 1 cells and are also located in the second row of cells (from the outside of the rack) separating the Region 1 cells from the Region 3 cells,

3. Region 3 cells are located on the interior of the rack and are prohibited from being located in the outer two rows of the rack, and

4. Two empty rows of storage locations shall exist between the fuel assemblies in a Boraflex rack and the fuel assemblies in an adjacent Metamic rack in the fuel storage pool.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.00 weight percent with a tolerance of + 0.05 weight percent,

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.12 of the Unit 1 UFSAR and Section 9.1 of the Unit 2 UFSAR,

c. <u>Unit 1</u>
$K_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.12 of the UFSAR,

<u>Unit 2</u>
$K_{eff} \leq 0.95$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR, and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage

Unit 1
The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 750 feet - 10 inches.

Unit 2
The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 751 feet - 3 inches.

4.3.3 Capacity

Unit 1
The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1627 fuel assemblies.

Unit 2
The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1088 fuel assemblies (Boraflex racks), 1690 fuel assemblies (Metamic racks).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 173 TO RENEWED

FACILITY OPERATING LICENSE NO. NPF-73

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By application dated April 9, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091210251), as supplemented by letters dated June 15, 2009 (ADAMS Accession No. ML091680614), January 18, 2010 (ADAMS Accession No. ML100191805), March 18, 2010 (ADAMS Accession No. ML100820165), May 3, 2010 (ADAMS Accession No. ML101260059), May 21, 2010 (ADAMS Accession No. ML101460057), June 1, 2010 (ADAMS Accession No. ML101610118), August 9, 2010 (ADAMS Accession No. ML102240256), October 7, 2010 (ADAMS Accession No. ML102860124), October 18, 2010 (ADAMS Accession No. ML102940454), January 5, 2011 (ADAMS Accession No. ML110110217), February 18, 2011 (2 letters) (ADAMS Accession No. ML110530463 and ML110540328), March 18, 2011 (ADAMS Accession No. ML110800122), and March 21, 2011 (ADAMS Accession No. ML110800570), FirstEnergy Nuclear Operating Company (the licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit No. 2 (BVPS-2). The supplements dated June 15, 2009, January 18, 2010, March 18, 2010, May 3, 2010, May 21, 2010, June 1, 2010, August 9, 2010, October 7, 2010, October 18, 2010, January 5, 2011, February 18, 2011, March 18, 2011, and March 21, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 11, 2010 (75 FR 11566).

The changes would revise the BVPS-2 TSs to support the replacement of existing Boraflex neutron absorber fuel storage racks in the BVPS-2 spent fuel pool (SFP) with new high density, Metamic neutron absorber fuel storage racks, which will increase the total storage locations from 1,088 to 1,690.

1.1 Background

The existing BVPS-2 SFP storage racks are flux trap style[1] spent fuel storage racks, consisting of 17 individual spent fuel storage racks/modules that have capacity for 1,088 fuel assemblies. The 17 existing racks will be removed and replaced by 15 high density racks, increasing the storage capacity to 1,690 storage locations. All of the new high density racks are non-flux-trap racks and are designated in a mixed-zone three-region (MZTR) array, where loading patterns are used to control criticality. All SFP storage high density racks are freestanding and self-supporting. The principal construction materials for the high density racks are stainless steel sheet. The only non-stainless material utilized in the high density racks is the neutron absorber material, which is a boron carbide (B_4C) and aluminum metal matrix composite available under the patented product name Metamic.

Each rack module consists of a checkerboard arrangement of fabricated and developed cells that are formed by welding diagonally adjacent fabricated boxes together. Each stainless steel box has a nominal inside dimension of 8.8 inches and a nominal wall thickness of 0.075-inch. A Metamic plate that is nominally 7.5 inches wide, 146 inches long, and 0.106-inch thick with a nominal B^{10} areal density of 0.031 grams B^{10} per square centimeter (g/cm^2) is held in place on the outside faces of each fabricated box by a nominally 0.035-inch thick stainless steel wrapper that is welded to the box. The boxes are welded together such that there is a nominal 9.03 inches center-to-center spacing between adjacent storage locations. Corner angles and filler panels, with poison panels, are used as needed to complete the outer perimeter of each rack.

During installation of the new racks into the SFP, a new Metamic rack will temporarily be placed in the cask pit to provide additional fuel storage space. This is needed to provide enough fuel storage space to permit emptying the existing Boraflex racks for removal. Only fuel assemblies with at least 18 months of cooling time may be placed in the rack in the cask pit. As part of the new rack installation sequence, all fuel in the rack in the cask pit will eventually be moved into the SFP and the rack moved to its final position in the SFP. Following the movement of fuel from the SFP to the rack temporarily placed in the cask pit, the emptied existing rack will be removed from the SFP. A cover will be placed over the loaded rack in the cask pit to protect the rack and fuel during the movement of the racks. A new rack will then be installed into the SFP and loaded with fuel from existing racks. The emptied existing rack will then be replaced by a new rack. This fuel shuffling and rack removal and installation will continue until all the existing racks have been replaced with new racks. Once this has been completed, the fuel and rack temporarily placed in cast pit will be moved to the SFP.

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[1] Flux trap style racks contain a water-filled gap with neutron absorber on both sides, called a flux trap, between adjacent fuel storage locations.

2.0 REGULATORY EVAULATION

A short description containing the purpose and function of the SFP and the SFP Cooling and Cleanup System (SFPCCS) is provided below.

2.1 Description of Spent Fuel Storage Area

Chapter 9 of the BVPS-2 Updated Final Safety Analysis Report (UFSAR) describes the BVPS-2 spent fuel storage area, which is divided into three areas, separated by a stainless steel-lined concrete wall, with a removable gate provided between each area to allow movement of fuel elements between them. Each gate is equipped with an inflatable seal to prevent leakage from one area to another. The three areas are defined as the fuel cask area, the SFP, and the fuel transfer canal. Each area is lined with stainless steel and is normally filled with borated demineralized water. An important function of the SFP area systems is to maintain the fuel in a subcritical condition with the effective multiplication factor, k-effective (k_{eff}) less than or equal to 0.95.

The fuel cask area consists of two locations at different elevations, which allow for the safe movement of spent fuel into the shipping cask. The lower elevation provides a sufficient height of water above the fuel being transferred to allow for adequate shielding, while the upper elevation limits the potential spent fuel cask drop height and allows for preliminary decontamination using a floating spray ring.

The SFP houses the spent fuel storage racks. The SFP is designed such that the water level in the pool cannot be decreased below the top of the fuel stored in the spent fuel racks and sized to accommodate the storage of a minimum of one full core in the event the reactor must be emptied of fuel at any time during BVPS-2 life.

The fuel transfer canal houses the fuel transfer system, which provides for transfer of new and spent fuel elements between the fuel building and reactor containment during refueling. Spent fuel is transported between the fuel transfer canal, SFP, and the fuel cask area by the fuel building motor-driven platform crane.

2.2 Description of the SFPCCS

The BVPS-2 SFPCCS is designed to remove the heat generated by spent fuel assemblies stored in the SFP, as described in the Chapter 9 of the BVPS-2 UFSAR. The cooling portion of the SFPCCS is safety-related and designed to Seismic Category I, Quality Group C criteria. The system has two trains of cooling equipment (pumps and heat exchangers) capable of interconnection and designed for continuous use. The SFPCCS is designed to remove the decay heat of a normal full core offload and an abnormal full core offload with all available storage positions in the SFP filled.

2.3 Proposed TSs Changes

TS 3.7.14, "Spent Fuel Pool Storage," would be revised to show that Table 3.7.14-1B applies to the existing racks, those containing Boraflex, and adds Table 3.7.14-1C that applies to the high density racks, those containing Metamic. This TS would also be revised to show the applicability of TS 4.3.1.1.e for both types of racks. The Required Action and Surveillance are being simplified for

BVPS-2 by referring to the requirements of the Limiting Condition for Operation (LCO). This simplification is designed to reduce human performance errors associated with interpretation of the requirements imposed on the two different types of racks for BVPS-2. A Note would also be added that extends the SFP to include the fuel cask area for only BVPS-2. The Note would also state that it is applicable only during the installation phase of the reracking project. The Note will make it clear that the BVPS-2 fuel cask area is temporarily included in the applicability of TSs 3.7.12, Supplemental Leak Collection and Release System, 3.7.15, Fuel Storage Pool Water Level, 3.7.16, Fuel Storage Pool Boron Concentration and 4.3.2, Drainage.

TS 4.3.1, "Criticality," would be revised to show: 1) the boron concentration necessary to maintain k_{eff} ≤ 0.95 when the SFP is fully flooded with borated water, 2) the minimum center to center distance between fuel assemblies in each type of rack, and 3) the fuel storage constraints for each type of rack. The minimum boron concentration necessary to maintain $k_{eff} \leq 0.95$ when the SFP is fully flooded with borated water is 495 parts per million (ppm), which is the value from the new criticality analysis. This TS would also contain a requirement to have two empty rows of storage locations between the fuel assemblies stored in adjacent Boraflex and Metamic racks in the SFP during the installation phase of the reracking project.

TS 4.3.3, "Capacity," would be revised to show the maximum capacity of the BVPS-2 SFP with each type of rack.

2.4 Regulatory Requirements and Guidance

The following explains the applicability of GDC for BVPS-2. The construction permit for BVPS-2 was issued by the Atomic Energy Commission on May 3, 1974, and the operating license was issued on August 14, 1987. The plant GDC are discussed in the UFSAR, Chapter 3, "Design of Structures, Components, Equipment, and Systems," which describes how the design conforms to "GDC, Appendix A of 10 CFR [Part] 50, as amended through October 27, 1978." Furthermore, UFSAR section 15.1.5.3 page 15.1-22 notes that BVPS has Alternative Source Term and thus meets the later Dec. 23, 1999 GDC 19.

As discussed in the UFSAR, the licensee for BVPS-2 has made some changes to the facility over the life of the unit that has committed them to some updated GDCs in 10 CFR Part 50, Appendix A. The extent to which the updated Appendix A GDC have been invoked can be found in specific sections of the UFSAR and in other BVPS-2 licensing basis documentation, such as license amendments.

The following GDCs, which are applicable to BVPS-2, pertain to this license amendment:

2.4.1 GDC for BVPS-2

- 10 CFR 50.55a and GDC 1, as they relate to safety-related structures being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 2, as it relates to the design of the safety-related structures being capable to withstand the most severe natural phenomena such as wind, tornadoes, floods, and earthquakes and the appropriate combination of all loads.

- GDC 4, as it relates to safety-related structures being protected against dynamic effects, such as the loads imposed on structures by postulated missiles.

- GDC 62, "Preventing of Criticality in Fuel Storage and Handling," states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. The licensee must limit the potential for criticality in the fuel handling building and storage system by physical systems or processes.

- Section 9.1.3 of the BVPS-2 UFSAR states that the SFPCCS is designed in accordance with GDC 63, as it relates to monitoring systems provided to detect conditions that could result in a loss of decay heat removal capability, to detect excessive radiation levels, and to initiate appropriate safety actions. BVPS-2 UFSAR section 9.1.3.5 identifies the required instrumentation to meet the GDC. No changes have been proposed to the instrumentation and alarms associated with the SFPCCS.

2.4.2 Applicable CFRs and Guidance

While the technical requirements for specific areas of the review are discussed in the individual sections of the safety evaluation, the overall regulatory requirements and guidance on which the NRC staff based its acceptance of the LAR are provided below:

- 10 CFR 50.68(b)(1) states, "Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water."

- 10 CFR 50.68(b)(4) states, "If no credit for soluble boron is taken, the k_{eff} 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 unborated water. If credit is taken for soluble boron, the k_{eff} 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_{eff} must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

The BVPS-2 SFP nuclear criticality safety (NCS) analysis does credit soluble boron. Therefore, the regulatory requirement for the BVPS-2 SFP k_{eff} is to remain less than or equal to 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k_{eff} must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

10 CFR 50.36 contains the requirements for the content of TS. Pursuant to 10 CFR 50.36(c), TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) Surveillance Requirements; (4) design features; and (5) administrative controls.

10 CFR 50.36(c)(2)(ii) lists the criteria used to determine whether or not LCOs must be established in TS for items related to plant operation. If the item falls in to one of the

four categories below, an LCO must be established in TS to ensure the lowest functional capability or performance level of equipment required for safe operation of the facility will be met. The four criteria are:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3 A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4 A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

10 CFR 50.36(c)(4) states, "Design features. 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 section."

- Standard Review Plan (SRP) 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling Review Responsibilities," states, in part, the review should verify that the storage facilities maintain the new and spent fuel in subcritical arrays during all credible storage conditions and that the new and spent fuel will remain subcritical during fuel handling, in accordance with GDC 62 and 10 CFR 50.68.

- SRP 9.1.2, "Spent Fuel Storage," states, in part, that the review should ensure that there are no potential mechanisms that will: (1) alter the dispersion of any strong fixed neutron absorbers, and/or (2) cause physical distortion of the tubes retaining the stored fuel assemblies.

- Guidance for SFP systems is available in Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," Revision 1, June 1981, of NUREG-0800, which is referenced in the BVPS-2 UFSAR, Section 9.1.3.1. With respect to SFP cooling, the guidelines of SRP Section 9.1.3 were based on GDC 44 from Appendix A to 10 CFR Part 50. The criteria of GDC 44 included the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions, assuming a single active failure of a component.

The licensing basis for the SFPCCS at BVPS-2 was a recently revised by Amendment 126 to the BVPS operating license, which implemented a reduction in the minimum decay time required prior to fuel movement from 150 hours to 100 hours. The impact of this change on

SFP temperature was evaluated and an increase in the maximum bulk SFP water temperature limits was approved. The evaluations of SFP cooling for Amendment No. 126 considered a single failure in conjunction with a normal refueling and consider an abnormal heat load, consistent with the guidelines of SRP Section 9.1.3. However, the licensing basis established by Amendment No. 126 considers the heat load imposed by the routine full core offload rather than the much smaller refueling batch offload considered in SRP Section 9.1.3 guidelines. Furthermore, in Amendment No. 126, the NRC staff accepted relaxations relative to the guidelines of SRP Section 9.1.3, such as administrative controls on the rate of fuel transfer and a higher potential peak SFP temperature based on the rapid decrease in heat load associated with the full core offload. Lastly, in the review of Amendment No. 126, the NRC staff considered the capability of make-up water supplies to maintain SFP coolant inventory in the unlikely event the SFP cooling system would not be available.

To accomplish the movement of the new and existing spent fuel storage racks, the licensee proposed using a temporary crane installed above the SFP to enable the racks to be moved into and out of the pool. In the LAR, the licensee compared the temporary crane and heavy load handling associated with the installation and removal of the racks against the criteria provided in NUREG-0612, "Control of Heavy Loads." Section 9.1.5 of the BVPS-2 UFSAR references NUREG-0612 guidelines for the control of heavy loads. The guidelines provided in Sections 5.1.1 and 5.1.2 of NUREG-0612 apply to the temporary crane and heavy load handling activities associated with the rerack of the BVPS-2 SFP.

Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis", Revision 2, March 2007, provides guidance on the design requirements for spent fuel cooling, protection from damage, and other systems that operate in the SFP area. Section 9.1.3 of the BVPS-2 UFSAR states that the SFPCCS is designed in accordance with RG 1.13.

The NRC staff's acceptance criteria specific to the design of spent fuel racks can be found in Appendix D, "Guidance on Spent Fuel Pool Racks," of SRP Section 3.8.4, Revision 2, March 2007. Additional 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" (the OT Position Paper or Reference 13), as revised by letter dated January 18, 1979. These two letters were subsequently numbered NRC Generic Letter (GL) 1978-11, "Review and Acceptance of Spent Fuel Storage and Handling Applications," and GL 1979-04, "Modifications to NRC Guidance on 'Review and Acceptance of Spent Fuel Storage and Handling Applications,'" respectively.

Sections 5.1.1 and 5.1.2 of NUREG-0612 apply to the temporary crane and heavy load handling activities associated with the rerack of the BVPS-2 SFP.

Amendment No. 121 to Operating License NPF-73, dated August 30, 2001, contains the most recent NRC staff review of the licensee's dose analysis of the fuel handling accident (FHA) at BVPS-2 (ADAMS Accession No. ML012330496). Amendment No. 121 revised BVPS-2 FHA analysis to implement an alternative radiological source term. The regulatory guidance and requirements for which the NRC staff based its acceptance are:

- 10 CFR 50.67, "Accident source term;"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000;
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000; and
- UFSAR Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents".

The NRC has issued similar license amendments for SFP reracking requests for Waterford Steam Electric Station, Unit 3, on July 10, 1998 (ADAMS Accession No. ML021790559), Turkey Point Nuclear Plant, Units 3 and 4, on July 17, 2007 (ADAMS Accession No. ML071800198), and Cooper Nuclear Station on September 6, 2007 (ADAMS Accession No. ML072130023).

3.0 TECHNICAL EVALUATION

3.1 SFP Criticality Analysis Methodology

Currently, there is not a generic methodology for performing SFP criticality analyses. The NRC staff issued an internal memorandum on August 19, 1998, containing guidance for performing the review of SFP criticality analysis (Reference 21). The memorandum is known colloquially as the "Kopp Letter." While the Kopp Letter does not specify a methodology, it does provide some guidance on more salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), borated and unborated SFPs.

The NCS analysis supporting the licensee's LAR was performed by HOLTEC International Inc. (Holtec) and documented in HI-2084175, Revision 6, "Licensing Report for Beaver Valley Unit 2 Rerack," (Enclosure B of Reference 10). In Enclosure B of Reference 10, the k_{eff} was determined, with no soluble boron in the SFP, to be less than 0.995 at a 95 percent probability, 95 percent confidence level, providing approximately 0.005 Δk_{eff} of reserved analytical margin to the 10 CFR 50.68 requirement for k_{eff} to be less than 1.0. Additionally, the soluble boron required to keep k_{eff} no greater than 0.95 at a 95 percent probability, 95 percent confidence level is no greater than 1,212 ppm, which is significantly less than the 2,000 ppm of soluble boron required by the BVPS-2 SFP TS.

The methodology employed in HI-2084175(P), Rev. 6, included:

- Depletion calculations are performed using conservative reactor operating parameters.
- A limiting fuel bundle design was identified.
- Conservative axial burnup distributions were used that were derived from individual assembly axial burnup distributions from simulations of actual BVPS operating cycles.
- A reactivity control penalty was calculated and used to conservatively incorporate the affects of burnable absorbers and water displacement rods on fuel composition calculations.

3.1.1 Computer Code Validation

For use in NCS analyses, the ability of a calculational methodology to accurately predict the k_{eff} of a system must be well understood. The understanding of a calculational methodology's bias in predicting a given system's k_{eff} is obtained through the validation process. Validation includes identification of the difference between calculated and experimental results. This difference, called the bias, and the uncertainty associated with the bias are used in combination with additional biases and uncertainties to determine a given system's k_{eff} with a 95 percent probability, 95 percent confidence level. Therefore, the validation of the criticality codes is necessary to demonstrate compliance with the 10 CFR 50.68 regulatory requirements.

In Enclosure B of Reference 10, both computer codes MCNP4a Monte Carlo code and CASMO-4 were used. The MCNP4a for k_{eff} calculations with primarily ENDF/B-V continuous energy nuclear data libraries distributed with MCNP. The code validation is presented in Enclosure E of Reference 10, Holtec Report HI-2094486, Rev. 0. EGC's code validation used NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Methodology," (Reference 22) as guidance. The MCNP4a validation was supplemented with the licensee's response to the RAI-38 in Attachment 1 of Reference 18.

As documented in Enclosure E of Reference 10, the validation of MCNP4a for the LAR was performed with a set of 243 critical experiments, including 48 low enrichment uranium experiments, 39 "non-HTC" mixed plutonium and uranium oxide (MOX) experiments, and 156 "HTC" MOX experiments from NUREG/CR-6979, Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data (Reference 23). The MOX experiments are used to approximate the plutonium present in U^{235} spent fuel. The "HTC" MOX experiments in NUREG/CR-6979 were specifically designed to represent the uranium and plutonium proportions present in U^{235} spent fuel. The "non-HTC" MOX experiments were not designed to represent the uranium and plutonium proportions present in U^{235} spent fuel, and therefore are not as representative of U^{235} spent fuel for validation analysis. These experiments cover a range of values in several key parameters to ensure that the safety case models are within the area of applicability of the validation suite. As stated in the licensee's response to RAI-38, the validation set was reduced to 165 critical experiments, eliminating experiments that were of questionable similarity to the safety analysis models. The selected set of critical experiments appears to be from sufficiently diverse sources and, with exception of fission products, provides a good basis for validation of the method used to compute k_{eff} values. Fission product k_{eff} validation is identified by the applicant as a validation gap. The analysis uses 12% of the fission product worth as an uncertainty to cover the fission product validation gap and the use of some "lumped fission products." The value is based on the analysts' review of fission product nuclear data. This approach used to determine the uncertainty and the derived values are considered adequate.

A comparison of the ranges of key parameters for the critical experiments and the safety analysis models is provided in Attachment 1 of Reference 10. This comparison adequately demonstrates that the safety analysis models are within the area of applicability of the critical experiments. Key parameters evaluated included uranium enrichment, plutonium content, fuel rod outer diameter and rod pitch, fuel density, soluble boron concentration, neutron poisons, interstitial and reflecting materials, and energy of average lethargy of neutrons causing fission.

The statistical methods used for trending analysis and bias and bias uncertainty determination are described in Enclosure E of Reference 10. The statistical methods used are from NUREG/CR-6698 (Reference 22) and NUREG/CR-6361(Reference 24) and are appropriate for this LAR. Trending analysis of key parameters was performed and, where statistically significant trends were identified, taken into account for bias and bias uncertainty determination. Significant trends were identified for soluble boron concentration, and plutonium content (g Pu per g Pu+U). The most restrictive bias and uncertainty obtained using a subset 81 experiments that did not include soluble boron was 0.0029 ± 0.0078 Δk. This bias and uncertainty was applied to calculated results for determination of the burnup credit loading curve. A boron concentration dependent bias was also calculated that demonstrates that the bias plus 95/95 bias uncertainty applicable to calculations with soluble boron is fairly insensitive to the soluble boron concentration. The bias varies about 130 percent millirho (pcm) over a soluble boron range of 500 to 1200 ppm boron. While it does not appear that the soluble boron trend bias and bias uncertainty were applied to the soluble boron concentrations calculated to meet a maximum reactivity of 0.945, the soluble boron calculation that was performed is more than conservative enough to offset this adjustment. For details, see Section 4.7.12 in Enclosure B of Reference 10. The MCNP4a bias and bias uncertainty was appropriately included in the calculation of the initial enrichment and burnup combinations.

In Enclosure B of Reference 10, the CASMO-4 code was used to deplete the fuel assembly lattices to generate burned fuel compositions. CASMO-4 is a multi-group, two-dimensional transport theory code with an in-rack geometry option where typical storage rack geometries can be defined on an infinite lattice basis. The library files used in the evaluation are the standard CASMO-4 70-neutron-energy-group library based on ENDFB-IV. Consistent with guidance in the Kopp Letter (Reference 21), the licensee adopted 5% of the reactivity decrement uncertainty from fresh fuel to the burnup of interest to cover lack of fuel composition calculation validation.

3.2. SFP

3.2.1 SFP Mechanical Uncertainties

Manufacturing tolerances within the SFP storage racks can affect the calculated k_{eff}. The licensee performed calculations that determined a maximum Δk_{eff} uncertainty that resulted from the following SFP storage rack mechanical tolerances: box inner dimension, box wall thickness, neutron absorber panel width, wrapper thickness, poison thickness, and minimum Boron-10 (B^{10}) loading. The calculations are performed for different enrichments (2.0 to 5.0 wt% Uranium-235 (U^{235})) at various burnups and with soluble boron concentrations of 0 ppm and 2000 ppm. The uncertainties at each iteration were then statically combined using the square root of the sum of the squares method to derive a combined SFP storage rack manufacturing tolerance Δk_{eff} uncertainty. The maximum fuel assembly manufacturing tolerance Δk_{eff} uncertainty of all the iterations was used at each burnup/enrichment combination to estimate k_{eff}.

The licensee cited ASTM C 992-89, "Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks." Consistent with ASTM C 992, the controlling parameter for neutron attenuation in boron based neutron absorbers is the material's B^{10} areal density. The B^{10} areal density is the B^{10} per unit area of a sheet, which is equivalent to the mass per unit volume of B^{10} in the material multiplied by the thickness of the material in which that isotope is contained. Therefore, the poison thickness, and minimum B^{10} loading are not independent uncertainties and should have been summed before being statistically combined with the other SFP storage racks manufacturing tolerances. This would have increased the maximum combined SFP storage rack manufacturing tolerance Δk_{eff} uncertainty from 0.0037 Δk_{eff} to 0.0048 Δk_{eff}. While this represents a notable increase, after this uncertainty is statistically combined with the square root of the sum of the squares method of all the other uncertainties, the final effect is minimal.

In response to RAI 3 (Reference 18), the licensee added a "METAMIC Measurement Uncertainty" to account for the measurement accuracy associated with the B^{10} content as measured by neutron attenuation during the surveillance program which was statistically combined using the square root of the sum of the squares method with all the other uncertainties.

Treating the poison thickness, minimum B^{10} loading, and the METAMIC surveillance measurement accuracy as uncertainties, rather than biases, essentially does not include any margin for degradation of the inserts.

With the discussion and disposition of the above listed items, the treatment of SFP storage rack mechanical tolerances is consistent with guidance in the Kopp Letter (Reference 21).

3.2.2 SFP Temperature Bias

NRC guidance provided in the Kopp Letter states that the criticality analysis should be performed at the temperature corresponding to the highest reactivity. The licensee calculated pool water temperature effects on reactivity in the MZTR racks with CASMO-4. The calculations were performed for different enrichments (2.0 to 5.0 wt% U^{235}) at various burnups and with soluble boron concentrations of 0 ppm and 2000 ppm. The results show that the SFP temperature coefficient of reactivity is negative, i.e., a higher temperature results in a lower reactivity over the range of credited soluble boron. Consequently, all CASMO-4 calculations are evaluated at 39.2 °F, which corresponds to a moderator density of 1.0 g/cc.

In MCNP4a, the Doppler treatment and cross sections may only be valid at 300 K (80.33 °F). Therefore, a Δk_{eff} is determined in CASMO-4 from 39.2 °F to 80.33 °F, and is included in the final k_{eff} calculation as a bias. The calculations are performed for different enrichments (2.0 to 5.0 wt% U^{235}) at various burnups and with soluble boron concentrations of 0 ppm and 2000 ppm. The maximum temperature bias from the entire burnup and enrichment range was used at each burnup/enrichment combination to estimate k_{eff}.

3.2.3 Fuel Assembly

3.2.3.1 Selection of Bounding Fuel Assembly Design

Comparisons of design parameters for all fuel bundle designs that have been used during operations at BVPS-2 were provided. A description of the analysis and results of the screening to determine the bounding fuel assembly design is provided in Table 4.5.1 of Enclosure B of Reference 10. Based on the total information provided, the NRC staff finds the licensee's use of the bounding design basis fuel assembly acceptable. Note that some of the fuel designs utilized Westinghouse Integral Fuel Burnable Absorbers (IFBA). A separate penalty is calculated to include potential positive reactivity contributions related to use of reactivity control devices such as IFBA, other burnable absorbers, and water displacement rods. The penalty calculation did look at the full range of reactivity control devices used and predicted to be used at BVPS-2, and effectively consider the impacts of reactivity control devices on the bounding fuel assembly design.

3.2.3.2 Fuel Assembly Mechanical Tolerances

Manufacturing tolerances within a fuel assembly can affect the calculated k_{eff}. The licensee performed calculations that determined a maximum Δk_{eff} uncertainty that resulted from the following fuel assembly mechanical tolerances: fuel rod pitch, fuel enrichment, pellet diameter, clad inner and outer diameters, and guide tube inner and outer diameters. The calculations are performed for different enrichments (2.0 to 5.0 wt% U^{235}) at various burnups and with soluble boron concentrations of 0 ppm and 2000 ppm. The uncertainties at each iteration were then statically combined using the root mean sum of the squares method to derive a combined fuel assembly manufacturing tolerance Δk_{eff} uncertainty. The maximum fuel assembly manufacturing tolerance Δk_{eff} uncertainty of all the iterations was used at each burnup/enrichment combination to estimate k_{eff}.

The fuel pellets were modeled as full right circular cylinders, thereby conservatively ignoring the material loss associated with fuel pellet dishing and chamfering. The fuel pellets were modeled with a U^{235} density of 10.6312 g/cc. This is 97% of the maximum theoretical density and is conservative with regard to the BVPS fuel pellets.

The treatment of fuel assembly mechanical tolerances is consistent with guidance in the Kopp Letter (Reference 21).

3.2.4 Spent Fuel Characterization

Characterization of fresh fuel is relatively straight forward. It is based primarily on U^{235} enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis. These tolerances and bounding values carry through to the spent fuel. The standard practice has been to treat the uncertainties as unaffected by the depletion. The characterization of spent fuel is more complex. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: a depletion uncertainty, the axial apportionment of the burnup, and the core operation that achieved that burnup.

To cover the depletion uncertainty, Enclosure B of Reference 10 adopted 5% of the reactivity decrement uncertainty to cover lack of fuel composition calculation validation, consistent with the guidance provided in Reference 21.

At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis has shown that this results in an under prediction of k_{eff}, generally the under prediction becomes larger as burnup increases. This is what is known as the "end effect." Judicious selection of the axial burnup profile is necessary to ensure k_{eff} is not under predicted due to the end effect. The axial apportionment of fuel assembly burnup was addressed using conservatively determined axial burnup distributions for fuel with enriched uranium (2.6 wt% U^{235}) blankets, with natural uranium blankets and with no blankets. These distributions were determined using the end-of-cycle assembly-wise burnup distributions for all BVPS-2 cycles available at the time of the analysis. The distribution derived from these distributions is conservative with respect to the actual distribution. The licensee used the three different axial burnup distributions to determine a loading curve specific to each type of blanket. The licensee has implemented procedures to screen axial burnup distribution for future cycles to ensure that future stored fuel will have axial burnup distributions that are consistent with those used in the analysis. The NRC staff finds that this method of selecting, using, and controlling the axial burnup profile results in a conservative characterization of the end effect. Note, usage of other blanket enrichments is not covered by the analysis.

Burnup credit analyses require simulation of reactor operations to calculate the burned fuel compositions. It is generally required that conservative values be used for core depletion parameters such as moderator temperature/density, fuel temperature, soluble boron concentration, and power density. In the licensee's response to RAI-14 (Reference 8), the results for sensitivity calculations were presented that confirm that the values used for moderator temperature, fuel temperature and soluble boron concentration are conservative. However, results were also provided for the sensitivity of reactivity to power density. From these results, the lower power density is consistently bounding and the value used in the analysis may be non-conservative by less than 100 pcm. Considering that high fuel temperature cannot occur at the same time as low power density, the fuel reactivity is more sensitive to fuel temperature than to power density, and that a nominal power density was used, the margin associated with use of conservative values of soluble boron concentration, fuel temperature, and moderator temperature/density more than compensates for the potential non-conservatism associated with use of the nominal power density.

Fixed and integral burnable poisons harden the neutron spectrum by absorbing thermal neutrons. The hardened neutron spectrum results in an increase in Plutonium-241 production resulting in higher reactivity for an equivalent U^{235} depletion. For this analysis, the criticality calculations involving burned fuel are based on fuel compositions calculated for a bounding fuel assembly design that did not directly include the effects of fixed and integral burnable poisons such as wet annular burnable absorbers (WABA), water displacement rods (WDR), and IFBA. The affect of fixed and integral burnable poisons is instead evaluated separately and included as an additional bias term. The licensee modeled the following combinations; IFBA only, IFBA and WDR, WDR only, and WABA. The licensee performed calculations over different burnups and loading combinations to determine a Δk_{eff} for each iteration. The maximum Δk_{eff} increase due to fixed and integral burnable absorbers was used at each burnup/enrichment combination to estimate k_{eff}.

3.2.5 Criticality Analysis

MCNP4a was used to calculate the k_{eff} values for the criticality analyses. Each rack module was modeled to explicitly reflect the MZTR rack module layout. The three zones can be described as:

- The peripheral cells contain an alternating pattern of fresh (Region 1) and highly burned (Region 2) fuel arranged such that no Region 1 assembly is perpendicularly or diagonally adjacent to any other Region 1 assembly in the same rack module. Region 1 fuel may be fresh fuel and may be enriched up to a nominal 5.0 wt % U^{235}. All Region 2 fuel must meet the minimum burnup requirements specified in Table 4.7.1 of Enclosure B of Reference 10.

- The next row and column in from the peripheral locations is loaded with highly burned (Region 2) fuel, meeting the minimum burnup requirements specified in Table 4.7.1 of Enclosure B of Reference 10.

- All cells inside the first two rows and columns may have a lower burnup (Region 3) fuel assembly, meeting the minimum burnup requirements specified in Table 4.7.2 of Enclosure B of Reference 10.

- As is illustrated in Figure 4.5.7 of Enclosure B of Reference 10, some of the outer two rows or columns of storage cells may be omitted. In these cases, the Region 1 and 2 cells will be omitted and not assigned to cell locations that would otherwise be Region 3 cells.

Note that the polynomial fit provided for Region 3 fuel with natural uranium blankets in Table 4.7.2 is slightly different from the fit provided in the last row of Table 4.7.7.b.

Table 4.7.2 -> $BU = -0.8134x^2 + 18.481x - 33.0634$
Table 4.7.7b -> $BU = -0.8134x^2 + 18.481x - 33.067$

The limit values provided in Table 4.7.2 and the values calculated from the fit and presented in the last row of Table 4.7.7b appear to be from the polynomial fit provided in Table 4.7.2. The Table 4.7.2 fit provides a slightly more restrictive limit than the fit from Table 4.7.7b. The limit values and polynomial fit from Table 4.7.2 are conservative with respect to Table 4.7.7b, consistent with the proposed TS changes (Attachment 1 to Enclosure A of Reference 10) and should be used.

The base analysis documented in Enclosure B of Reference 10 included the impact of reactivity control devices such as WABA, water displacement rods, and IFBAs by calculating a separate penalty and penalty uncertainty using the CASMO-4 program. In response to follow-up question 3 in Attachment 1 to Reference 18, the analysis was revised to use CASMO-4 to calculate fuel compositions for fuel depleted with reactivity control devices and MCNP4a to calculate the change in reactivity associated with the presence of the reactivity control devices. The revised three dimensional MCNP4a calculations yielded a smaller reactivity control device penalty than was obtained using only two-dimensional CASMO-4 calculations.

The normal conditions analysis is adequate. The analysis appropriately calculated k_{eff} values using the most conservative results from calculations modeling either uniform or axially varying burnup profiles. Asymmetric fuel assembly placement and spacing between rack modules appropriately considered. Compliance with the requirements of 10 CFR 50.68(b)(4) was demonstrated. The maximum k_{eff}, including biases and uncertainties, was shown by the licensee's analysis to be no greater than 0.995 at a 95 percent probability, 95 percent confidence level with no soluble boron present. The normal conditions analysis addressed stored fuel, interface conditions between new rack modules, interface conditions between new and old rack modules, and the temporary use of a single stand-alone new rack module in the cask pit. The proposed TS changes include Section 4.3.1.1.e.4, which requires that two empty rows be maintained between adjacent Metamic and Boraflex fuel storage racks.

The analysis included consideration of abnormal SFP water temperatures, SFP boron dilution, and dropped and mislocated assemblies, including assemblies outside, but next to the spent fuel storage rack. An additional unanalyzed miss-loaded fuel assembly configuration was identified during the review. In response to follow-up question 2 in Attachment 1 of Reference 18, the additional configuration was evaluated. The original most reactive abnormal configuration yielded a higher k_{eff} value than the additional configuration. The worst case configuration was a fresh fuel assembly loaded into an outer row position where only Region 2 burned fuel was permitted. The maximum soluble boron concentration required to ensure that k_{eff} is no greater than 0.945 under normal and credible abnormal conditions was calculated for the worst case miss-load to be 1,212 ppm boron by weight, which is well below the 2,000 ppm required by BVPS-2 TS.

3.3 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 the Metamic known to perform effectively as a neutron absorber and Al 6061 is a marine-qualified alloy known for its resistance to corrosion. Metamic has limited operating experience in SFP applications, although it has been approved and installed for use by other licensees.

The licensee proposed a Metamic surveillance program which consists primarily of monitoring the physical properties of the absorber material by performing periodic dimensional and visual checks to confirm the physical properties, and neutron attenuation testing to confirm the neutron absorption capabilities of the Metamic material are being maintained.

3.3.1 Program Description

The purpose of the licensee's Metamic surveillance program is to characterize certain properties of the Metamic to assess the capability of the Metamic panels in the racks to continue to perform their intended function. The program will monitor how the Metamic absorber material properties change over time under the radiation, chemical, and thermal environment found in the SFP.

Metamic coupons will be installed on a coupon tree that holds 8 to 10 coupons. The coupon tree will be placed in the SFP at a location that will ensure a representative dose to the coupons. Coupons will be examined after 2, 4, 8, 10, 15, 20, 25, 30, and 40 years of service life of the new storage racks.

3.3.2 Monitoring Changes in the Physical Properties and Testing of Coupons

The Metamic surveillance program will require a coupon to be removed from the SFP for testing after 2, 4, 8, 10, 15, 20, 25, 30, and 40 years of service. The licensee stated that when a coupon is removed in accordance with the sampling program, the following measurements will be performed:

1. Visual observation and photography:
    a. 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:
    a. 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% B^{10} content, as determined by neutron attenuation, is acceptable
- An increase in thickness at any point should not excess 10% of the initial thickness at that point

Prior to installing the coupons in the SFP, each coupon is pre-characterized. At a minimum, the coupons are pre-characterized for weight, dimensions and B-10 loading which will be used as a baseline when determining if the measurements meet the acceptance criteria. 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). The property changes are confirmed, and then the licensee will perform an engineering evaluation to determine if further testing or corrective action is necessary.

The NRC staff finds that the testing and acceptance criteria are adequate to identify material property changes in the Metamic before significant degradation occurs. After testing is completed, the coupons that have not been destructively analyzed may be returned to the storage pool and remounted on the tree for future evaluation.

3.3.3 NRC Staff Evaluation of the Metamic Surveillance Program

The NRC staff reviewed the licensee's material and surveillance program to confirm compliance with GDC 62 and 10 CFR 50.68(b). Since the licensee credits Metamic in the criticality analysis, the licensee must provide reasonable assurance that the Metamic will be able to perform its intended function for the life of the pool. One acceptable way to ensure that the Metamic will be able to perform its intended function is to monitor the material in the SFP with a surveillance program.

Based on its review of the licensee's Metamic Surveillance program, the NRC staff concludes that the Metamic neutron absorber made from type 6061 aluminum and B_4C particles is compatible with the environment of the SFP. Also, the NRC staff finds the proposed Metamic surveillance program, which includes visual, physical, neutron attenuation and confirmatory tests, is capable of detecting potential degradation of the Metamic material that could impair its neutron absorption capability at a frequency that is acceptable to the NRC staff. Therefore, the NRC staff finds that the use of Metamic as a neutron absorber panel in the new spent fuel racks is acceptable.

3.4 Structural Behavior of the Replacement Rack Structures and the SFP Structure

The NRC staff evaluated the effects of the rerack on the structural behavior of the replacement rack structures and the SFP structure. The NRC staff's technical evaluation of the SFP structure also includes a review of the analyses performed on the cask pit platform structure. Additional consideration is given to any variance in the structural behavior of the BVPS-2 fuel handling building, resulting from the additional weight of the re-racked SFP. The effects of the proposed rerack on these structures are reviewed, in detail, due to the increased loads and altered structural characteristics on the aforementioned SSCs resulting from the proposed rerack. Chapter 3 of the BVPS-2 UFSAR provides the design bases requirements relative to the design of SSCs at BVPS-2. Chapter 9 of the BVPS-2 UFSAR delineates the design bases of the BVPS-2 spent fuel storage SSCs at BVPS-2.

The NRC staff's technical review covered information found in the original LAR (Reference 1), multiple RAI responses, RAI supplements provided by the licensee, and a revised version of the LAR (Reference 25), which updated the information in Reference 1 to include changes to the LAR resulting from RAIs and RAI supplements. The NRC staff's review of the revised LAR also considered the information found in References 19 and 20. Reference 19 includes a summary of the March 9, 2011, conference call held between the NRC staff and the licensee in order for the NRC staff to obtain clarification on a select number of the changes made to the original LAR. Subsequent to the March 9, 2011, conference call, the licensee provided a supplement to the revised LAR which outlined the changes made to the original LAR and also provided additional information on content which was presented in an abbreviated form in the revised LAR (Reference 20).

3.4.1 Replacement Rack Structural Evaluation

The NRC 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 staff's acceptance criteria relative to this review area.

3.4.1.1 Replacement Rack Construction

The general construction of the replacement racks is outlined in Section 2.0 of Enclosure C to Reference 1. The replacement racks are freestanding structures, in that they will not be connected to any other racks or SFP structural components. Additionally, the replacement racks are manufactured as a honeycomb structure consisting of a number of welded, interconnecting cells, each of which holds one spent fuel bundle. The cells are connected to a baseplate at the bottom of the rack structure, which extends beyond the perimeter of the cells. Each rack contains four or five pedestals connected to the baseplate, which rest on bearing pads. In some cases, the bearing pads rest on the existing sub-base beam structure in the SFP. The bearing pads act to distribute the load imparted by the replacement racks on the SFP slab and the sub-base beam structure. The primary material of construction for the racks is SA240-304L stainless steel.

3.4.1.2 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 (Reference 14) and Section IV of the OT Position Paper (Reference 13). While the acceptance criteria within each reference are nearly identical, the licensee indicated that the replacement racks at BVPS-2 were evaluated in a manner which demonstrates compliance with the SRP criteria and the criteria in the OT position paper.

Chapter 3 of the BVPS-2 UFSAR indicates that the SFP, SFP liner, and associated SFP structures, including the current SFP racks, at BVPS-2 were designed against the criteria found in Revision 1 of Section 3.8.4 of the SRP. In response to an NRC staff request for additional information (RAI) regarding the use of SRP Section 3.8.4 in support of the licensee's LAR, the licensee indicated that Revision 2 of SRP Section 3.8.4 had been utilized. However, the licensee indicated that the structural acceptance criteria for the SFP structures remain the same when Revision 1 and Revision 2 of SRP Section 3.8.4 (including the Appendix D portion of SRP Section 3.8.4) are compared. The NRC staff considers the licensee's use of Revision 2 of SRP Section 3.8.4 acceptable, with respect to the proposed LAR at BVPS-2, 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.

Revision 3 of SRP Section 3.8.4 was issued in May 2010. However, Revision 3 of SRP Section 3.8.4 contained only administrative updates to Revision 2 of this SRP section. Therefore, given that the BVPS-2 rerack LAR was submitted prior to the issuance of Revision 3 of SRP Section 3.8.4 and no technical changes were made between the two revisions, the NRC staff considers the acceptance criteria outlined in Revision 2 of SRP Section 3.8.4 adequate, with respect to the review of the BVPS-2 rerack LAR.

As indicated in both references, the construction materials utilized in SFP racks should conform to the provisions within Subsection NF of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, which contains the requirements for the material, design, fabrication, installation, and examination of Class 3 component 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 which must be considered for the applicable service limits are tabulated in each reference. Both references indicate that the applicable stress limits for each loading combination are the stress limits found in Subsection NF of the ASME Code. These 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 3.4.3 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 Code, Section III, in this LAR. 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 Code. In response to an NRC staff RAI, the licensee confirmed in Reference 6 that the design of the replacement racks at BVPS-2 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 Code, Section III, Division 1. The licensee noted in response to RAI 2.c in Reference 6 that the buckling criteria within the ASME Code for linear supports have been relocated to Subsection NF; specifically NF-3321.1(b). This provision of the ASME Code is identical to the buckling requirements which were found in Appendix XVII within previous editions and addenda of the ASME 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 BVPS-2 are designed to be free-standing 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 free-standing 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 reracking, the SRP and OT Position Paper require the evaluation of the differential heating effect between a SFP rack cell which contains spent fuel and one empty SFP rack cell.

3.4.1.3 Analysis Methodology

As indicated in References 13 and 14, 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 licensee employed the use of a proprietary computer program, DYNARACK, in order to perform Whole Pool Multi-Rack (WPMR) time-history analyses of the replacement rack structures which will be utilized at BVPS-2. DYNARACK has been used for similar SFP reracking projects at many operating nuclear facilities. Table 5.4.1 of Enclosure C to Reference 1 provides a partial listing of previous reracking 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 BVPS-2 reracking effort are described in Section 5.4 of Enclosure C to Reference 1. Section 5.4.2 of Enclosure C to Reference 1 provides a general overview of the procedure for the seismic analysis of the BVPS-2 SFP rack structures as it relates to the use of DYNARACK. In response to an NRC staff RAI regarding the application of the BVPS-2 seismic excitation to the DYNARACK simulations, the licensee confirmed that the seismic accelerations along all three orthogonal directions were imposed simultaneously in the time history analyses performed for BVPS-2 (Reference 6).

It is noted that DYNARACK accounts for fluid coupling effects on the structural behavior of the rack structures using classical methods, which have been validated by multiple experiments and comparisons to prior reracking projects. DYNARACK's consideration of impact forces between the rack structures and other structures, including other racks and the SFP structure, are described as being represented by compression-only gap elements. These elements allow DYNARACK to capture the impact forces and displacements between the aforementioned structures. In response to an NRC staff RAI regarding whether DYNARACK is capable of capturing the impacts resulting from racks which are not aligned corner-to-corner, the licensee confirmed in its January 18, 2010, correspondence that DYNARACK does account for every potential impact location between racks utilizing compression-only gap elements at each rack corner.

The licensee provided additional specificity regarding the mathematical model of the rack structure within Section 5.4.2.1 of Enclosure C to Reference 1. It is noted that the rack 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; twelve describing the motion of the rack and ten describing the motion of the fuel assemblies.

The licensee stated in Section 5.5.2 of Enclosure C to Reference 1 and Reference 2 that the synthetic time history accelerations in three orthogonal directions (two horizontal and one vertical) used in the structural evaluation of the SFP replacement racks were generated in compliance with Section 3.7.1 of the SRP and the design basis requirements of BVPS-2. In Reference 5, the NRC staff requested justification from the licensee which would demonstrate that the simulations performed using full racks bounded interim conditions which would exist with partially loaded racks. In its response, the licensee indicated that an additional WPMR simulation for a case of partially loaded racks was performed and determined that the stresses, loads and displacements for the case of fully loaded racks bound those same parameters for a reasonably arranged interim condition in which racks would be partially loaded (Reference 6).

In its January 18, 2010, RAI responses, the licensee provided additional justification for modeling the rack structure as a single-beam, two-node structure based on the natural frequency of the rack. Based on the licensee's response and the number of variables which can influence the dynamic response of the racks within the SFP, the NRC staff issued an additional RAI (Reference 5) in an effort to obtain information allowing for a more specific validation of the DYNARACK mass model properties and the adequacy of this simplified model to predict the anticipated time history seismic responses.

By letter dated May 21, 2010, the licensee's response to RAI-20 indicated that a detailed benchmarking between a rack developed using a finite element analysis (FEA) and a rack modeled in DYNARACK had been performed in order to compare the solutions produced by the lumped mass model within DYNARACK and those produced by the FEA model. The licensee noted that a rack from the Sizewell nuclear plant, located in England, was modeled in the quality assurance-validated FEA code LS-DYNA. The LS-DYNA FEA of the Sizewell rack explicitly modeled all components of the rack and fuel assemblies explicitly. A seismic analysis of the Sizewell rack was performed in air (rather than in water) by simultaneously applying three orthogonal acceleration time histories to the LS-DYNA model. Subsequently, the Sizewell rack was also modeled in DYNARACK (in air), using the lumped mass method, and subjected to the same time histories in an effort to benchmark the DYNARACK simplified mass model. The quantitative comparison between both solutions is outlined in Table RAI 20-1 in Reference 7 and it demonstrates that the DYNARACK results provided more conservative displacements and loads on the rack structure when the same seismic time histories were applied.

Based on the benchmarking of the DYNARACK results to the LS-DYNA results for the Sizewell rack, an extrapolation was carried out in order to demonstrate the ability of the DYNARACK simplified mass model to adequately predict the behavior of the BVPS-2 replacement racks. The licensee noted in the response to RAI-20 that the construction of the Sizewell and BVPS-2 racks are similar; this is articulated by data shown in Table RAI 20-2, which compare the key features of each rack's construction. Furthermore, the licensee performed a series of four simulations in DYNARACK (in water) to compare the behavior of the Sizewell and BVPS-2 rack designs under both site's seismic time histories. This enables an indirect benchmarking of the BVPS-2 rack mass model by virtue of the similarities of the BVPS-2 and Sizewell rack designs and the prior benchmarking of the Sizewell rack against an explicitly modeled rack in LS-DYNA. The results of these four simulations are presented in Table RAI 20-5 and demonstrate favorable results in the sense that the racks from each site behave similarly under the same seismic time histories.

3.4.1.3.1 NRC Staff's Evaluation of the Analysis Methodology

The NRC staff finds the benchmarking of the simplified mass model used in DYNARACK for the time-history analyses at BVPS-2 acceptable because the licensee demonstrated that the dynamic response using the simplified mass model is more conservative than a detailed FEA. This demonstrates the acceptability of the structural properties of the simplified mass model and its ability to adequately predict the anticipated time history seismic responses. However, given that the NRC staff's acceptance of the benchmarking of the simplified mass model used in DYNARACK here within is based solely on design parameters specific to BVPS-2, it should be noted that the NRC staff's acceptance does not constitute a generic acceptance. Benchmarking of the simplified mass model should be performed for site-specific replacement racks and it will continue to be reviewed on a case-by-case basis.
With respect to the fuel loading arrangement utilized within DYNARACK, the NRC staff finds the licensee's justification for evaluating the structural adequacy of the racks using time-history analyses which incorporate fully-loaded racks acceptable. This acceptability is based on the fact that the licensee demonstrated that the structural conditions which exist in this scenario are more limiting than those of an interim condition, as tabulated in Reference 6. Additionally, the synthetic acceleration time histories are in compliance with the March 2007 revision of SRP Section 3.7.1, "Seismic Design Parameters" and the design basis requirements of BVPS-2.

3.4.2 Structural Analysis Results

The results of the analyses performed to demonstrate compliance with the aforementioned acceptance criteria are documented in Section 5.6 of Enclosure C to Reference 1. As indicated above, 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. Additional results which were not provided by the DYNARACK postprocessor are also presented in Section 5.6.

The stress factors used to evaluate the structural adequacy of the rack against the ASME Code criteria are described in Section 5.5.3.1 of Enclosure C to Reference 1. 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 Code provisions is confirmed. As indicated in Section 5.6.5 of Enclosure C to Reference 1, 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 0.888. 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 NRC staff issued one RAI to the licensee regarding the material properties used to develop the stress limits for the rack structural evaluation. In its May 3, 2010, response, the licensee indicated that the material properties used in the analyses were based on a reference temperature of 200 °F. The justification provided by the licensee for the use of this reference temperature was that the

SFP is designed to maintain the bulk pool temperature at or below 170 °F, making 200 °F a conservative reference temperature, given that its use provides for lower material allowable stress limits.

In response to an NRC staff RAI regarding the loading combinations, which were evaluated as part of the structural evaluation of the rack structure, the licensee indicated in its May 3, 2010, correspondence that the thermal stresses were not combined with the primary stresses in the structural analyses described above. The licensee's justification was based on a provision within Subsection NF of the ASME Code, which indicates that self-equilibrating stresses, such as thermal stresses, need not be considered in the design of structures under Subsection NF. However, the NRC staff requested supplemental information regarding this position, given that the SRP and OT Position Paper include the combination of thermal and primary stresses, when comparing these loading combinations against the provisions in Subsection NF of the ASME Code, Section III.

By letter dated January 5, 2011, the licensee provided supplemental information regarding its previous RAI response which detailed a thermal analysis to evaluate the stresses on the rack structure utilizing the finite element method. The FEA performed by the licensee represented a full rack which was partially loaded with fuel that had been recently offloaded from the reactor core in order to maximize the temperature gradient. The thermal stress contours presented in Figure 6-4 of the supplemental response provided quantitative results for the stresses resulting from the abnormal temperature distribution. To demonstrate compliance with the stress criteria specified in the SRP and OT Position Paper, the thermal stresses were combined with the limiting primary stresses (developed in DYNARACK and described above) and demonstrated that these loading combinations did not exceed the prescribed stress limits for the given service conditions. The thermal stress FEA described in Reference 12 was also utilized to provide additional detail on the cell-to-cell welds, which are discussed below.

Section 5.6.7 of Enclosure C to Reference 1 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 5.6.7 of Enclosure C to Reference 1, the limiting baseplate-to-rack cell welds, baseplate-to-pedestal welds, and cell-to-cell welds were evaluated against the aforementioned ASME Code provisions for the applicable service limits. The results demonstrate that each of the limiting weld locations have margins which are sufficient and acceptable, when compared to the ASME Code stress limits. In response to an NRC staff RAI regarding the methodology used to determine the stresses in the baseplate-to-rack cell welds, the licensee provided additional information regarding the conversion of stress ratios for base metal components to stress ratios used to structurally qualify the weld components. The NRC staff's RAI inquired on the licensee's use of area ratios to determine both flexural and axial stresses developed in the weld. The response provided a quantitative comparison demonstrating that the use of area ratios to convert stress ratios for these welds is nearly equivalent to the method utilizing moments of inertia (Reference 6).

3.4.2.1 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 5.6.10.2 of Enclosure C to Reference 1. 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 6.0 of Enclosure C to Reference 1 revealed that the limiting cell temperature differs from the bulk pool temperature by a maximum value of 40 °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 Code provision.

The licensee supplemented its discussion regarding the differential heating effects on the weld shear stresses by letters dated August 9, 2010, and January 5, 2011. The licensee's August 9, 2010, response demonstrated that the maximum shear stresses in the cell-to-cell welds, due to thermal and seismic loading, were within the stress limits of the applicable ASME Code provisions. The licensee's January 5, 2011, response included an FEA to evaluate the thermal stresses induced within the rack resulting from an abnormal temperature distribution. The results of this FEA confirmed the conclusions reached in the August 9, 2010, submittal, which demonstrated compliance with the applicable design code provisions.

3.4.2.2 Rack Displacements

With respect to the evaluation of postulated overturning of a rack structure, the licensee reported that the maximum displacement of the top of any rack in the BVPS-2 DYNARACK simulations was 2.79 inches (Section 5.6.1 of Enclosure C to Reference 1). Comparing this displacement value with a minimum rack width of 81 inches for the replacement rack structures, the licensee concluded that overturning was of no concern and the acceptance criteria specified in the March 1981 revision of SRP Section 3.8.5, "Foundations" were satisfied.

With respect to evaluating the potential impact loadings resulting from the DYNARACK simulations, the structural integrity of the racks was assessed based on the impact force due to a fuel assembly within a cell wall and the loads imposed on the racks resulting from rack-to-rack impacts. For the former case, the licensee presented the limiting impact load resulting from a fuel-to-cell wall impact, based on the time history accelerations applied to the SFP rack system in DYNARACK. The DYNARACK results yielded a maximum fuel-to-cell wall impact load of 610.2 pounds (lbs), compared to an allowable value of 3,204 lbs (Section 5.6.4.2 of Enclosure C to Reference 1).

3.4.2.3 Buckling Evaluation

The evaluation of the rack-to-rack impacts resulting from the DYNARACK simulations was presented in Section 5.6.4.1 of Enclosure C to Reference 1. The licensee noted that the placement of the 0.75 inch-thick baseplates are strategically oriented such that the baseplates extend well beyond the rack perimeter. Additionally, the spacing between adjacent baseplates is minimized. As such, the arrangement works to ensure that impact between the racks is minimal and concentrated at the baseplates during a seismic event. However, the licensee noted that there are instances where the tops of the racks do impact one another during the DYNARACK simulations. Subsequently, a buckling analysis was performed to demonstrate the structural adequacy of the impacted region at the top of the rack. The licensee also stated that no rack-to-wall impacts occur in any of the dynamic simulations.

In response to an NRC staff RAI regarding this buckling analysis, the licensee indicated in its January 18, 2010, response that the original buckling analysis had been re-performed to revise the boundary conditions present in the analysis. Based on the results of the revised analysis, a 0.25 inch-thick, 10 inch-deep, reinforcement bar was added to the top of each rack. As stated in the response, the revised buckling analysis utilized the LS-DYNA FEA program to determine the critical buckling value at the impact location on the top of the rack. The LS-DYNA results indicated that the critical buckling load due to impact at this location would be 153,500 lbs (153.5 kips); this LS-DYNA analysis included the addition of the reinforcement bar. By comparing this critical buckling value with the limiting impact load of 101.8 kips developed in DYNARACK, it was confirmed that the ASME Code buckling criteria, which requires a buckling load of no more than two-thirds the value of the critical buckling value, was satisfied.

An additional buckling analysis for the limiting cell wall location, located at the rack base junction, was also performed as part of the structural evaluation of the replacement racks. As described in Section 5.6.10.1 of Enclosure C to Reference 1, the licensee stated that classical methods were employed to perform the buckling analysis of the outermost cell. The results from the DYNARACK time history analyses were utilized to compare the maximum compressive stress developed on the cell wall to the critical buckling stress determined using classical plate buckling analysis. The results demonstrate sufficient and acceptable margin between the compressive stresses in the cell wall and the critical buckling stress to satisfy ASME Code buckling criteria.

3.4.2.4 Fatigue Evaluation

The licensee acknowledged in Reference 20 that inspection of the pedestal-to-rack baseplate junctions is not feasible due to their location underneath the rack structure. The inaccessible nature of these locations is exacerbated by the fact that the rack baseplates are essentially abutted against neighboring racks, coupled with the fact that there are spent fuel bundles within the racks. The licensee performed a fatigue evaluation to demonstrate that inaccessible locations of the rack structure, which exhibit limiting stress intensities under seismic loading conditions, maintain sufficient margin against fatigue failure. The licensee utilized the results of the BVPS-2 time history analyses to develop a cumulative damage factor based on an assumed number of seismically-induced loading cycles, consisting of one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE), in accordance with the BVPS-2 design basis requirements. The licensee demonstrated that the cumulative damage factor of the limiting stress locations at the bottom of the rack would remain below the limit of 1.0, satisfying ASME Code, Section III, Subsection NB acceptance criterion. As such, the fatigue failure of the critical stress locations at the bottom of the rack should not be expected.

While fatigue evaluations were performed for locations on the rack which are inaccessible for post-earthquake inspection, the licensee indicated in Reference 20 that the BVPS-2 post-earthquake operating procedure will be revised to require inspections of the accessible portions of the rack structures. The updated procedure will require the post-earthquake inspection of rack-to-rack and rack-to-wall gap measurements to verify that minimum gap spacing exists in accordance with the BVPS-2 design basis requirements. Additionally, the updated procedure will require post-earthquake visual inspections of accessible areas of the rack structures to account for signs of physical damage, based on which appropriate corrective actions would be taken. The licensee indicated that these inspections would be carried out in accordance with

RG 1.167, "Restart of a Nuclear Plant Shut Down by a Seismic Event," issued in March 1997. This RG endorses the use of the Electric Power Research Institute (EPRI) Report NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," with conditions, which was issued in December 1989.

3.4.2.5 NRC Staff's Evaluation of the Replacement Rack Structures

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 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 and the OT Position Paper. The NRC staff considers the stress limits used by the licensee, which are based on a material reference temperature of 200 °F, acceptable, given that the SFP bulk temperature is expected to remain below this temperature and the use of a higher temperature results in more limiting material properties, thus lowering the stress limits.

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 NRC staff finds the licensee's buckling analyses performed to demonstrate acceptable margin against buckling at the top of the rack, due to rack-to-rack impact, and 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 Code requirements related to buckling are satisfied for both scenarios. The NRC 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 NRC staff finds the fatigue analysis of the rack structure to determine safety margin of the pedestal-to-rack baseplate junction against fatigue failure acceptable because (1) the number of earthquake cycles used in the fatigue analysis is consistent with the BVPS-2 UFSAR, Section 3.7B.3.2, which requires the assumption of five OBE and one SSE and a minimum of ten maximum stress cycles per earthquake, when the structural analysis considers cyclical seismic loads; and (2) the licensee determined the cumulative damage factor of the limiting stress locations at the bottom of the rack to be 0.615 (Reference 1), which is below the limit of 1.0, satisfying the ASME Code, Section III, Subsection NB, acceptance criterion. The NRC staff also considers the licensee's use of RG 1.167 and, correspondingly, EPRI NP-6695, acceptable for use in performing post-earthquake inspections of the racks for identifying physical damage and verifying inter-rack and rack-to-wall gap measurements.

Based on the review outlined above, the NRC staff has determined that reasonable assurance has been provided which demonstrates that the replacement rack structures at BVPS-2 will maintain adequate structural margin under operating and abnormal loading conditions.

3.4.3 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 FHAs evaluated in support of the proposed BVPS-2 rerack include a straight drop on the top of a rack (i.e., the shallow drop accident), a straight drop through an individual cell all the way to the bottom of the rack (i.e., the deep drop accident), 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 racks due to the postulated drop of a fuel bundle onto another fuel bundle stored in a rack, a postulated drop of the gate, separating the three areas of the BVPS-2 SFP, on the rack, and the postulated effects on the rack due to the uplift force resulting from a stuck fuel assembly. The licensee also performed an assessment of the structural behavior of the SFP structure following a postulated drop of one replacement rack onto the floor of the SFP.

In Reference 25, the licensee also evaluated two additional accidents, which postulated the drop of a rack and the drop of the cask pit platform in the cask pit area to demonstrate that the cask pit area would maintain adequate structural integrity following these accidents. However, as indicated in Reference 19, the licensee confirmed during the March 9, 2011, clarification conference call that rerack activities at BVPS-2, which involved modifications and work in the cask pit area, would be performed under the provisions of 10 CFR 50.59, "Changes, tests and experiments." As such, these accidents were not associated with the revised LAR and no prior NRC approval for plant modifications related to these accidents is required prior to completing work in the cask pit area.

3.4.3.1 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 require 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 3.5.3 of the BVPS-2 UFSAR.

3.4.3.2 Shallow Drop Accident

Section 7.2 of Enclosure C to Reference 1 provides the details of the accident analyses performed in support of the proposed LAR, including the shallow drop accident discussed here within. Reference 1 includes information detailing the mathematical model used to demonstrate the structural integrity of the replacement rack structure under postulated accident conditions, the pertinent parameters used in the accident simulations (Table 7.4.1 of Enclosure C), and the results of the accident analyses performed in support of the LAR (the original analysis). The shallow drop analysis was revised twice based on NRC staff RAIs and discussions facilitated by the September 27, 2010, public meeting between the NRC staff and the licensee. The results of these two revisions are documented in References 9 and 12.

3.4.3.2.1 Shallow Drop Accident Analysis Methodology

The analyses performed for the shallow drop accident scenario at BVPS-2 are based on computer simulations using the LS-DYNA FEA computer code. Previous SFP re-rack LARs have made extensive use of the LS-DYNA computer code for mechanical accident 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 accident analysis is included in Table 7.4.1 of the original analysis. These critical simulation parameters include the impact weight (i.e., the combined weight of one BVPS-2 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. In its January 18, 2010, response to an NRC staff RAI which requested justification for assuming a 24-inch drop height, the licensee indicated that physical limitations prevent the fuel from being held any higher than 24 inches above the racks, thus making this the limiting drop height. The original 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.

In response to an NRC staff RAI requesting the licensee to provide more information regarding the stress-strain curve used to represent the material behavior of the base metal (SA240-304L stainless steel) in the LS-DYNA simulation, the licensee indicated in Reference 3 that a true stress-strain curve with a failure strain value of 1.204 in/in was used in the original LS-DYNA shallow drop accident simulation. By letter dated August 9, 2010 (Reference 9), the licensee submitted the results of a second shallow drop accident analysis. The second shallow drop accident analysis incorporated the following: (1) 98% exceedance probability (EP) failure strains found in Table B.1 of NUREG-1864 (Reference 15) for both base metal and weld materials; (2) a triaxiality factor (TF) of 0.6065, based on the Hancock-Mackenzie Model; and (3) a strain rate amplification factor, based on information in Reference 16, for both base metal and weld metal material models.

A public meeting was held on September 27, 2010, to discuss the licensee's August 9, 2010, supplemental RAI responses, including the supplemental response to RAI 17. During the discussions held between the licensee and the NRC staff regarding RAI 17 at the public meeting, the NRC staff informed the licensee that the use of TF of 0.6065, based on the Hancock-Mackenzie model, is possibly non-conservative. Additionally, the NRC staff informed the licensee that additional justification would be necessary regarding the application of the strain rate effects cited in Reference 16, to the weld material. The NRC staff also requested that the licensee provide a detailed synopsis of how the TF and strain rate effects were implemented into the LS-DYNA simulation.

By letter dated January 5, 2011 (Reference 12), the licensee submitted supplemental information regarding the shallow drop accident analysis. The licensee performed two additional iterations of the shallow drop accident analysis in order to demonstrate the ability of the replacement rack structure to withstand the impact associated with the shallow drop accident (the third and fourth analyses). Two analyses were performed based on a revised geometrical model of the rack structure. The revised geometry included the 10 inch-wide band of 0.25-inch thick SA240-304L stainless steel, which was welded around the top perimeter of the rack structure. As such, the third analysis included this new structural feature, while still modeling the impact of the fuel bundle on the peripheral edge of the rack. The fourth analysis modeled the fuel bundle impacting an interior panel, as opposed to the outer panel, given that the outer panel could no longer be assumed as the limiting impact location due to the added reinforcement.

The licensee confirmed that the strain rate amplification factors were only applied to the stress values found on the stress-strain curve, i.e., the true strain at failure was not affected by this amplification, however the area under the curve was increased. In crediting the effects of strain rate on the material behavior, it was stated in Reference 12 that an iterative procedure was utilized in applying the strain rate effect, whereby the instantaneous strain rate was calculated by LS-DYNA during the analyses and the corresponding amplification factor (based on Figure 17-1 of Reference 9) was applied. For the third and fourth analyses, it was also stated that strain rate effects were not applied to the weld material. The licensee also stated that a revised TF of 0.5 had been used, based on the bi-axial state of stress present in the cell wall material, to determine the failure strain limits for the base metal material and the weld material, which are used in LS-DYNA as constant value properties.

3.4.3.2.2 Shallow Drop Accident Evaluation Results

The licensee summarized the results of the evaluations performed to demonstrate the replacement rack's ability to withstand the impact loads generated due to a postulated shallow drop accident in Table 17-2 (fuel assembly drop onto perimeter cell) and Table 17-3 (fuel assembly drop onto interior cell) of Reference 12. The licensee stated that (1) the LS-DYNA analysis results demonstrate that the plastic deformation in the rack cell walls resulting from a shallow drop accident (onto a perimeter cell or an interior cell) does not extend down into the neutron absorber zone, which is defined as the vertical length of the cell blanketed by the fixed neutron absorber panel; (2) for the BVPS-2 spent fuel racks, the minimum distance from the top of the rack to the top edge of the neutron absorber panel (the neutron absorber zone) is 19.75 inches; and (3) from the LS-DYNA simulations, the dropped fuel assembly moves downward crushing the impacted cell wall to a maximum depth of 3.15 inches (interior cell drop), and the plastic strain in the impacted cell wall diminishes to zero at a distance of 12 inches (perimeter cell drop) below the top of the rack. As such, the licensee concluded that since the depth of damage by either measure is less than 19.75 inches, the neutron absorber panels do not suffer any damage and, therefore, the shallow drop accident has no adverse effect on the criticality safety analysis for the BVPS-2 spent fuel racks.

3.4.3.2.3 NRC Staff's Evaluation of the Shallow Drop Accident

The licensee utilized 98% EP strains found in Table B.1 of NUREG-1864 for both base metal and weld materials to perform the shallow drop accident analyses. The NRC staff finds the use of the 98% EP strain acceptable based on the fact that the use of this value provides a statistically significant assurance regarding the reliability of the material properties. Additionally, the NRC staff considers the use of a TF a necessary requirement when incorporating a true stress-strain curve into an analysis which models nonlinear material behavior. This necessity is due to the fact that true failure strains are derived from a one-dimensional state of stress, which contrasts with the reality that a two or three-dimensional state of stress may exist in an analysis which prohibits plastic flow within the material, effectively lowering the failure strain.

The use of many of the factors associated with true stress-strain curves is discussed in Appendix B to NUREG-1864. While the primary focus of NUREG-1864 is dry cask storage systems, the NRC staff finds the information presented in Appendix B, regarding the construction of true stress strain curves to model nonlinear material behavior, applicable to the BVPS-2 rerack LAR due to the fact that the information is being applied in order to model the rack material behavior, without a significant consideration for the structure. With respect to TFs, the formulation of these factors in Appendix B of NUREG-1864 results in a TF of 0.5 for a state of biaxial tension. The NRC staff has also previously documented its acceptance of a TF of 0.5 for cases of biaxial tension in the SE Report developed for Amendment 7 to the Certificate of Compliance for the Holtec International HI-STORM 100 cask system (Reference 17). Therefore, the NRC staff finds the use of a TF of 0.5 for the BVPS-2 shallow drop accident analysis acceptable based on the conformance with NRC staff-approved methods for calculating this value for a state of biaxial stress.

The NRC staff also evaluated the merits of the licensee's use of strain rate amplification factors. The NRC staff considers the use of the strain rate amplification factors in the shallow drop accident analyses acceptable based on the fact that (1) the data on which these factors are based, were derived from a significant amount of work performed by Idaho National Laboratory (INL) (Reference 16); (2) as documented in Reference 16, numerous tests were performed at INL on SA240-304L stainless steel (the same material of which replacement racks are constructed) at varying temperatures and strain rates to develop corresponding factors which effectively increase the failure strain of this material at certain strain rates; (3) an iterative procedure was utilized in applying the strain rate effect which allowed for a "real time" application of the amplification curve at the correct strain rates; and (4) the licensee only applied the strain rate amplification factors to the base material, consistent with the information available in Reference 16, and no strain rate amplification factor was used for the weld material.

The NRC staff also finds the analysis acceptable based on the fact that the assessment evaluates two bounding impact locations. By modeling the impact of the fuel bundle on an inner and outer cell, the licensee was able to demonstrate that the deformation resulting from both cases remains within the acceptance criteria limits.

Based on the analytical assessments performed to demonstrate the functional integrity of the BVPS-2 replacement racks subject to shallow drop impact loads, the NRC staff's review concludes that the licensee's assessment is acceptable, based on the results of the analytical evaluations, which yielded plastic deformation in the rack cell walls that does not extend down into the neutron absorber zone, with an adequate margin of safety (i.e., calculated maximum plastic deformation of 12" versus available distance of 19.75" to the neutron absorber zone). As indicated above, this acceptance is based on the satisfaction of the NRC staff's criteria associated with the use of a true stress-strain curve using 98% EP strains for both base and weld materials, including the use of minimum material properties and a TF. Furthermore, given the sensitivities inherent in the construction of true stress-strain curves used in these analyses and the inherent uncertainties associated with the modeling and analysis of the replacement racks, the NRC staff's acceptance of the use of the methodologies described above is not a generic acceptance and will continue to be reviewed on a case-by-case basis.

3.4.4 Deep Drop Accident

The description of the mechanical accidents considered in the evaluation of the functional integrity of the replacement SFP racks proposed for use at BVPS-2 is included in Section 7.2 of Enclosure C to Reference 1. The licensee notes that two types of deep drop accidents were evaluated; one in which a fuel bundle falls completely through an empty rack cell, and impacts the base plate at a location which is not supported underneath by a rack pedestal, the second being a similarly modeled accident in which the bundle is simulated to fall completely through an empty SFP rack cell to a location on the base plate directly above a rack pedestal. The most limiting scenario between these two deep drop accident variations is the former case, in which the fuel bundle impacts the base plate of the replacement rack structure, 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 of Enclosure C in Reference 1. The licensee stated in Reference 25 that the deep drop scenario two, through an empty cell over a support pedestal, is bounded by the postulated rack drop event since a rack drop has more impact energy than a dropped fuel assembly.

In performing the analyses, the licensee employed LS-DYNA to evaluate the deep drop accident 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 accident whereby the fuel bundle impacted the rack at the top edges. The results of the most limiting deep drop accident scenario, with respect to the functional integrity of the rack, are summarized in Section 7.5.2 of Enclosure C to Reference 1 and presented graphically in Figure 7.5.2 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 baseplate deformation due to the deep drop accident postulation is considered in the criticality analyses. The licensee confirmed in Reference 20 that the material properties utilized for the shallow drop accident were also incorporated in the deep drop accident, which was presented in the revised LAR (Reference 25). However, the licensee stated that utilizing the revised material properties resulted in the same base plate deformation as the original analysis performed for the deep drop scenario, which utilized engineering stress-strain curves.

3.4.4.1 NRC Staff's Evaluation of the Deep Drop Accident

The NRC staff finds the deep drop accident assessment performed by the licensee in support of the rerack of the BVPS-2 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 accident. This deformation remained constant when the licensee incorporated the use of the true stress-strain material model in the revised deep drop accident analysis. The licensee was able to demonstrate that the deep drop accident 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.

The licensee concluded that the deep drop accident, which models a fuel bundle impacting the base plate of the rack at a location supported by a pedestal directly beneath the impact location, is bounded by the evaluation performed for the rack drop accident. The NRC staff finds this acceptable based on the impact data provided in Table 7.4.1 of Enclosure C to Reference 1. This data demonstrates that the energy imparted by the rack pedestals on the SFP structure, including the SFP liner, would be greater in the rack drop event than the imparted energy resulting from a falling fuel bundle, which would also transmit its impact energy through the pedestal. In this respect, the accidents are the same, given that they both impart energy to the SFP and liner through the pedestal, with the rack drop accident being more severe, and thus, bounding.

3.4.5 Inclined Drop Accident

The OT Position Paper requires the consideration of a FHA whereby an inclined fuel bundle is postulated to impact an SFP rack. Following this impact, the functional integrity of the SFP rack must be demonstrated. The LAR submitted in support of the rerack of the BVPS-2 SFP did not include information regarding the postulation of an impact of an inclined fuel assembly on the replacement racks at BVPS-2. In response to an NRC staff RAI regarding the justification for not evaluating the inclined fuel assembly accident, the licensee indicated that this accident was not explicitly evaluated and was bounded by the results performed for the shallow drop accident evaluation. Included in the licensee's justification were the following assumptions regarding the rationale for the lack of an explicit inclined impact evaluation: (1) the physical handling of the fuel assembly maintains the fuel assembly in an upright manner, minimizing the chance of an inclined orientation if the fuel assembly was to separate from the fuel handling crane; (2) the quarter-symmetry of the fuel bundle would prevent the bundle from rotating out of an upright orientation in the event of a drop of the bundle; and, (3) based on administrative procedures, the fuel bundles are not moved more than 24 inches above the height of the SFP racks, thus minimizing the height available for the fuel bundle to achieve an orientation other than vertical during a postulated drop.

3.4.5.1 NRC Staff's Evaluation of the Inclined Drop Accident

The NRC staff finds the licensee's justification for not performing an explicit evaluation of an inclined fuel assembly impact acceptable based on the following discussion. The licensee has demonstrated that administrative procedures at BVPS-2 limit the possibility that a fuel assembly could achieve any significant inclined motion, if the assembly was to separate from the fuel handling crane. The NRC staff finds this reasonable given that the small amount of vertical space between the top of the SFP racks and the bottom of the fuel assemblies (24 inches) is likely not enough space for the assembly to achieve a significant horizontal component of impact. This rationale is reinforced by the licensee's demonstration that the quarter-symmetry of the assembly will maintain the bundle upright due to the symmetry of the weight and drag forces imposed on the bundle in water. The shallow drop accident maximizes the energy imparted to the top of the rack by assuming that the bundle remains vertical throughout the shallow drop accident; the NRC staff considers this acceptable. It should be noted that the NRC staff's acceptance of the above reasoning is specific to BVPS-2 and, therefore, is not a generic acceptance and the necessity of performing an inclined drop event analysis, as specified in the OT position paper, will continue to be reviewed on a case-by-case basis.

3.4.6 Rack Drop Accident

The licensee performed an assessment of a postulated accident whereby an empty replacement rack structure, resembling the heaviest version of the BVPS-2 replacement racks, is dropped from a height equal to the top of the water level in the SFP in order to evaluate the ability of the SFP structure, including the liner, to satisfactorily absorb the impact imparted by the rack pedestals. This evaluation is described in Section 7.2 of Enclosure C to Reference 1 with the pertinent evaluation parameters tabulated in Table 7.4.1 of the enclosure. The original analysis presented in Enclosure C to Reference 1 utilized LS-DYNA to perform an analysis on a quarter-symmetry model of the rack drop event. The results of the postulated rack drop accident analyses were presented in Section 7.5.3, whereby the licensee concluded that the SFP liner strain did not exceed the failure strain of the liner stainless steel material and the SFP concrete experienced only local damage.

While the original analysis explicitly modeled the SFP concrete, liner, and rack components in a manner similar to the analyses described above for other postulated accidents, the NRC staff issued an RAI by letter dated March 14, 2010, which requested justification from the licensee for not considering the global behavior of the SFP slab, given that only a small representation of the SFP slab was included in the quarter-symmetric model utilized in the FEA of the accident. Based on the FEA model geometry, the analysis yielded the results for only a localized portion of the SFP.

By letters dated March 3, 2010, August 9, 2010, and January 5, 2011, the licensee provided supplemental responses regarding the rack drop analyses. The January 5, 2011, supplemental response included additional information which provided confirmation that the dead load used in the SFP slab analysis included the weight due to the SFP water, the weight of the fully loaded replacement racks, and the weight of the reinforced SFP slab.

In the January 5, 2011, supplemental response, the licensee also included the details regarding the revised evaluation performed for the rack drop accident analysis. This evaluation was performed in order to provide a more explicit quantification of the local and global behavior of the SFP due to the rack drop accident. Utilizing LS-DYNA, the licensee developed a quarter-symmetry model of the BVPS-2 SFP, which included explicit representations of the slab concrete, the reinforcement within the slab, and the liner; the model is illustrated in Figure 5-1 of Reference 12. The licensee performed impact analyses utilizing this LS-DYNA model to capture the SFP slab structural behavior due to the rack drop. The LS-DYNA impact analysis also included the effects of the self-weight of the reinforced concrete, the weight of the SFP water, and the weight of the fully loaded SFP replacement racks to take into account the strain energy used by the static loading.

The results of the LS-DYNA analysis simulating the rack drop event are summarized in tabular form and by four contour plots representing the SFP liner strain, the stresses induced in the top and bottom slab reinforcement layers, and the SFP compressive concrete stresses. For the bottom reinforcement layer, the resulting stress outputs from the LS-DYNA analysis of the rack drop accident were combined with the tensile stresses induced in the bottom reinforcement to account for the through-thickness temperature gradient effects which were not explicitly considered in the LS-DYNA analysis. The resulting combined stress of 16,659 pound per inch squared (psi) is well below the yield value of 40,000 psi for the steel reinforcement. The results also show that stresses induced in the top layer of reinforcement are also below the yield strength of the steel reinforcement. The licensee included information pertaining to the BVPS-2 ductility ratio requirements related to reinforced concrete missile barriers at the plant; these ductility ratio requirements are located in Section 3.5.3 of the BVPS-2 UFSAR. Furthermore, the licensee confirmed that these requirements were met by indicating that the yield point was not reached during the rack drop event, thus satisfying the ductility ratio requirements applicable to BVPS-2.

In quantifying the local effects due to the loads imposed on the SFP concrete and liner by the rack pedestal, the licensee noted that the maximum plastic strain of the SFP liner was 0.0163 in/in, which is well below the failure strain value of 0.362 in/in, which was derived in support of the shallow drop accident analysis for 304L stainless steel. With respect to the compressive stresses induced in the SFP concrete, the licensee indicated that the results show that local crushing of the concrete does occur at the location of the pedestal impact. However, given that the liner strain is well below the failure strain, the licensee concluded that there is no SFP leakage postulated due to the rack drop accident.

Given that the structural capacity of the SFP slab, which is essential in performing the evaluation described above, is dependent on the concrete compressive strength, the NRC staff requested the licensee to confirm that the compressive strength value used in the SFP structural evaluations was in accordance with the BVPS-2 design bases. As such, the licensee's January 5, 2011, supplemental response to RAI-19 confirmed that the design basis compressive strength value of 3,000 psi was utilized in all of the evaluations performed in support of the rerack, including the evaluation of the rack drop accident, as detailed above.

3.4.6.1 <u>NRC Staff's Evaluation of the Rack Drop Accident</u>

The NRC staff finds the licensee's assessment of the rack drop accident acceptable based on the following discussion. The licensee was able to demonstrate that the loads imposed on the SFP liner and SFP reinforced concrete slab, by the postulated drop of a BVPS-2 replacement rack, did not result in unacceptable performance of either the liner or slab. The NRC staff finds this analysis acceptable based on the fact that the licensee included all of the pertinent loads, including the dead loads due to the rack and water weights and the applicable thermal loads, while imposing the impact load on these structures using a replacement rack size which bounds the rack sizes being utilized in the rerack of the BVPS-2 SFP. The licensee's acceptance criteria related to the liner strain have been previously discussed, and accepted by the NRC staff, for use in the shallow drop analysis. The NRC staff finds the licensee's use of its design basis value of 3,000 psi for concrete compressive strength acceptable, given that it is consistent with the BVPS-2 design basis requirements. By meeting the aforementioned material acceptance criteria, the licensee satisfied the overall NRC staff acceptance criteria found in the OT Position Paper and the SRP, which require the SFP and the SFP liner, to maintain functional integrity following a postulated accident.

3.4.7 <u>Uplift Force Evaluation (Stuck Fuel Assembly Accident)</u>

Section 7.2 of Enclosure C to Reference 1 includes a description of the accident analysis performed to demonstrate the satisfactory structural response of the BVPS-2 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 7.5.4 of Encosure C to Reference 1 indicate that the maximum stress induced in a replacement rack cell, resulting from a 5,000 lbs uplift force, is 6,500 psi (6.5 ksi). Based on a yield stress value of 28 ksi for the SA240-304L material presented in Table 7.4.2 of Enclosure C to Reference 1, the licensee concluded that the uplift force-induced stress of 6.5 ksi meets the acceptance criteria for the loads induced on the replacement racks due to FHAs, given that there is no plastic deformation which could result in a loss of functional capabilities.

In response to an NRC staff RAI regarding the relationship between the capacity of the motor-driven platform crane, which is used to handle spent fuel at BVPS-2, and the uplift force used in the accident evaluation, the licensee indicated in Reference 6 that administrative controls exist such that the uplift load induced by the crane does not approach the crane capacity. Specifically, when the platform crane is utilized for moving loads over fuel assemblies in the SFP, the two crane hoist load cells are set to trip when the load on each exceeds 2,000 lbs; this period of limited operation is applicable during the movement of fuel assemblies themselves. Therefore, total loads above 4,000 lbs would not be permitted during the movement of spent fuel assemblies at BVPS-2, making the 5,000 lbs uplift force a bounding evaluation parameter.

3.4.7.1 NRC Staff's Evaluation of the Stuck Fuel Assembly Accident

The NRC staff finds the licensee's assessment of the replacement rack's structural integrity during a postulated stuck fuel assembly accident acceptable, based on the following discussion. The licensee utilized a bounding value for the uplift force which could be expected due to a stuck fuel assembly. The NRC staff finds this as an acceptable and bounding value, based on the fact that administrative controls at BVPS-2 prevent the platform crane from inducing a load of more than 4,000 lbs when fuel assemblies are traversing load paths over the SFP. As such, the evaluation performed by the licensee using a value of 5,000 lbs uplift force bounds the conditions which could be expected at BVPS-2. 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 NRC 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.

3.4.8 Gate Drop Accident

In addition to evaluating the effects of a postulated shallow drop of a fuel bundle on the BVPS-2 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 Enclosure C to Reference 1, postulates a drop of the transfer canal gate onto the top of the replacement rack structure. This gate, located at the fuel transfer canal, acts as a barrier between the reactor containment building and the SFP in the fuel handling building. As with the other accidents considered in support of the evaluation of the BVPS-2 replacement racks, the pertinent evaluation parameters for this accident are located in Table 7.4.1 of Enclosure C to Reference 1.

The results of the gate drop event evaluation are presented in Section 7.5.1 of Enclosure C to Reference 1. In response to an NRC staff RAI regarding the postulated gate drop accident, the licensee indicated in Reference 3 that the transfer canal gate is classified as a "heavy load." In accordance with administrative procedures in place at BVPS-2, the movement of heavy loads above spent fuel racks is prohibited. As such, the licensee indicated that the event was conservatively analyzed in support of the proposed re-rack, and that an analysis was not required.

3.4.8.1 NRC Staff's Evaluation of the Gate Drop Accident

The NRC staff finds the licensee's assessment of the gate drop accident acceptable based on the administrative controls present at BVPS-2, which prevent the movement of heavy loads over spent fuel. Given that the BVPS-2 transfer canal gate is classified as a heavy load, the administrative procedures preclude the potential for the postulated accident involving the drop of the transfer canal gate on to the BVPS-2 replacement racks.

3.4.9 Fuel-to-Fuel Drop Accident

The licensee performed a new postulated mechanical fuel drop analysis in the fuel building described as the Fuel-to-Fuel Drop event. The Fuel-to-Fuel Drop event assumes 10 rods fail in the target assembly due to dropping the assembly on top of another stored in the fuel building storage racks. The drop height is 24 inches above the top of the rack. No fuel rods are predicted to fail in the dropped assembly. According to its May 3, 2010, letter, the licensee stated that a minimum coverage of 23 feet (ft) of water exists above the damaged fuel during the postulated accident. Based on the depth of water above the damaged fuel being 23 ft or greater, the decontamination factors (DF) for the elemental and organic iodine species are 500 and 1, respectively. This results in an overall effective DF of 200, which ensures adequate removal of iodine from airborne releases of activity. The licensee's assumptions follow the guidance of RG 1.183, and therefore, are acceptable to the NRC staff.

The design-basis analysis in Amendment No. 121 demonstrated that the maximum expected fuel rod damage is from an FHA occurring in the reactor containment building. The most limiting case involves a vertical drop approximately 16.7 ft, which results in 137 damaged fuel rods. Based on this information, the damage resulting from a 24-inch drop does not exceed that of the multiple foot design-basis drop. Therefore, the NRC staff finds the Fuel-to-Fuel event analysis to be bounded by the design-basis FHA analysis, which is described in BVPS-2 UFSAR Section 15.7.4, "Radiological Consequences of Fuel Handling 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 DBAs.

3.4.10 SFP Structural Evaluation

The NRC staff's technical review related to the structural integrity of the SFP structure focused on ensuring that the design basis requirements related to the SFP structure, including the SFP reinforced concrete walls and slab, SFP liner, and the existing sub-base beams, will continue to be satisfied following the rerack of the BVPS-2 SFP. As such, the NRC staff's evaluation included a review of the analysis methodology and results, based on the revised loading conditions on the SFP structure, to ensure that the applicable acceptance criteria, described below, are satisfied. The evaluations described below are not coupled with those performed for the mechanical accidents which affect the SFP.

3.4.10.1 Acceptance Criteria

The acceptance criteria related to ensuring the continued structural adequacy of the BVPS-2 SFP following the proposed rerack are found in Appendix D to SRP Section 3.8.4, the OT Position Paper and the BVPS-2 UFSAR. Specifically, the OT Position Paper notes that the design loads utilized in the applicable loading combinations for the SFP walls and slab should be evaluated to ensure that the SFP structure can accommodate any thermal loading changes due to a revised temperature distribution in the SFP. Similarly, the SRP specifies that increased loads resulting from a re-rack must be accounted for in the structural analyses performed in support of any modification which may alter the structural behavior of the SFP. The SRP guidance related to ensuring that the SFP liner maintains its leak-tight integrity is addressed primarily in Section 3.4.3 (Mechanical Accidents) of the SE, given that the loadings due to potential mechanical accidents are the primary challenge to the structural integrity of the SFP liner. However, due to the increased thermal and seismic loads imposed on the SFP liner by the reracking, the structural integrity of the liner with respect to these loads was evaluated in support of the overall reracking project.

In addition to satisfying the acceptance criteria delineated above, the licensee must demonstrate continued compliance with the design basis requirements related to the structural integrity of the BVPS-2 SFP and SFP liner; these requirements are located in Section 3.8.4 of the BVPS-2 UFSAR. As indicated in the BVPS-2 UFSAR, the SFP was designed in accordance with the provisions of ACI 318-71. Therefore, the pertinent provisions in ACI 318-71 must continue to be satisfied following the proposed re-rack. The specific loading combinations which must be considered for concrete structures at BVPS-2 are located in Section 3.8.3.3 of the BVPS-2 UFSAR. The licensee indicated in Reference 20 that, consistent with the original design of the SFP liner, the design stress limits of Mandatory Appendices I, XIII and XIV of the 1974 Edition, with Addenda up to Summer 1976, of the ASME Code, Section III, Division 1, were used to evaluate the structural adequacy of the SFP liner for the proposed re-rack at BVPS-2. Additionally, the licensee noted in Reference 9 that the structural evaluations performed for the sub-base beam structure were carried out in accordance with the provisions of the American Institute of Steel Construction (AISC) Manual for Steel Construction.

3.4.10.2 SFP Evaluation

Section 5.9 of Enclosure C to Reference 1 provides a summary of the licensee's evaluation of the interface loads imposed on the SFP structure as a result of the reracking of the BVPS-2 SFP. Figure 1.1 of the enclosure provides a schematic of the BVPS-2 SFP, including the high-density racks which will replace the current racks at BVPS-2. The licensee notes that the BVPS-2 SFP is a mirror reflection of the BVPS-1 SFP. As such, the evaluation performed to demonstrate the structural integrity of the BVPS-2 SFP was based primarily on the evaluations performed to support the reracking of the BVPS-1 SFP, for which a license amendment, including the corresponding SE, was issued on November 1, 1993.

Using the revised loads resulting from the rerack at BVPS-2, the licensee determined the induced moments on the BVPS-2 SFP structural components (i.e., each of the six walls and the slab) by linearly interpolating the results of the BVPS-1 SFP structural evaluation to the BVPS-2 results for the applicable service conditions. A comparison of these induced moments, due to the rerack, with a component's moment capacity, yields the revised component safety factors. These safety factors are presented in the aforementioned section of the LAR.

By letter dated March 19, 2010, the NRC staff issued an RAI which requested the licensee to provide additional information regarding the SFP component safety factors based on one-way and two-way shear, given that only the safety factors based on moment capacities were previously provided. This RAI also requested information on how the structural adequacy of the SFP components were assessed with respect to the temperature rise expected following the rerack. In its May 21, 2010, response to this RAI, and in its subsequent August 9, 2010, and January 5, 2011 letters, the licensee provided further supplementary information on this issue. The supplemental information found in Reference 9 included an extensive discussion on the loading combinations which must be satisfied for concrete structures in accordance with the BVPS-2 design basis.

The supplemental response to RAI 8 in Reference 12 discusses additional considerations which were given to the response of the SFP liner plate due to the increased temperature gradients associated with the proposed re-racking. The licensee notes that while the temperature rise does affect the thermal loads on the liner plate, the design bases for this component bounds any temperature rise which will accompany the re-racking of the BVPS-2. As such, the design basis for the SFP liner is not affected by the rerack. However, due to the fact that the liner plate is anchored to the SFP, the thermal expansion of the liner will induce tensile loads in the concrete walls. The licensee indicated that revised analyses were performed to determine the shear and moment capacities for the SFP concrete components, considering the tensile loads resulting from the expected temperature rise. The licensee summarized the assumptions utilized to calculate these tensile loads and tabulated the resulting load values in Table 8-1 of Reference 12. The licensee provided the revised safety factors for the one-way and two-way shear evaluations, in addition to the revised safety factors for the bending moment evaluations, which reflect the addition of the aforementioned tensile loads. All of the safety factors reported maintained a value greater than one, demonstrating that the loads induced in each evaluation were below the design basis allowable values derived from ACI 318-71. The licensee confirmed in the supplemental response to RAI 8 that the appropriate capacity reduction factor was utilized in accordance with ACI 318-71.

Given that the capacities of the concrete SFP components, which are referenced extensively above, are dependent on the concrete compressive strength, the NRC staff requested the licensee to confirm that the compressive strength value used in the SFP structural evaluations was in accordance with the BVPS-2 design bases. As such, the licensee's January 5, 2011, supplemental response to RAI-19 confirmed that the design basis compressive strength value of 3,000 psi was utilized in all of the evaluations performed in support of the rerack.

3.4.10.3 Bearing Stress and Sub-Base Beam Evaluations

The description of the BVPS-2 rerack project discussed in this SE noted that the installation of freestanding racks included the placement of a number of bearing pads on which the replacement rack pedestals would rest. Section 5.8 of Enclosure C to Reference 1 provides additional information regarding the bearing pads and discusses the ability of the pads to distribute the pedestal loads such that the compressive stresses induced by peak pedestal loads do not violate the design basis requirements for the BVPS-2 SFP concrete. Select bearing pads will be utilized to accommodate the pedestals which will rest on the existing sub-base beam structure, which is located on the SFP floor. These bearing pads are necessitated due to the fact that the existing sub-base beam structure in the BVPS-2 SFP is not being removed. Therefore, a limited number of pedestals will rest on the existing beam structure.

Utilizing the pedestal loads generated through the aforementioned DYNARACK time history analyses, the compressive stresses induced in the concrete by these limiting pedestal loads were compared with ACI allowable values. The licensee confirmed in References 15 and 16 that the compressive stresses induced in the slab by the pedestal loads were all found to satisfy the BVPS-2 SFP design basis requirements of ACI 318-71. As previously indicated, in demonstrating compliance with the design code of record, the licensee confirmed in Reference 12 that the limits of ACI 318-71 incorporated the use of the design basis concrete compressive strength of 3,000 psi.

The NRC staff also issued an RAI regarding the specific evaluations performed to demonstrate that the sub-base beam structure would remain structurally adequate following the reracking. In the licensee's May 3, 2010, RAI response, it was indicated that only 3 of the 63 rack support pedestals, attached to the base of the freestanding replacement racks, would rest on the existing U-shaped sub-base beams. With respect to the structural evaluation of the existing beams, the licensee indicated that the most limiting components in the load path from the pedestal to the SFP slab were the end welds on the sub-base beams. Based on the results of the structural analyses of these components, the licensee indicated that a safety factor of 1.42 for the end welds would be maintained following the rerack.

In the licensee's response to a subsequent NRC staff request for supplemental information regarding the evaluations performed for the sub-base beam structure and bearing pads (Reference 9), it was indicated that for the 60 pedestal locations not located on top of an existing sub-base beam, the DYNARACK time history analyses provided quantitative confirmation that the horizontal motion of the base, resulting from seismic motion, was less than the space allotted between the pedestals and the sub-base beams, demonstrating that the possibility for interaction between the SFP rack pedestals and the sub-base beam structure under SSE conditions is precluded. Additionally, the licensee stated in its supplemental response that the safety factor derived for the limiting beam end welds, mentioned above, was based on compliance with the AISC allowable stresses for welds.

3.4.10.4 NRC Staff's Evaluation of the SFP

The NRC staff finds the licensee's evaluation of the BVPS-2 SFP, including the SFP concrete components, the SFP liner, and the sub-base beam structure, acceptable because: (1) the licensee provided quantitative information demonstrating that the SFP structure will remain in compliance with the provisions of ACI 318-71, which is the design code of record for the SFP structure, following the proposed re-rack of the SFP; (2) the licensee confirmed that all of the loading conditions, including the effects of thermal expansion of the SFP liner, due to the proposed re-rack were explicitly considered in the structural evaluations in accordance with the BVPS-2 design basis requirements; (3) the licensee demonstrated that the safety factors for the one-way shear, two-way shear, and bending moment evaluations all remain above 1.0 and, therefore, the design basis requirements relative to the BVPS-2 SFP concrete components are satisfied; (4) the licensee demonstrated that the applicable provisions of BVPS-2 design code of record, ACI 318-71, related to concrete bearing stresses will be satisfied following the proposed rerack; (5) the licensee confirmed that the design basis thermal condition for the SFP liner bounds the expected temperature rise due to the proposed reracking of the BVPS-2; (6) the licensee stated in Reference 20 that the stresses induced in the SFP liner due to the seismic interaction with the SFP replacement racks, combined with stresses due to thermal loading, were compared against the original design basis limits for the SFP liner and a minimum safety factor of 1.29 is maintained; (7) the licensee demonstrated through time history analyses that the expected displacements of the rack base are bounded by the displacement which would be required for any interaction with the sub-base beam to occur during a seismic event; and (8) the licensee demonstrated that for the most limiting structural component of the sub-base beam structure (the end welds), adequate margin against the design code of record exists.

Based on the aforementioned considerations, and the licensee's demonstration that all of the design basis requirements relative to the SFP structural components will continue to be satisfied following the rerack, the NRC staff has determined that there is reasonable assurance that the structural integrity of the BVPS-2 SFP will be maintained following the reracking at BVPS-2.

3.4.11 Cask Pit Platform Evaluation

Section 5.7 of Enclosure C to Reference 1 documents the evaluation of the cask pit platform. The platform will be utilized in the cask pit area of the BVPS-2 SFP in order to provide stable support for a fully loaded replacement rack. As such, the platform has been designed to rest on the cask pit floor and to accommodate the loads induced by the support pedestals. The platform was designed in accordance with the provisions of Subsection NF of the ASME Code, Section III. The analysis performed to demonstrate the structural adequacy of the platform, utilized DYNARACK to model a single rack resting on the platform, whereby the loads induced on the platform are the result of a seismic event. The single rack time history analysis performed to simulate this scenario is described in Section 5.5.4 of Enclosure C to Reference 1.

In response to an NRC staff RAI regarding the coupling effects related to the interaction between the platform and the single rack, the licensee indicated in its May 21, 2010, RAI response that a revised analysis had been performed utilizing DYNARACK to perform a time history analysis which modeled both the platform and the single rack. This RAI response included a mathematical representation of the coupling, which is modeled in DYNARACK, illustrating that both the platform structure and the rack were included in the revised analysis. A tabulated comparison between the coupled and decoupled analyses demonstrated that the inclusion of the platform in the DYNARACK time history analysis results in a more limiting structural analysis, based on the fact that a majority of the loads and displacements in the coupled analysis were higher than the decoupled analysis. The licensee indicated that the small displacements resulting from the coupled analysis are insignificant and will not cause the rack to slide off of the cask pit platform nor to tip over. Additionally, the licensee stated that the provisions of Subsection NF of the ASME Code, Section III, continue to be satisfied during the coupled analysis. As part of the revised LAR (Reference 25), which was supplemented by information in Attachment 2 to Reference 20, the licensee indicated that the material and component size specifications for the cask pit rack platform had been modified to incorporate the use of stainless steel in lieu of carbon steel. As a result, the minimum factor of safety resulting from the coupled time history analysis for the cask pit platform and single rack was increased from 1.084 for carbon steel to 1.27 for stainless steel, as this factor relates to satisfying the aforementioned provisions of the ASME Code.

3.4.11.1 NRC Staff's Evaluation of the Cask Pit Platform Evaluation

The NRC staff finds the licensee's assessment of the structural integrity of the cask pit platform acceptable based on the following discussion. The NRC staff finds the licensee's evaluation of a coupled system, consisting of the single replacement rack and the pit platform, bounding, given that it was demonstrated, quantitatively, that the results from this analysis bound those of a decoupled analysis. Furthermore, based on the results of the coupled analysis, the licensee demonstrated that the ensuing displacements will not result in a slide-off or tip-over of the single replacement rack during a seismic event. The licensee also demonstrated that the provisions of the design code of record for the cask pit platform, Subsection NF of the ASME Code, are satisfied for Level D service limits. Accordingly, the NRC staff has determined that there is reasonable assurance that the structural integrity of the cask pit platform will be maintained under the limiting loading events which have been postulated and analyzed.

3.4.12 Fuel Handling Building

The LAR prepared in support of the rerack at BVPS-2 did not include information relative to the effects of the rerack on the behavior of the BVPS-2 fuel handling building, which is described in Section 3.8 of the BVPS-2 UFSAR. As such, by letter dated March 19, 2010, the NRC staff issued two RAIs to the licensee regarding the behavior of the fuel handling building in response to the additional dead weight imposed on the building due to the rerack. In its May 3, 2010, RAI response, the licensee noted that the dynamic model of the BVPS-2 fuel handling building, illustrated in BVPS-2 UFSAR Figure 3.7B-10, includes the weight contribution of the spent fuel in mass number two. Based on the additional weight imposed on the building by a fully-loaded rack configuration, using the replacement racks, the licensee indicated that the total increase due to the added weight represents only 6% of the total building mass and less than 10% of the total mass assigned to mass number two. As such, the licensee concluded that no additional evaluations of the fuel handling building seismic analysis were necessary.

In response to the second RAI, regarding the effects of the rerack on the fuel handling building soil bearing pressure, building sliding analysis, and building overturning analysis, the licensee provided additional information relative to the effects of the rerack on these evaluations. With respect to the soil bearing pressure, the licensee noted that a slight increase in the soil bearing pressure due to the rerack will occur and, as such, will reduce the safety factors associated with the soil bearing capacity for the fuel handling building (Table 2.5.4-4 of the BVPS-2 UFSAR contains these safety factors). However, the licensee noted that the safety factor associated with static loading will only decrease by 1.0 from 11.0 to 10.0, while the dynamic safety factor will be decreased to 5.0 from 6.0, due to the increased weight from the replacement racks.

Comparing these values to the acceptable value of 3.0 for both safety factors in accordance with the BVPS-2 design basis, the licensee concluded that the soil bearing capacity for the fuel handling building remains acceptable.

With respect to the evaluations performed for the overall sliding and overturning of the BVPS-2 fuel handling building, the licensee indicated in its RAI response that the current analyses of record for these evaluations neglect the weight of the SFP racks, including the weight due to the spent fuel. By letter dated August 9, 2010, the licensee confirmed that by including the weight of the replacement racks, including spent fuel, in the sliding and overturning analyses, the limiting safety factors for each analysis were increased. Therefore, the licensee confirmed that the analyses of record for the sliding and overturning evaluations remain bounding with respect to the additional loads imposed on the fuel handling building due to the rerack of the BVPS-2 SFP.

3.4.12.1 NRC Staff's Evaluation of the Fuel Handling Building Structural Analysis

With respect to the effects of SFP reracking on the overall seismic analysis of the fuel handling building, the NRC staff finds the licensee's assessment acceptable because: (1) the mass contribution of the spent fuel racks is at the lowest elevation of the fuel building dynamic model depicted in BVPS-2 UFSAR Figure 3.7B-10 (mass number two); (2) the mass increase associated with the SFP reracking is less than 10% of the total mass assigned to mass number two; and (3) the mass increase associated with the SFP reracking accounts for approximately 6% of the total mass used in the fuel building dynamic model, a value which has a negligible effect on the building's frequency content. The licensee demonstrated that the safety factors related to the soil bearing capacity will continue to satisfy the BVPS-2 UFSAR requirements following the rerack. Additionally, the licensee demonstrated that the current analyses of record for the fuel handling building sliding and overturning evaluations are bounding with respect to any effects induced on these evaluations due to the added weight from the replacement racks.

Based on the above considerations, the NRC staff has determined that there is reasonable assurance that the fuel handling building will maintain its structural integrity following the rerack based on the fact that the licensee has demonstrated that the applicable design basis requirements will continue to be satisfied following the rerack.

3.4.13 NRC Staff Findings on the Structural Behavior of the Replacement Rack Structures and the SFP Structure

Based on its review as described above, the NRC staff finds that the LAR regarding the complete rerack of the BVPS-2 SFP, is acceptable. This acceptance is outlined above and is based on the licensee's compliance with the BVPS-2 design basis requirements related to the structural adequacy of the SSCs affected by the rerack, including the replacement SFP racks, the SFP structure, including the SFP liner, the BVPS-2 cask pit platform, and the BVPS-2 fuel handling building, 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 BVPS-2 associated with the rerack. Therefore, there is reasonable assurance that the structural integrity of the affected SSCs will be maintained following the implementation of the rerack.

3.5 SFP Cooling, Cask Area Cooling, and Heavy Load Handling

3.5.1 SFP Cooling

The current temperature limits on the SFP cooling system are established in Amendment No. 126 to the BVPS-2 operating license. Based on a full core offload with all other storage locations filled and both trains of SFP cooling operational, the NRC staff accepted a maximum bulk SFP water temperature of 159.2 °F for BVPS- 2. The licensee committed to use administrative controls to limit the maximum bulk SFP temperature to 170 °F for a normal full core offload with the failure of one SFP cooling pump, and the licensee determined that the bulk

SFP temperature should not exceed 173 °F for an unplanned full core offload occurring 36 days following shutdown for a previous planned refueling. The administrative controls associated with the normal full-core offload specified a minimum decay time prior to the start of refueling, based on component cooling water temperature to ensure that the heat removal capacity of the SFPCCS with one pump in operation would be adequate to maintain SFP temperature below 170 °F.

In support of the LAR, the licensee performed a thermal-hydraulic evaluation to demonstrate that the BVPS-2 SFP meets the thermal-hydraulic requirements for the safe storage of spent fuel when utilizing the high density racks. The NRC staff has reviewed the inputs, assumptions, and methodology of the licensee's evaluation in order to determine acceptability of the LAR.

The licensee calculated maximum bulk SFP temperature following a normal full core offload with a single active failure and an abnormal full core offload. This calculation was included in Enclosure B to Reference 1. In the normal case (Case I), a full core is transferred to the SFP after a normal operating cycle and one SFPCCS pump is taken as a single active failure. In the abnormal case (Case II), a full core is transferred to the SFP as the result of an abnormal shutdown 36 days after a normal refueling and no single failure is assumed. Both Case I and Case II assume that all available storage locations in the SFP are filled with previously discharged fuel, that movement of spent fuel from the reactor vessel does not start until at least 100 hours after shutdown, and that fuel assemblies are transferred at the rate of 6 fuel assemblies per hour.

The fuel discharge scenarios in Case I and Case II were the same as those used in the analyses supporting Amendment No. 126. However, the analysis provided with the LAR revised portions of the methodology for calculating the heat generation and heat removal from the SFP. The revised methodology assumed a constant component cooling water inlet temperature of 100 °F, included credit for evaporative heat loss from the pool surface, and utilized an alternative, previously-accepted model to determine decay heat generation rate. The results of these calculations are summarized in the table below.

Scenario	Maximum Bulk SFP Temperature (°F)	Coincident Time After Shutdown (hrs)	Coincident Heat Load (MBtu/hr)
Case I	169.0	135	36.13
Case II	169.8	134	42.83

In both cases, the NRC staff found that the assumptions and conditions of the calculation conservatively maximized the heat generation rate within the SFP. However, Figure 6.6.2, "SFP Bulk Temperature, Net SFP Heat and Passive Heat Loss Profiles - Normal Full Core," in Enclosure B to Reference 1 indicated that approximately 2 MBTU/hr resulted from passive heat loss mechanisms (evaporation) at the maximum SFP temperature. The evaporative heat loss compensated for the higher cooling water inlet temperature and the additional heat load resulting from the additional stored fuel capacity. The result of which was that the calculated maximum temperature in both cases remained within the temperature limits approved by the NRC staff in Amendment No. 126 to the BVPS-2 operating license.

The NRC staff found that the crediting the evaporative heat losses in the methodology was insufficiently supported by data verifying its applicability to the BVPS-2 SFP and inconsistent with the guidelines of SRP Section 9.1.3. In response to a RAI regarding the effects of the change in methodology (i.e., credit for evaporation) relative to the licensing basis established by Amendment No. 126, the licensee provided the results of a revised evaluation in the attachment to Reference 9. The licensee later provided a complete revised evaluation in Enclosure B to Reference 10. The revised evaluation calculated the component cooling water temperature necessary to maintain SFP temperature at 170°F for the normal full core offload case (Case I), assuming failure of one SFP cooling pump and without taking credit for evaporative heat losses. The required component cooling water temperatures were calculated with fuel transfer beginning at decay times of 100, 125, and 150 hours.

The licensee found that the existing commitment to administratively control decay time before fuel transfer based on component cooling water supply temperature, which was provided in association with Amendment No. 126, was more restrictive than the current analysis. In Enclosure A to Reference 10, the licensee stated that the commitment remains unchanged and the cooling water temperature and fuel transfer rate would be controlled by procedure. Therefore, the staff found that the licensee had demonstrated adequate cooling capacity for the normal planned refueling full core offload case, by assuming a single failure of one of the SFP cooling pumps and maintaining temperature of the SFP below 170°F.

The licensee conducted further analysis to assess the thermal hydraulic behavior of the SFP in the event of a complete loss of SFP cooling. The time to reach boiling at the SFP surface, the time for water level to drop below 10 ft above the top of the fuel (the adequate shielding depth as described in RG 1.13), and the maximum rate of water loss were calculated for Case I and Case II. For these calculations, the full loss of SFP cooling was assumed to take place when the SFP temperature reached its maximum value identified in the maximum bulk SFP temperature calculation. Additionally, no recovery actions were credited and makeup water to the SFP is assumed to be unavailable. These calculations represent the worst-case scenarios for loss of SFP cooling, and establish available time to identify the condition and initiate corrective actions.

The resulting time to boil, as determined by the licensee, was 2.24 hours for Case I and 1.87 hours for Case II. The time to reach the minimum shielding depth in both scenarios, without any credited action, was greater than 20 hours. Should the SFP cooling pumps trip or SFP water temperature begin to rise, there are multiple, diverse indications available to operators. As described in Section 9.1.3.5 of the BVPS-2 UFSAR, indication is provided for SFP cooling pumps and valves, as well as SFP level, SFP temperature, and fuel pool ion exchanger flow. Annunciation is provided in the main control room for low SFP cooling pump discharge pressure, cooling pump trip, high SFP temperature, and low SFP water level. At 140 °F, the Fuel Pool Demineralizer Supply Temperature High alarm activates, and at 160 °F, the Fuel Pool Temperature High alarm activates. The variety of indication available is sufficient to conclude that operators would identify a loss of cooling and take corrective actions before the temperature at the surface of the SFP reached boiling, and the minimum shielding depth would not be challenged. The licensee has made a commitment to evaluate the current SFP temperature alarms, and modify them if necessary, to ensure these alarms remain appropriate for the SFP following the rerack.

Makeup to the SFP can be provided by the primary grade water system, refueling water storage tank, or fire protection system (via four hose connections around the SFP), and a Seismic Category I source of makeup water is available via the service water system. The makeup capacity of these systems exceeds the maximum water loss rates calculated by the licensee for both worst case scenarios. The NRC staff finds that makeup capacity exists to mitigate a complete loss of SFP cooling.

The licensee performed a calculation of the maximum local water temperature and fuel cladding temperature within the high density racks to demonstrate that nucleate boiling would not occur within the racks. The calculated maximum local water temperature and fuel cladding temperature were 202.8 °F and 227.7 °F, respectively, both of which are below the saturation temperature of the water at the top of the racks. The NRC staff finds that these calculated values, and the conservatisms assumed in the calculation, provide adequate assurance that nucleate boiling will not occur within the high density racks.

By letter dated October 7, 2010, the licensee provided the results of an additional calculation of maximum bulk SFP temperature and time to boil with bounding conditions that are representative of a typical planned refueling outage. This calculation also ignored heat losses, due to evaporative cooling, and considered the realistic condition of both SFP cooling pumps in operation. The bounding conditions of this calculation were provided by the licensee as follows:

- Decay heat load is calculated assuming a full core offload initiated at 100 hours after reactor shutdown following a full power cycle of 2,918 megawatts thermal with the SFP containing more than 1,690 assemblies and previous batch discharges consisting of 72 assemblies, each beginning in the fall 2009 refueling outage; and

- Heat removal capability is based on an inlet component cooling water (CCW) temperature of 88 °F, a total cooling system flow rate of 2,200 gallons per minute (gal/min), a total tube side flow rate of 1,500 gal/min, two SFP cooling pumps, and two SFP cooling heat exchangers having a tube side fouling factor of 0.0010 [$1/Btu/hr/ft^2/°F$] and a shell side fouling factor of 0.0005 [$1/Btu/hr/ft^2/°F$]; and

- Alternate heat removal paths, such as evaporative cooling, are conservatively neglected.

With these conditions, the calculated maximum bulk SFP temperature was 150 °F. Assuming a loss of SFP cooling concurrent with the maximum bulk SFP temperature, the time to boil is greater than three hours and the boil-off rate is less than 80 gal/min. This demonstrates that normal refueling operations will not challenge the currently licensed maximum SFP temperature considered in Amendment No. 126 with both SFP cooling water pumps in operation at expected refueling component cooling water inlet temperatures

3.5.2 <u>Cask Area Cooling</u>

In order to provide additional fuel storage space during the installation of the new high density racks, one new rack will be temporarily placed in the cask area and loaded with spent fuel. As described in the BVPS- 2 UFSAR, the cask area is designed to the same criteria as the SFP and is connected to the SFP by an open gate. Spent fuel is not normally stored in the cask area, so the licensee performed a separate calculation to determine the maximum local water and cladding temperatures to demonstrate the acceptability of temporary storage of fuel in this area.

In its RAI response, the licensee described the calculations for the fuel rack in the cask pit in detail. The licensee responded that there was no difference in the bulk SFP temperature calculation between the cask pit and the SFP because the two areas communicate across an open gate.

The BVPS-2 UFSAR description of the SFP Cooling System states that the discharge of the fuel pool cooling pumps penetrates the SFP area and the cask area, providing a supply of cooled water in both areas. In the maximum local water temperature calculation, the licensee conservatively assumed that water entering the rack in the cask pit was at the maximum bulk SFP temperature (170 °F). The licensee also stated that fuel placed in the cask pit area for temporary storage must have been in the SFP for at least 18 months, further reducing the heat load in this rack. The maximum local water and fuel cladding temperatures in the cask pit were calculated to be 177 °F and 179 °F, respectively.

The NRC staff reviewed the inputs, assumptions, and methodology for the evaluation of cooling in the cask pit area and finds that acceptable cooling would be provided for fuel stored temporarily in this area. The maximum local water and fuel cladding temperature are below the saturation temperature of the water at the top of the rack in the cask area, ensuring that nucleate boiling does not occur. Decay time of at least 18 months is an assumption in the licensee's calculation; therefore, fuel cooled for less than 18 months may not be stored in the cask area.

3.5.3 Heavy Load Handling

The replacement of the old spent fuel storage racks in the BVPS-2 SFP would require removing all 17 existing racks and installing 15 high density racks. In order to perform the rack lifts, a temporary crane capable of safely handling heavy loads will be installed on the rails used by the existing movable platform. The licensee has stated that all heavy load handling would be done in compliance with the guidelines of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The heavy load handling operations and equipment were described in the LAR and the supplement to the LAR.

Section 5.1.1 of NUREG-0612 provides seven criteria for heavy load handling systems. Criterion 1 of Section 5.1.1 provides guidance for the selection of acceptable safe load path for the movement of heavy loads. The LAR supplement describes the following details of the safe load paths:

- Heavy loads will never be moved over stored fuel in the SFP.
- The cask pit will be protected with a cover (designed to withstand the impact of a dropped rack) when new racks are carried above it.
- A minimum horizontal distance of 3 feet will be maintained between lifted racks and stored fuel.

- Racks will be lowered to the minimum height above the pool floor before commencing any horizontal movement.

- Fuel shuffles will maintain the greatest possible between stored fuel and a heavy load being lifted out of the pool.

The selection of safe load paths outlined above is acceptable and satisfies the guidance of criterion 1 of Section 5.1.1.

Criteria 2 and 3 of Section 5.1.1 give guidance on the procedures that should be in place to govern heavy load handling operations and the training provided to crane operators. Procedures covering load handling operations have been identified in the LAR, and address the inspection of heavy loads prior to movement, steps for moving loads, defining the safe load path, and other special considerations pertaining to the installation and removal of the racks. The licensee has stated that crane operators will be qualified and trained in accordance with Chapter 2-3 of ANSI B30.2-1976, "Overhead and Gantry Cranes." The NRC staff finds that the identified procedures and training requirements satisfy criteria 2 and 3 of section 5.1.1.

Criteria 4, 5, 6, and 7 provide standards for the design and testing of specially designed lifting devices, all other lifting devices, and the crane. Specially designed lift rigs will be used for the movement of racks during heavy load handling operations. As stated in the amendment request all lifting devices should comply with the provisions of ANSI N14.6-1992, "Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More." The LAR did not describe the use of other lifting devices during the load handling operations. The design of the temporary crane is described in the LAR supplement and meets the criteria of Chapter 2-1 of ANSI B30.2-1976. Furthermore, the licensee described that a load test at 125% of rated capacity prior to installation, and testing on site as recommended in Chapter 2-2 of ANSI B30.2-1976 will be performed. The NRC staff finds that the description of design and testing requirements contained in the LAR satisfy criteria 4, 5, 6, and 7 of Section 5.1.1.

Section 5.1.2 of NUREG-0612 outlines additional guidelines for handling heavy loads in the SFP area. These guidelines include the use of a single failure proof crane, or one with mechanical stops or interlocks combined with acceptable load drop analyses. The temporary crane does not meet the guidelines of Section 5.1.2, because it is not single-failure-proof and does not have mechanical stop or electrical interlocks preventing the movement of the heavy load within 25 feet horizontal of "hot" spent fuel. The design of the temporary crane and special lifting devices, combined with the limited number and duration of lifts and the use of safe load paths, makes the likelihood and consequence of a load drop very small. The load drop analysis provided by the licensee demonstrates that a postulated drop would not affect the water-tight integrity of the pool and damage to the pool liner would be minimal. As a result, the NRC staff has determined that the temporary crane would be sufficient to meet the intent of Section 5.1.2 of NUREG-0612.

The NRC staff finds that the licensee's proposed control of heavy load handling operations during installation of the new racks and removal of the current racks to be acceptable. In conformance with the intent of NUREG-0612, the potential for a load drop and the effects of any drop would be minimized.

3.5.4 NRC Staff Findings of the SFP Cooling, Cask Area Cooling, and Heavy Load Handling

The licensee has provided sufficient information to demonstrate the SFP cooling system will remain capable of providing adequate cooling with an increased number of spent fuel assemblies stored in the SFP, using administrative controls to control the timing of fuel transfer. The SFP cooling and makeup capacity continue to satisfy the criteria the NRC staff applied in the review of Amendment No. 126. During SFP rerack operations, the licensee described acceptable measures to ensure sufficient cooling would be provided to the cask pool area and all heavy load lifts would be performed in accord with the intent of NUREG-0612. Therefore, the NRC staff finds that the proposed increase in fuel storage capacity and the associated controls on spent fuel cooling and heavy load handling applied during the storage capacity expansion are acceptable.

4.0 LICENSEE COMMITMENT

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the regulatory commitments below are provided by the licensee's administrative processes, including its commitment management program (See Regulatory Issue Summary 2000-017, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff"). The NRC staff has agreed that Nuclear Energy Institute 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," provides reasonable guidance for the control of regulatory commitments made to the NRC staff. The NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

- A Metamic surveillance program will be implemented for the BVPS-2 SFP in order to monitor the integrity and performance of Metamic.

- A process will be established prior to receipt of the next reload batch of BVPS-2 fuel to ensure that the design features and operating parameters of fuel used in the future at BVPS-2 are consistent with the assumptions of the criticality analysis.

- The licensee will evaluate and if necessary, modify the current 140 °F and 160 °F SFP alarm setpoints, in conjunction with implementation of the BVPS-2 rerack LAR. The evaluation will ensure that the alarm setpoints are consistent with the analysis assumptions representative of bounding conditions associated with planned refueling outages.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 (75 FR 11566). 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.

7.0 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.

8.0 REFERENCES

1. Letter from P. P. Sena, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, License Amendment Request No. 08-027, Unit 2 Spent Fuel Pool Rerack," with Enclosure B (proprietary) and Enclosure C (non-proprietary), "Licensing Report for Beaver Valley Unit 2 Rerack," dated April 9, 2009. (ADAMS Accession Nos. ML091210251 (letter) and ML091210263 (Enclosure C)).
2. Letter from P. P. Sena, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Additional Technical Information Pertaining to License Amendment Request No. 08-027, (TAC No. ME1079)," dated June 15, 2009. (ADAMS Accession No. ML091680614).
3. Letter from R. A. Lieb, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Response to Request for Additional Information for License Amendment Request No. 08-027, Unit 2 Spent Fuel Pool Rerack (TAC No. ME1079)," dated January 18, 2010. (ADAMS Accession No. ML100191805).
4. Letter, from R. K. Brosi, FirstEnergy Nuclear Operating Company, to U. S. NRC Document Control Desk, re: "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73 Response to NRC Staff Request for Additional Information Regarding Criticality Analyses Supporting a Spent Fuel Pool Re-rack for Unit 2, (TAC No. ME1079)" March 18, 2010. (ADAMS Accession No. ML100820165).
5. Letter from N. S. Morgan, NRC, to P. A. Harden, FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station, Unit No. 2 – Request for Additional Information Re: Spent Fuel Pool Rerack License Amendment (TAC No. ME1079)," dated March 19, 2010. (ADAMS Accession No. ML100760584).
6. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No.

NPF-73, Response to Request for Additional Information Related to Beaver Valley Power station Unit No. 2 Spent Fuel Pool Rerack License Amendment Request (TAC No. ME1079)," dated May 3, 2010. (ADAMS Accession No. ML101260059).

7. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Response to Request for Additional Information Related to Beaver Valley Power station Unit No. 2 Spent Fuel Pool Rerack License Amendment Request (TAC No. ME1079)," dated May 21, 2010. (ADAMS Accession No. ML101460057).

8. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to U. S. NRC Document Control Desk, re: "Beaver Valley Power Station, Unit No. 2, Docket No. 50-412, License No. NPF-73, Remainder of Responses to NRC Staff Request for Additional Information Regarding Unit 2 Spent Fuel Pool Rerack Criticality Analyses (TAC No. ME1079)," June 1, 2010. (ADAMS Accession No. ML101610118).

9. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Response to Request for Additional Information for License Amendment Request No. 08-027, Unit 2 Spent Fuel Pool Rerack (TAC No. ME1079)," dated August 9, 2010. (ADAMS Accession No. ML102240256).

10. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to U. S. NRC Document Control Desk, re: "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73 License Amendment Request for Unit 2 Spent Fuel Pool Rerack (TAC No. ME1079)," October 18, 2010. (ADAMS Accession No. ML102940454).

11. Public Meeting Summary, "Summary of September 27, 2010, Category I Meeting with FirstEnergy Nuclear Operating Company on the Request for Additional Information Regarding the Spent Fuel Pool Rerack for Beaver Valley Power Station, Unit No. 2 (TAC No. ME1079)," dated October 18, 2010. (ADAMS Accession No. ML102790351).

12. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Supplemental Information for Beaver Valley Power Station Unit 2 Spent Fuel pool Rerack License Amendment Request (TAC No. ME1079)," dated January 5, 2011. (ADAMS Accession No. ML110110217).

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

14. 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 2, March 2007.

15. U.S. Nuclear Regulatory Commission, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System At a Nuclear Power Plant," NUREG-1864, March 2007.

16. Morton, D.K., Snow, S.D, Rahl, T. E. & Blandford, R. K, Idaho National Laboratory, "Impact Testing of Stainless Steel Material at Room and Elevated Temperatures," *Proceedings of the 2007 ASME Pressure Vessels and Piping Division Conference, 22-26 July 2007*, San Antonio, TX.

17. 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).

18. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to U. S. NRC Document Control Desk, re: "Beaver Valley Power Station, Unit No. 2, Docket No. 50-412, License No. NPF-73, Responses to NRC Staff Request for Additional Information Regarding Unit No. 2 Spent Fuel Pool Rerack Criticality Analyses (TAC No. ME1079)," February 18, 2011. (ADAMS Accession No. ML110540328).

19. Conference Call Summary,"Summary of Clarification Call Re: Beaver Valley 2 Spent Fuel Pool Rerack AMD (ME1079)," dated March 10, 2011. (ADAMS Accession No. ML110750478).

20. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Supplemental Information for Beaver Valley Power Station Unit 2 Spent Fuel pool Rerack License Amendment Request (TAC No. ME1079)," dated March 21, 2011. (ADAMS Accession No. ML110800570).

21. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998. (ADAMS Accession No. ML003728001).

22. *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, NUREG/CR-6698, U.S. Nuclear Regulatory Commission, Science Applications International Corporation, January 2001.

23. *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*, NUREG/CR-6979 (ORNL/TM-2007/083), U. S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 2008.

24. *Criticality Benchmark guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*, NUREG/CR-6361 (ORNL/TM-13211), U. S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 1997.

25. Letter from P. A. Harden, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, Chapters 5 and 7 of Holtec Licensing Report, In Support of License Amendment Request for Unit 2 Spent Fuel Pool Rerack (TAC No. ME1079)," dated February 18, 2011. (ADAMS Accession No. ML110530463).

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Date: April 29, 2011

April 29, 2011

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SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING THE SPENT FUEL POOL RERACK (TAC NO. ME1079)

Dear Mr. Harden:

The Commission has issued the enclosed Amendment No. 173 to Renewed Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit No. 2 (BVPS-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 9, 2009, as supplemented by letters dated June 15, 2009, January 18, 2010, March 18, 2010, May 3, 2010, May 21, 2010, June 1, 2010, August 9, 2010, October 7, 2010, October 18, 2010, January 5, 2011, February 18, 2011, March 18, 2011, and March 21, 2011.

The amendment modified TSs to support the replacement of existing Boraflex neutron absorber fuel storage racks in the BVPS-2 spent fuel pool with new high density, Metamic neutron absorber fuel storage racks, which will increase the total storage locations from 1,088 to 1,690.

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

Sincerely,
/ra/
Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412
Enclosures:
1. Amendment No. 173 to NPF-73
2. Safety Evaluation
cc w/encls: Distribution via Listserv

ADAMS Accession No.: ML110890844 *Input provided. No substantive changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	DCI/CSGB/BC	DRA/AADB/BC	DSS/SBPB/BC	DE/EMCB/BC
NAME	NMorgan	SLittle	RTaylor*	TTate*	GCastro*	MKanna*
DATE	4/6/2011	4/6/2011	8/21/2009	6/8/2010	2/8/2011	3/31/2011

OFFICE	DSS/SRXB/BC	DIRS/ITSB/BC	OGC	LPLI-1/BC
NAME	TUlses*	RElliott	DRoth	NSalgado
DATE	3/24/2011	4/6/2011	4/29/2011	4/29/2011

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