September 28, 2007

Mr. Timothy G. Mitchell
Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISIONS TO TECHNICAL SPECIFICATIONS TO SUPPORT PARTIAL RE-RACK AND REVISED LOADING PATTERNS IN THE SPENT FUEL POOL (TAC NO. MD4994)

Dear Mr. Mitchell:

The Commission has issued the enclosed Amendment No. 273 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 30, 2007, as supplemented by letter dated June 13, 2007.

The amendment revised TS 3.9.12, "Fuel Storage," and its associated tables, figures, and surveillance requirements, TS 5.3, "Fuel Storage," and added TS 6.5.17, "Metamic™ Coupon Sampling Program." The ANO-2 TS 3.9.12 is changed to: (1) support higher fuel assembly uranium-235 (U-235) enrichment; (2) apply the appropriate loading restrictions; and (3) delete the dry cask loading restrictions. ANO-2 TS 5.3.1 b is changed to reflect a different spent fuel pool boron concentration that is needed to assure K-effective remains less than or equal to 0.95. ANO-2 TS 5.3.2a is modified to reflect a higher fuel assembly U-235 enrichment. A new coupon sampling program is added as TS 6.5.17, and TS 4.9.12.d is added to direct performance of the coupon sampling program.

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

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures: 1. Amendment No. 273 to NPF-6
2. Safety Evaluation

cc w/encls: See next page

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273
Renewed License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated March 30, 2007, as supplemented by letter dated June 13, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 273, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Renewed Facility Operating License No. NPF-6 and Technical Specifications

Date of Issuance: September 28, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 273

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Renewed Facility Operating License No. NPF-6 and 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.

Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3/4 9-14	3/4 9-14
3/4 9-15	3/4 9-15
3/4 9-16	--
3/4 9-17	--
3/4 9-18	--
3/4 9-19	--
3/4 9-20	--
3/4 9-21	--
5-2	5-2
--	6-18a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 273 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By application dated March 30, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071000250) (Reference 1), as supplemented by letter dated June 13, 2007 (ADAMS Accession No. ML071780610) (Reference 14), Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit No. 2 (ANO-2). Specifically, the proposed changes would revise the ANO-2 TSs 3.9.12 and 5.3 to:

1. Credit Metamic™ racks that will replace the existing Boraflex racks in Region 1 of the ANO-2 spent fuel pool (SFP),

2. Redefine the loading pattern based on the new Region 1 racks and the remaining racks, which are designated as Region 2 racks,

3. Allow an increase in the maximum fuel assembly uranium-235 (U-235) enrichment from the current U-235 enrichment of 4.55 ± 0.05 weight percent (wt%) to a maximum of 4.95 wt%,

4. Delete TS 3.9.12.d and the related surveillance requirement (SR) associated with dry cask loading, and

5. Revise the boron concentration credited to assure the effective neutron multiplication factor (K-effective or K_{eff}) remains less than 0.95.

The change also includes the addition of a new TS 6.5.17 for Metamic™ coupon sampling program to monitor the potential changes in the characteristics of Metamic™.

The supplement dated June 13, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 8, 2007 (72 FR 26175).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," provides a list of the minimum design requirements for the nuclear power plants. The following GDCs are applicable to this review:

- GDC 2, "Design bases for protection against natural phenomena," and GDC 4, "Environmental and dynamic effects design bases," to demonstrate that the structural integrity of the fuel racks and the SFP structure are maintained.

- GDC 62, "Prevention of criticality in fuel storage and handling." The licensee must prevent criticality in the fuel handling and storage system by physical systems or processes, preferably by the use of geometrically safe configuration.

Section 50.68 of 10 CFR, "Criticality accident requirements," provides the NRC regulatory requirements and acceptance criteria for prevention of criticality in the spent fuel storage racks as they apply to ANO-2.

NUREG-0800, "Standard Review Plan for the Review Safety Analysis Reports for Nuclear Power Plants" (Reference 6), provides guidance for performing design and analysis to demonstrate compliance with the regulation. Standard Review Plan (SRP) Section 9.1.2, "Spent Fuel Storage," ensures that there are no potential mechanisms that will: (1) alter the dispersion of boron carbide (B_4C) in the Metamic™ panels, and/or (2) cause physical distortion of the tubes retaining the stored fuel assemblies. This TS request includes the implementation of a coupon sampling program to confirm the capability of the Metamic™ material to perform the intended safety function in the SFP and a thermal-hydraulic analysis of the racks to ensure that adequate natural circulation is provided to cool the stored fuel. Additional guidance for SFP and fuel rack compliance is found in SRP Section 9.1.3, "Spent Fuel Storage Facility Design Basis" (Reference 18), Section 3.7.1, "Seismic Design Parameters," Section 3.8.4, "Other Seismic Category I Structures," Section 3.8.5, "Foundations," and Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis."

The L3 crane is a 130-ton capacity single failure-proof bridge crane designed in accordance with NUREG-0554, "Single Failure-Proof Cranes for Nuclear Power Plants." NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines and recommendations to assure safe handling of heavy loads such as the SFP fuel storage racks. American National Standards Institute (ANSI) N14.6-1993 provides criteria to be met for special lifting devices being used in conjunction with a single failure-proof crane. The NRC staff evaluated the proposed load handling plan according to this guidance.

3.0 TECHNICAL EVALUATION

ANO-2 uses an SFP for storage of irradiated nuclear fuel. The SFP is licensed for storage of 988 fuel assemblies in rack modules. The ANO-2 SFP contains 12 independent rack modules designed to hold the spent fuel assemblies and rod control assemblies in storage for long-term decay. These modules are divided into two regions. The current Region 1 racks employ Boraflex as the neutron absorbing (poison) material. These racks are being replaced by racks built by Holtec International (Holtec), which use Metamic™ panels as the poison material. The Region 2 racks do not have any poison material. Region 1 presently consists of one 8-cell x 9-cell rack module and two 9-cell x 9-cell rack modules. Region 2 consists of nine rack modules. The new Holtec racks consist of one 8-cell x 9-cell rack module, and two 9-cell x 9-cell rack modules, which match the existing configuration for the racks being replaced. The racks are freestanding and rest on a base plate which in turn rests on four pedestals at the bottom of the pool. Motion by sliding is restrained by friction between the rack module pedestal supports and the SFP floor. The 12 racks, originally designed by Westinghouse, are self-supporting and are not connected to each other or to the SFP walls. The Holtec International racks with Metamic™ panels are also dimensionally similar to the existing Westinghouse Boraflex racks. The cell inside diameter and the cell pitch remain the same, but the overall height is slightly greater for the new racks. The Metamic™ panels are considered non-structural, and are therefore included in the rack models only as added mass. Since these panels are “softer” than the stainless steel cell walls and sheathing, demonstration of the structural adequacy for the cell walls and sheathing also provides assurance of the structural adequacy of the Metamic™ panels. Decay heat from these fuel assemblies is removed by the spent fuel cooling system. The spent fuel cooling system operates by recirculating SFP coolant water through two pumps and one shell and tube heat exchanger. The spent fuel cooling system draws water from the top of the SFP and returns cooled water to the opposite side of the SFP; flow through the fuel assemblies at the bottom of the pool is achieved by natural circulation.

The proposed change will modify the ANO-2 TS to support a partial re-rack of the storage racks in the ANO-2 SFP. The proposed TS changes support a planned modification to the ANO-2 SFP that will replace the Region 1 fuel racks with new racks containing Metamic™ neutron poison panels, but does not make any change to the number of fuel assemblies stored in this region. Entergy submitted by letter dated August 8, 2002, a topical report (Reference 2) to support the use of Metamic™ in SFP applications. The topical report was approved by the NRC on June 17, 2003 (Reference 3).

All lifting and transferring of the new racks will be performed by the L3 fuel handling crane in conjunction with a designated special lifting device. The L3 crane is a 130-ton capacity single failure-proof bridge crane designed in accordance with NUREG-0554. The special lifting device consists of a tubular, rectangular frame with four lift poles extending below. The crane lifts the frame using attached angle plates and the poles engage to lift the rack. All transferences of the new and existing fuel racks will take place along safe load paths determined prior to the lifting.

3.1 Proposed Changes to the ANO-2 TSs

3.1.1 TS 3.9.12, Refueling Operations, Fuel Storage

TS 3.9.12, defines the actions, the associated SRs and figures that define the storage restrictions, the maximum fuel enrichment, and the minimum required SFP boron concentration for the ANO-2 SFP. A discussion of the proposed changes to the TS 3.9.12 is provided below:

- TS 3.9.12.a, which defines the fuel assembly U-235 enrichment, will be changed to allow storage of fuel assemblies with an initial U-235 enrichment of 4.95 wt%.

- TS 3.9.12.b and Figure 3.9-2 define loading restrictions for fuel assemblies that are stored in the ANO-2 SFP. Figure 3.9-2 will be deleted. A new Table 3.9-1 will be added to TS 3.9.12.b, which will provide the revised SFP loading restrictions.

- TS 3.9.12.d will be deleted.

- SR 4.9.12.a will be modified to reflect the new fuel assembly U-235 enrichment.

- SR 4.9.12.b will be changed to reference the proposed Table 3.9-1 instead of deleted Figure 3.9-2.

- SR 4.9.12.d and Figure 3.9-1 will be deleted as they are no longer applicable and replaced by a new SR. The new SR will direct performance of the coupon sampling program.

3.1.2 Technical Specification 5.3, Design Features, Fuel Storage

TS 5.3.1.b defines that K_{eff} will be maintained less than or equal to 0.95 if the SFP racks are fully flooded with at least 240 parts per million (ppm) of borated water. The criticality analysis, as allowed by 10 CFR 50.68, will continue to credit boron (revised from 240 to 452 ppm) to assist in maintaining K_{eff} less than or equal to 0.95 during normal operating conditions.

The boron concentrations for each region of the SFP, as determined by the criticality analyses to assure that K_{eff} remains below 0.95, are bounded by the TS value. The fuel loading patterns in the proposed changes, as defined by the criticality safety analysis, are governed, as they are currently, by procedure.

TS 5.3.2.a defines a maximum U-235 enrichment of 4.55 ±0.05 wt% for fuel assemblies stored in the new fuel storage racks. The proposed change will allow the maximum U-235 enrichment to be 4.95 wt% in the new fuel storage racks.

3.1.3 Technical Specification 6.5.17, Metamic™ Coupon Sampling Program

A coupon sampling program will be added as proposed TS 6.5.17. As noted above, SR 4.9.12.d will be replaced with a new SR to direct performance of the coupon sampling program. The provisions of SR 3.0.2 are applicable to the Metamic™ Coupon Sampling Program. The provisions of SR 3.0.3 are not applicable to the Metamic™ Coupon Sampling Program.

3.2 Evaluation of Proposed Changes to ANO-2 TSs

The NRC staff has reviewed the following aspects for the use of Metamic™ in the ANO-2 SFP: 1) criticality safety analyses, 2) boron dilution, 3) 10 CFR 50.68, "criticality accident requirements," 4) Metamic™ coupon sampling program, 5) thermal-hydraulic considerations of the new racks, 6) heavy load handling, 7) structural and seismic analyses, 8) dose analysis, and 9) occupational radiation exposure.

3.2.1 Criticality Safety Analyses

The objective of the SFP criticality analysis is to insure that the K_{eff} is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity. In addition, the analysis must demonstrate that K_{eff} is less than 1.0 under the assumed loss of soluble boron in the pool water. The maximum calculated reactivities include a margin for uncertainty in the reactivity calculations, including manufacturing tolerances, and are calculated with a 95 percent probability at a 95 percent confidence level. Reactivity effects of abnormal and accident conditions are also evaluated to assure that under all credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit of 0.95.

The NRC has defined acceptable methodologies for performing SFP criticality analyses in three documents:

1. NUREG-0800, Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Draft Revision 4;

2. Proposed Revision 2 to RG 1.13, “Spent Fuel Storage Facility 3 Design Basis,” and

3. Memorandum from L. Kopp (NRC) to T. Collins (NRC), “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants” (Reference 15).

3.2.1.1 Evaluation of Criticality Safety Analyses

The specific evaluations performed by the licensee for the ANO-2 SFP are:

The Region 1 racks were evaluated for storage of fresh fuel assemblies with a maximum nominal initial enrichment of 4.95 wt% U-235. Calculation of the maximum K_{eff} is done for the cases of no soluble boron and with soluble boron.

The Region 2 racks were evaluated for storage of spent fuel assemblies with specific burnup requirements as a function of initial enrichment between 2.0 wt% and 4.95 wt% U-235 and cooling times between 0 and 20 years. Minimum burnup values at varying enrichment and cooling time were summarized, and the calculation of the maximum K_{eff} for 4.95 wt% U-235 at zero years cooling time with no soluble boron, and with soluble boron was determined.

The Region 2 racks were evaluated for storage of fresh fuel assemblies with a maximum nominal enrichment of 4.95 wt% U-235 in a checkerboard configuration with empty storage cells. The Region 2 racks were also evaluated for storage of lower burned assemblies on the rack periphery, facing the spent fuel walls. Minimum burnup versus enrichment values for the peripheral cells are summarized in Table 4.7.9 (Reference 1), and the calculation of the maximum K_{eff} for 4.95 wt% U-235 at zero years cooling is given in Table 4.7.6 (Reference 1) with no soluble boron and Table 4.7.7 (Reference 1) with soluble boron.

Reactivity effects of abnormal and accident conditions have also been evaluated. A summary of the types of accidents analyzed and the soluble boron required ensuring that the maximum K_{eff} remains below 0.95, are shown in Tables 4.7.10 and 4.7.11 (Reference 1), for Region 1 and Region 2, respectively. The most limiting accident condition involves misleading a fresh fuel assembly, enriched to 4.95 wt% U-235, in an empty storage location of the Region 2 storage rack, when a checkerboard configuration is used. A minimum soluble boron concentration of 881 ppm must be maintained in the SFP to ensure that the maximum K_{eff} is less than 0.95 under accident conditions.

In addition to the analysis performed for each individual rack detailed above, the possibility of an increased reactivity effect due to the rack interfaces within and between the racks was analyzed. Interfaces within the rack include spent and fresh fuel loading patterns within the same rack to determine acceptability. Interface calculations between racks include Region 2-Region 2 and Region 1-Region 2. The calculated reactivity from the interface calculation is then compared to the calculated reactivity from the reference infinite array calculations. From the summary of the results, the NRC staff has concluded the following regarding the reactivity effect of the interfaces:

- In the Region 2 racks, a fresh fuel checkerboard and uniform spent fuel loading may be placed adjacent to each other in the same rack. If both patterns are placed in a single rack, no fresh fuel assembly may be placed with more than one face adjacent to a spent fuel assembly.

- In the Region 2 racks, if adjacent racks contain checkerboard of fresh fuel assemblies, the checkerboard must be maintained across the gap (i.e., fresh fuel assemblies may not face each other across a gap).

- In the Region 2 racks, one rack may contain a checkerboard of fresh fuel and empty storage locations and the adjacent rack may contain spent fuel with no loading restrictions.

3.2.1.1.1 Computational Methodology

The principal method used by the licensee for the criticality analysis of the high-density storage racks is the three-dimensional Monte Carlo code MCNP4a (Reference 16). MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National

Laboratory. MCNP4a was selected because it has been used previously and verified for criticality analyses and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V and ENDF/B-VI. Exceptions are two lumped fission products used by the CASMO-4 depletion code that do not have corresponding cross sections in MCNP4a. For these isotopes, the CASMO-4 cross sections are used in MCNP4a. This approach has been validated by showing that the cross sections result in the same reactivity effect in both CASMO-4 and MCNP4a. Benchmark calculations indicate a bias of 0.0009 with an uncertainty of ±0.0011 for MCNP4a, evaluated with a 95 percent probability at the 95 percent confidence level. All these codes are industry standard codes that were validated through benchmarking to relevant critical experiments. The NRC staff has historically found these codes acceptable for licensing applications.

Fuel Depletion analyses during core operation were performed with CASMO-4 (Reference 17) (using the 70-group cross-section library), a two-dimensional multi-group transport theory code based on capture probabilities. CASMO-4 is used to determine the isotopic composition of the spent fuel. In addition, the CASMO-4 calculations are restarted in the storage rack geometry, yielding the two-dimensional infinite multiplication factor (K_{inf}) for the storage rack to determine reactivity effect of fuel and rack tolerances, temperature variation, depletion uncertainty, and to perform other studies. For all calculations in the SFP racks, the Xe-135 concentration in the fuel is conservatively set to zero.

Furthermore, to assure the true reactivity is always less than the calculated reactivity, the following conservative design assumptions are employed:

1) Moderator is borated or unborated water at a temperature in the operating range that results in the highest reactivity, as determined by analysis.

2) Neutron absorption in minor structural members is neglected (i.e., spacer grids are replaced by water).

3) The effective multiplication factor of an infinite radial array of fuel assemblies is used in the analyses, except for the assessment of certain abnormal/accident conditions in which neutron leakage is inherent, such as for the analysis of the peripheral rack cells.

4) The B_4C loading in the neutron absorber panels is nominally 30.5 wt%, with an uncertainty of +0.5/-1.0 wt%.

5) The presence of burnable absorbers (B_4C, Gadolinium, Erbium, Integral Fuel Burnable Absorber) in fresh fuel is neglected. This is conservative as burnable absorbers will reduce the reactivity of the fresh fuel assembly.

6) When multiple enrichments are used within an assembly, the average enrichment is used for all fuel pins (i.e. distributed enrichments are neglected).

The maximum K_{eff} is determined from the MCNP4a-calculated K_{eff}, the calculational bias, the temperature bias, and the applicable uncertainties and tolerances (bias uncertainty, calculational uncertainty, rack tolerances, fuel tolerances, depletion uncertainty). The licensee used the preceding variables in the following equation to determine the maximum K_{eff}:

$$\text{Max } K_{eff} = \text{Calculated } K_{eff} + \text{biases } [\textstyle\sum_{I} (\text{uncertainty}_{i})^2]^{1/2}.$$

The NRC staff has concluded that this equation is acceptable for the calculation of the maximum K_{eff}.

Based on the review of information submitted to the NRC staff regarding Metamic™ rack inserts, and the analyses of increasing allowable initial fuel assembly U-235 enrichment to a maximum U-235 enrichment of 4.95 wt%, the NRC staff finds that under all credible abnormal and accident conditions, the reactivity in the SFP will not exceed the regulatory limit of 0.95. TS 3.9.12.d and SR 4.9.12.d have no function and can be deleted as the storage restrictions it references have been deleted (Figures 3.9-1 and 3.9-2). The new loading patterns are contained in Table 3.9-1, “SFP Loading Restrictions,” which maintains the new loading restrictions for Regions 1 and 2. Therefore, the proposed changes to the TS 3.9.12.a and SRs 4.9.12.a and b, and the deletion of TS 3.9.12.d and SR 4.9.12.d for ANO-2, are acceptable with regard to the SFP criticality safety analyses and the use of Metamic™.

3.2.1.2 Boron Dilution Evaluation

In order to assure K_{eff} remains less than 0.95 for the allowable storage configuration of both spent fuel and fresh fuel assemblies, a minimum SFP boron concentration is required. TS 5.3.1.b defines that K_{eff} will be maintained less than or equal to 0.95 if the spent fuel racks are fully flooded with at least 240 ppm of borated water. The soluble boron in the SFP water is required by TSs to be maintained at all times at greater than 2000 ppm. For this calculation, it is conservatively assumed that the soluble boron in the SFP is at the minimum TS value of 2000 ppm. An evaluation was performed based on the ANO-2 SFP data.

It is determined that the required minimum soluble boron concentration is 452 ppm under normal conditions and 881 ppm for the most serious credible accident scenario. The volume of water in the pool is approximately 198,000 gallons. Therefore, large amounts of unborated water would be necessary to reduce the boron concentration from 2000 ppm to 881 ppm or to 452 ppm. This will provide operators ample time to restore the boron concentration. Based on this evaluation, the NRC staff finds the proposed change to TS 5.3.1.b to increase the minimum soluble boron concentration from 240 to 452 ppm is acceptable.

Based on the above, the NRC staff finds that the proposed license amendment request (LAR) meets the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The NRC staff concludes that the licensee conducted the necessary analysis in accordance with NRC guidelines and the ANSI standards. The analysis shows that the design and operation of ANO-2 will maintain the maximum neutron multiplication factor K_{eff} within the acceptance criteria under all postulated accident conditions.

3.2.1.3 Dry Cask Loading Activities

TS 3.9.12.d with referenced Figure 3.9-1 and SR 4.9.12.d will be deleted. The limiting condition for operation and loading restriction were added to support dry cask loading activities. Section 50.68 of 10 CFR was revised to clarify that the criticality safety requirements under paragraph 50.68(b) which are applicable to spent fuel assemblies in the pools do not apply to spent fuel assemblies in storage or transportation casks placed in the pools under 10 CFR Part 71 or Part 72 approvals. The purpose of rulemaking was to draw the boundary between 10 CFR 50.68 and 10 CFR Part 71 or Part 72 with respect to spent fuel criticality safety requirements in the pool and in the casks within the pools. The amended 10 CFR 50.68, "Criticality accident requirements," allows the deletion of TS 3.9.12.d and SR 4.9.12.d.

3.2.1.4 Metamic™ Coupon Sampling Program

The proposed TS changes support a planned modification to the ANO-2 SFP that will utilize Metamic™ neutron absorber material in the new racks. The licensee committed to implement a coupon sampling program to confirm the capability of the Metamic™ material to perform its intended safety function in the SFP. The coupon sampling program will be incorporated as TS Section 6.5.17, Metamic™ Coupon Sampling Program. In addition, the licensee will replace SR 4.9.12.d with a new SR which will direct the performance of the coupon sampling program.

Metamic™ is a cermet composed primarily of B_4C and Aluminum (Al) 6061. A cermet is a composite material composed of ceramic and metallic materials. 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. With these corrosion resistant positive properties, Metamic™ is being introduced at a number of plants for SFP applications.

In its submittal dated March 30, 2007, the licensee has provided its Metamic™ Coupon Sampling Program which consists primarily of monitoring the physical properties of the absorber material by performing periodic neutron attenuation testing to confirm the physical properties.

3.2.1.4.1 Program Description

The purpose of the licensee's Metamic™ coupon surveillance program is to ensure that the physical and chemical properties of Metamic™ behave in a similar manner as that described in a vendor topical report on simulated performance testing of Metamic™. The coupon sampling program will monitor how the Metamic™ absorber material properties change over time under the radiation, chemical, and thermal environment found in SFP.

The coupons will be installed on a stainless steel coupon tree that holds 10 coupons. Each coupon is approximately 8 inches long, 4 inches wide, and 0.106 inches thick. Coupons are identical in composition and manufacturing process as the Metamic™ in the racks.

Each coupon will be mounted in stainless steel jackets simulating the actual insert design. The coupon tree will be placed in the SFP at the location where localized burn-up is greater than the assembly average burn-up. In addition, this location will accurately simulate the flow characteristics, pool chemistry, and differential metal interfaces that the Metamic™ racks will experience.

The coupon samples contain 30.5 wt% B_4C, which is consistent with the B_4C content in the Metamic™ used in the new spent fuel racks.

During a conference call on May 11, 2007, between NRC staff and ANO-2 staff, it was clarified that coupons will be examined on a 2-year basis for the first three operating intervals with the first coupon removed for inspection on or before October 31, 2009, and thereafter on 4 to 5-year intervals over the service life of the inserts. A different coupon will be removed each surveillance interval.

3.2.1.4.2 Monitoring Changes in the Physical Properties and Testing of Coupons

The licensee stated that when a coupon is removed in accordance with the sampling program, the following measurements will be performed:

1. Physical observations (visual and photography) to observe for physical indications on the surface to detect pitting, swelling, discoloration, or any other degradation. In addition, the licensee indicated that special attention will be paid to the development of any edge or corner defects.

2. Dimensional measurements of length, width, and thickness.

3. Weight and density measurements.

4. Neutron attenuation testing to confirm the Boron-10 (B_{10}) concentration or destructive chemical testing to determine the boron content.

The licensee's acceptance criteria for the dimensional and B_{10} measurements are as follows:

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

- Any change in the thickness of ±0.01 inches.

- Any change in weight and density of ±5 percent

- Any change in B_{10} content of 5 percent.

Prior to installing the coupons in the SFP, each coupon is pre-characterized. The physical characteristics presented above are documented for each coupon. Measurements on post-irradiated coupons will be made at the same location as the original measurements made on pre-irradiated coupons.

When a coupon is removed, measurements and physical observations will be recorded and evaluated for any physical or visual change when compared to the original data. If the measurements taken do not meet the established acceptance criteria, the licensee may perform an investigation and engineering evaluation which may include early retrieval and measurement of one or more of the remaining coupons to confirm the indicated change(s).

After all testing is finished, the coupons may be returned to the coupon tree, depending on whether the integrity of the coupon is compromised or contamination levels are too high.

The licensee's coupon measurement schedule is as follows:

Coupon #	Years of Exposure in SFP	Sampling Period (Years)
1	2	2
2	4	2
3	6	2
4	10	4
5	15	5
6	20	5
7	25	5
8	30	5
9	35	5
10	40	5

As shown in the above table, there is a sufficient number of coupons to last 40 years, which bounds the current operating license for ANO-2. The NRC staff concludes that the proposed surveillance program which includes visual, physical, and confirmatory tests will be capable of detecting potential degradation of the Metamic™ material that could impair its neutron absorption capability. The NRC staff concludes that the proposed TS 6.5.17 and SR 4.9.12.d for a new Metamic™ coupon sampling surveillance program will verify that the assumptions used in the SFP criticality analyses will remain valid and is, therefore, acceptable.

3.2.2 Thermal Hydraulic Analyses

The licensee performed a number of thermal-hydraulic analyses to demonstrate compliance of the new fuel racks, using methods similar to other comparable SFP storage projects. In each calculation, appropriate assumptions are made to ensure the conservatism of the results and are noted in the Holtec Licensing report (Attachment 8 to the LAR).

SFP decay-heat contributions were calculated as input for subsequent analyses. The decay heat calculation was performed using Holtec's quality assurance (QA)-validated computer DÉCOR, which incorporates the ORIGEN2 code. ORIGEN2 has previously been accepted by the NRC for use in similar decay-heat calculations for Virgil C. Summer Nuclear Power Station (safety evaluation dated August 30, 2002 (ADAMS Accession No. ML022330203), and the Clinton Power Station (safety evaluation dated October 31, 2005 (ADAMS Accession No. ML053070593). The decay-heat loads were calculated for a partial core offload of 92 fuel assemblies, and for a full core offload of 177 fuel assemblies. Both scenarios exceed the storage capability of the SFP for added conservatism, and the full-core offload is bounding.

SRP 9.1.3 and RG 1.13 emphasize the importance of retaining coolant inventory in the SFP during accident conditions. Minimum time-to-boil and maximum boil-off rates are calculated to verify that the new racks did not have an adverse effect on the ability to maintain SFP coolant inventory under accident conditions. For conservatism, a complete loss of forced cooling is assumed to occur coincident with the highest decay heat and maximum SFP bulk temperature. In the full-core offload scenario, the resulting minimum time-to-boil is 1.56 hours. This is representative of a worst-case scenario, and would still provide adequate time to initiate corrective measures before boiling began in the SFP. The maximum boil-off rate would be 88 gallons per minute (gpm), allowing substantial time to supply makeup water before the SFP level dropped far enough to compromise the shielding effect of the SFP coolant.

The maximum SFP local water and cladding temperature was calculated to verify that adequate natural circulation occurs throughout the fuel racks and ensures that nucleate boiling and film boiling do not occur within the assemblies, as stated in RG 1.13, position C.11, and SRP 9.1.2, review item III.2.1. The maximum local temperature analysis was performed using a computational fluid dynamics approach, and calculations were made using the Holtec QA-validated FLUENT program. FLUENT has previously been accepted by the NRC for use in similar computational fluid dynamics calculations for Virgil C. Summer Nuclear Power Station (safety evaluation dated August 30, 2002) and the Clinton Power Station (safety evaluation dated October 31, 2005). Assumptions are made to ensure conservatism, and the calculation method is designed to be bounding of all actual scenarios. Under normal operating conditions, the peak local water temperature was found to be 188 degrees Fahrenheit (°F) and the peak local cladding temperature was 218.3 °F. both substantially lower than the boiling temperature of water at the depth of the fuel assemblies (241 °F).

In an accident condition with complete loss of cooling water, the water temperature will rise over time until boiling begins. Peak local cladding temperature is calculated under this scenario to evaluate the performance of the rack under a worst-case bounding scenario. Assuming a temperature of 212 °F at the inlet to the fuel rack, and using the conservative assumptions of the previous maximum cladding temperature calculation, the maximum local cladding temperature will be 280 °F. While this temperature is above the boiling temperature of water, the total heat flux between the cladding and the water (2200 watts per cubic meter (W/m^3)) is far below the heat flux required for departure from nucleate boiling ($\sim10^6$ W/m^3). It is thereby determined that the integrity of the cladding is maintained even under this postulated accident scenario.

Based on the calculations provided by the licensee, the NRC staff finds the thermal-hydraulic considerations of this partial re-rack are acceptable. The new racks provide adequate natural circulation for the removal of heat during normal and accident conditions, and the design of the racks prevents nucleate and film boiling, in accordance with SRP 9.1.2 and RG 1.13. The worst-case effects of boiling in the SFP are not detrimental to cladding integrity, and the licensee has sufficient time to institute corrective measures and provide makeup water to the SFP prior to the onset of boiling.

3.2.3 Heavy Load Handling

All lifting and transferring of the racks during installation will be performed by the L3 fuel handling crane in conjunction with a special lifting device. The L3 crane is designed to be single failure-proof and, therefore, dropping of the racks during installation is not postulated.

The structural capabilities of the lifting device are designated to be acceptable for operation without redundant links between the rack and the L3 crane, in accordance with the guidance of ANSI N14.6-1993. Calculated stresses at any given section of the device are less than the minimum of one-fifth the material ultimate strength or one-third the material yield strength. Dynamic effects were also taken into account. Additionally, a factor of two was used for the reduction in allowable strengths. These design features provide assurance of the integrity of the special lifting device during the proposed lifts.

Prior to load testing, all welds will be 100 percent quality control (QC)-inspected. Load testing is 300 percent of rated load to be held for 10 minutes. Post testing (PT) will consist of QC visually inspecting the lifting devices for signs of deformation, and PT of all applicable load bearing welds.

The licensee has stated that during removal and installation of the racks in Region 1 of the ANO-2 SFP, all work will be controlled and performed in strict accordance with the written guidance of NUREG-0612. The licensee has stated that safe load paths will be determined to ensure that any drop will not impact irradiated fuel in the SFP or any safe shutdown equipment. These load paths will be defined in written procedures which should be followed during all lifting activities.

The NRC staff finds that the licensee has provided adequate assurance that its planned actions for the handling of heavy loads during the partial re-rack of the SFP will be consistent with the approach described in NUREG-0612. Therefore, the NRC staff concludes that the use of the L3 crane in conjunction with the described special lifting device is in conformance with NUREG-0612, specifically Sections 5.1.2(1) and 5.1.6, and is acceptable.

3.2.4 Structural and Seismic Analysis

The response of a freestanding rack module to seismic loadings involves combinations of rocking, twisting, turning, and sliding. This could potentially cause impacts between the fuel assemblies and the cell walls, and between modules, and between the modules and the pool walls. The rack pedestals are restricted from the lateral motion only by friction with the pool floor. In addition, there are large fluid coupling effects due to the water around the assemblies and the adjacent structures. An appropriate simulation of the response of the rack modules and components can only be obtained by direct time integration of the non-linear equations of motion, with three-directional pool slab seismic time-histories applied as forcing functions acting on the structures simultaneously.

Design requirements, as set forth in the Final Safety Analyses Report (FSAR) and applicable provisions of the SRP, for the SFP storage racks and its components, are provided in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, subsection NF (Reference 7), and those for the SFP structure are provided in American Concrete Institute (ACI) Standards ACI-318, "Building Code Requirements for Structural Concrete" (Reference 8), and ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures" (Reference 9). In addition, NRC letter dated April 14, 1978, with Enclosure 1, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, with addendum dated January 18, 1979 (Reference 10), provides additional guidance for design of spent fuel storage racks.

The structural/seismic analysis of the SFP racks is described by the licensee in Attachment 6 to the LAR. The hydraulic and dynamic interaction of the closely spaced racks was simulated in the analysis by including all modules in one comprehensive simulation using a Whole Pool Multi-Rack analysis. The analyses considered the non-linear behavior of the freestanding rack structures with gaps between various components including hydro-dynamic interaction from pool water. All rack modules were modeled simultaneously, including the coupling effect due to the multi-body motion. The models for the individual racks were developed from single-rack analyses for rack modules in both Regions 1 and 2 of the ANOD-2 SFP.

The models developed consist of beam elements to model the rack and fuel elements. The rack material data was obtained from the ASME Code, Section III, Appendices (Reference 11). Spring/gap elements and contact surfaces were used to account for the racks being un-anchored, and for possible impacts between the racks and the pool walls, rack-to-rack impacts, and fuel element-to-cell wall impact. Mass elements were used to include the added mass to account for hydrodynamic effects. These were calculated in accordance with Lawrence Livermore Laboratory Report UCRL Report 52342 (Reference 12). The methodology used is consistent with industry practice as stated in NUREG/CR-5912, "Review of the Technical Basis and Verification of Current Analysis Methods Used to Protect Seismic Response of Spent Fuel Storage Racks" (Reference 13), for analysis of spent fuel racks, shielding blocks, and dry fuel casks. The analyses were performed using the public domain finite element analysis program SOLVIA, which has the capability to perform general non-linear finite element analysis and contains non-linear "gapped truss" elements to model the stiffness of the gaps between the structural elements. The impact loading effects were implicitly included in the model using these gap elements that open and close during analysis.

In the analyses, synthetic time-histories in three orthogonal directions were generated to simulate the design-basis earthquake (DBE) and the operating-basis earthquake (OBE), in accordance with SRP Section 3.7.1, "Seismic Design Parameters." The OBE and DBE were developed to correspond to 2 percent damped horizontal OBE and DBE target plant response spectra. The 2 percent spectra were chosen since this is the damping used in the analysis of weld structures, which applies to the structures being reviewed in this amendment. The time-histories were generated to satisfy the enveloping criterion for synthetic time-histories and the requirements for statistical independence in SRP 3.7.1.

The licensee performed a detailed structural safety evaluation of the spent fuel storage replacement and existing rack configuration, and the SFP structure. The evaluation considered the loads from dead weight (including rack assembly and Metamic™ panel weights), the OBE and DBE, thermal loads resulting from normal operating or shutdown, and postulated abnormal design conditions, to determine the margin of safety and the structural integrity of the fuel racks, the SFP, and the Metamic™ panels. The loads, load combinations, and structural acceptance criteria used in the analyses were based on the ASME Code, Section III, subsection NF, SRPs 3.8.4 and 3.8.5, and the "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 10).

The analytical results indicate that the maximum seismic displacements under DBE conditions do not pose any potential for impact between the top of the racks and the pool walls or between racks. The rack-to-rack or rack-to-wall gap elements did not close during the analysis simulations (i.e., no impacts were recorded between the racks, and between the racks and the pool walls for all analysis cases).

The stresses in the rack modules were evaluated from full three-dimensional (3-D) rack models using the ANSYS computer program. The resultant member and weld stresses in the racks were all shown to be below the ASME Code-allowable stresses. In addition, the limiting maximum combined rack stress interaction coefficient for axial and bending stresses under the OBE and DBE load combinations was shown to be 0.878 <1.0 allowable. Likewise, the limiting maximum combined rack stress interaction coefficient for any of the welds under the OBE or DBE load combinations was shown to be 0.75 <1.0 allowable.

The licensee also performed local stress evaluations due to impact between the fuel assembly and cell wall, and demonstrated that the stresses are considerably lower than the allowable stresses.

The licensee also performed postulated accident drop analyses. These analyses consisted of 3-D models of the cells, using the public domain program LS-DYNA, subjected to applicable mechanical loads resulting from the accidental conditions. LS-DYNA has a large deformation capability. Three types of accidents were postulated:

- A shallow fuel assembly and gate drop accident. The licensee's stated acceptance criterion for this accident was the allowable depth of a cell that could be crushed during impact without affecting the criticality of the structure. This depth was determined as 31.8 inches.

- A deep drop fuel assembly through the central cell (which affects the base plate). The licensee's acceptance criterion was stated as the impact will not lead to a gross failure or excessive deflection of the base plate.

- A deep drop through the cell located above a pedestal. The licensee's stated acceptance of criterion was that the liner will remain intact or exhibit small plastic deformation, and that the concrete structural integrity will not be affected.

The criterion for the deep fuel assembly drop accident was exceeded, and therefore, required a criticality analysis. The licensee stated that all criteria of this analysis were satisfied and all other criteria for the other accidents were also satisfied. Thus, the licensee stated that the results of the analyses indicate that the functionality of the racks would not be affected by any of the postulated accident drop conditions.

The licensee also evaluated the structural integrity of the SFP for increased loads from the spent fuel racks. A review of the pool structure was performed using the 1981-1982 analysis by the licensee in support of the re-rack project for ANO-2 with the applied loads including the rack load effects. These effects were amplified using conservatively determined factors to account for the increased loads from the racks. Specifically, the dead weight loading of the racks was factored up by the ratio of the maximum increase for any of the racks. The seismic load combination (which consisted of combined seismic effects for the pool structure, the water, plus the rack seismic loads) was in general recalculated by factoring the seismic rack loads by the maximum ratio calculated for the worst-case rack in either the horizontal or vertical directions.

The results of this review demonstrated that the increased loads from the racks had minimal effects on the pool structural elements, and that the structural integrity of the pool structure was maintained.

The licensee determined that the results of the analyses performed for the fuel storage racks including the Metamic™ panels were in compliance with applicable acceptance criteria, and these structural components would be able to maintain their intended safety functions and structural integrity when subject to pertinent design-basis loads.

The NRC staff has evaluated the material presented by the licensee in Attachment 6 to its LAR including analysis methods, loading and load combinations, the acceptance criteria, and the results of the analysis as summarized in this safety evaluation. To complete its review, the NRC staff requested additional information (RAI) from the licensee regarding the seismic/structural analysis presented by the licensee. By letter dated June 13, 2007, the licensee provided its responses to the RAI.

In Questions 1 and 2, the NRC staff requested clarification regarding the geometry of the Metamic™ racks, the weights of the Metamic™ panels and the method of attachment of the panels to the cell walls. In its response, the licensee provided a (proprietary) Holtec drawing of the SFP racks, showing the Region 1 and Region 2 rack layouts. The licensee also described the method of construction of the Metamic™ racks, and provided a (proprietary) Holtec drawing showing the details of the rack construction and the method of attaching the Metamic™ panels to the cell walls. The licensee stated that the Metamic™ panel weight is approximately 3628 pounds (lbs) in a 9-cell x 9-cell rack, and 3225 lbs in the 9-cell x 8-cell rack. The Metamic™ panels are encased in the sheathing welded along the cell walls. The NRC staff found the clarification acceptable, because it clarified the modeling of the structures used in the licensee's seismic/structural analysis.

In Question 3, the NRC staff requested analytical or experimental verification for the non-linear analysis method using gap elements in the computer program SOLVIA, or to provide a reference where the staff had reviewed and approved this program. In its response, the licensee stated that the program SOLVIA was used on the ANO-1 SFP structural analysis, similar to the one described in this LAR for ANO-2. The NRC staff reviewed and accepted the use of SOLVIA in its safety evaluation of the ANO-1 LAR for SFP modifications dated January 26, 2007 (ADAMS Accession No. ML070160038) (Reference 4), as corrected by letter dated February 26, 2007 (ADAMS Accession No. ML070440038) (Reference 5). The NRC staff, therefore, finds that it has reviewed and approved the use of this program.

In Question 4, the NRC staff requested a detailed description of the method used for calculating the mass and stiffness of the gap elements used in the program SOLVIA. In its response, the licensee stated that the purpose of the gap elements is to account for gaps and clearances between the fuel assemblies and the inside wall of the cells, clearances between the racks, and clearances between the racks and the SFP walls. The individual gap stiffness is zero when the gap is open, and becomes effective when the gap closes, signifying impact has occurred. No mass was applied to the gap elements themselves. The mass at the gaps was either lumped at the appropriate points or included by the use of the mass matrix elements. The stiffness of the gap elements was in general determined by modeling the local parts' affected components and structures. Stiffness coefficients were obtained from these local models by loading with unit forces or displacements statically at the potential points of impact. The licensee stated that for seismic loading, the velocities at impact and the resulting strain rates would be low and, therefore, the stiffness obtained by this methodology is considered appropriate. The NRC staff agrees with the licensee that this methodology is appropriate for determining gap stiffness because it is a commonly used procedure in non-linear structural analysis.

In Question 5, the NRC staff indicated that the acceptance criterion specified that the compression allowable loads for the DBE load combination applicable to the overall rack structure is limited to two-thirds of the critical buckling load, where the critical buckling load was identified as the Euler buckling load. The Euler equation is applicable to long columns and is valid as long as the calculated critical buckling stress does not exceed one-half times the yield stress of the column material. The NRC staff requested that the licensee provide the critical buckling or provide justification for not doing so. In its response, the licensee agreed that the buckling criterion referenced in the acceptance criteria applies to “long” columns subjected to elastic buckling only. The licensee stated that it was included from the reference criteria basis but not actually used for the rack analysis. The licensee also stated that the racks as constructed do not exhibit an overall buckling mode. The racks consist mostly of plate and shell-type elements. The licensee further stated that local buckling of the cell wall panels could be possible; however, from the magnitude of the calculated compressive stresses, localized buckling was not indicated and, therefore, was not explicitly checked.

Based on its review, the NRC staff finds that the licensee’s analysis method was adequate and acceptable because it was consistent with the positions presented in NUREG/CR-5912.

3.2.4.1 Structural and Seismic Analyses Conclusion

The NRC staff reviewed the evaluation methodology and results presented by the licensee for structural qualification of the spent fuel Metamic™ racks and the SFP structure due to the modifications proposed in the LAR. The NRC staff concludes that the licensee has addressed the structural issues associated with the LAR by appropriately considering additional loading on the SFP racks, adequately simulating in the analysis the non-linear dynamic behavior of the racks under seismic loading, choosing loading combinations that comply with the "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," and the SRPs 3.8.4 and 3.8.5, using acceptance criteria as defined in Subsection NF of the ASME code, and demonstrating that all components of the rack assembly remain within acceptable limits. This conclusion is based on the fact that the evaluation methodology adopted by the licensee conforms to the applicable requirements of the ANO-2 FSAR, and the loading combinations and acceptance criteria in accordance with the NRC-approved guidance and relevant ASME Code sections.

The NRC staff also concludes that the licensee has performed an adequate evaluation of the SFP structure to account for the effects of using the Metamic™ racks in the SFP. This conclusion is based on the fact that the licensee has appropriately reviewed the existing design calculations of the structure, and demonstrated that even with very conservative estimates of increased loading from the racks using loading combinations in accordance with SRPs 3.8.4 and 3.8.5. Therefore, there is reasonable assurance that the structural integrity of the SFP structure will be maintained when subjected to applicable design-basis loads.

Based on the above evaluation, the NRC staff concludes that, with respect to the seismic design adequacy and the structural integrity evaluation, the licensee has demonstrated compliance with GDCs 2 and 4. The NRC staff has concluded there is adequate and acceptable technical justification in support of its proposed TS change to support a partial re-rack of the ANO-2 SFP.

3.2.5 Dose Analysis

The NRC staff has reviewed the licensee's analysis of the proposed action on design-basis accident (DBA) dose analyses. The licensee stated that the radiological consequences of accidents in the fuel building or containment are not affected by the replacemnet of the spent fuel racks or increase in fuel enrichment. The NRC staff evaluated this statement with respect to the radiological consequences of DBAs for operation after the new fuel storage racks have been installed and also during the installation of the racks in the SFP.

During removal and installation of the fuel storage racks in the SFP, the licensee will ensure that all work in the SFP area will be controlled and performed in strict accordance with specific written guidance. Any movement of fuel assemblies required to support removal and installation of racks will be performed as during normal refueling operations, and no shipping cask movement will be performed during this time frame. The licensee will determine and follow safe load paths and written procedures to ensure that no racks are carried over any portions of the existing fuel storage racks containing fuel assemblies. With the proposed limitations on rack and cask movement, there would be no release of radioactive material from damaged spent fuel and no radiological consequences due to fuel rack installation.

The proposed TS changes support a planned modification to the ANO-2 SFP that will replace the Region 1 fuel racks with new racks containing Metamic™ neutron poison panels, but does not make any change to the number of fuel assemblies stored in the region. The fuel assembly U-235 enrichment will be changed to allow storage of fuel assemblies with an initial U-235 enrichment of 4.95 wt%. The average fuel burnup value is unchanged and limited by ANO-2's operating license to 60 megawatt-days/kilograms of uranium. The current licensing bases for the dose consequences associated with a fuel handling accident and occupational exposure were performed assuming a maximum U-235 enrichment of 5.0 wt% and a maximum burnup of 65 megawatt-days/kilograms of uranium. As these assumptions are unchanged from the current licensing bases analyses and the number of fuel assemblies is also unchanged, the NRC staff concludes that the current dose analyses bounds the changes proposed in this amendment and the current licensing basis for the dose analyses is unchanged for this proposed amendment.

3.2.6 <u>Occupational Radiation Exposure</u>

The licensee plans to replace the three Boraflex racks in the SFP with three new Metamic™ racks. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the dose to perform the fuel rack installation will be less than 1 person roentgen equivalent man (rem). This estimate includes the radioactive waste processing of the existing contaminated racks, as well as the projected dose to divers.

As noted in Section 3.4.5 the partial rerack of the SFP does not increase the number of fuel assemblies that can be stored in the racks of the ANO-2 SFP, change the allowable fuel enrichment previously analyzed, or change the average fuel burnup limit, which is restricted by the ANO-2 operating license. As such, the impact on dose rates associated with the new rack design is minimal. The dose rate at the surface of the pool from direct radiation is typically not affected unless the allowable burnup of the fuel being stored has increased. The allowable burnup of the fuel being stored has not increased. Increasing storage capacity or adding racks to an area of the pool previously unoccupied can have an effect by changing the local dose rate above the pool. The storage capacity of the SFP is unchanged and the new replacement racks will occupy approximately the same area of the SFP. Additionally, the critical parameters of the new replacement racks (i.e., center-to-center spacing (pitch) and cell inner diameter) are preserved in the new design.

Dose rates above the pool from radionuclides in the water are typically governed by the cleanup systems associated with the pool and not the number of assemblies in the pool or the type of racks. There are no changes proposed to the spent fuel purification system. The proposed modification does not impact the pool water level that provides shielding to ensure that the pool surface dose rate remains below 5 millirem per hour (mrem/hr). The dose rate from a fuel assembly in transit is therefore unchanged. The calculated gamma dose rate through the wall of the SFP is a result of the radioactive decay from the fuel. Current calculations very conservatively estimate this dose rate to be 5.5 mrem/hr at 100 hours cooling in the most active fuel region. The actual dose rates through the wall have been demonstrated to be lower than this value and are determined periodically, or as needed. Stay times are appropriately determined, if necessary. If a general area dose rate is found to be higher than expected, the site can re-categorize an area/room as needed, including updates to the radiation maps in the FSAR, if required.

The NRC staff concludes that the ANO-2 SFP re-racking operations can be performed in a manner that will ensure that doses to workers will be maintained ALARA (as low as is reasonable achievable). The NRC staff concludes that the projected dose for the project of less than 1 person rem is well within the range of doses for similar modifications at other nuclear plants, and is, therefore, acceptable.

3.3 Conclusion on Proposed Changes to ANO-2 TSs

Based on the NRC staff's review of the 1) criticality safety analyses, 2) boron dilution, 3) 10 CFR 50.68, "Criticality accident requirements, 4) Metamic™ coupon sampling program, 5) thermal-hydraulic considerations of the new racks, 6) heavy load handling, 7) structural and seismic analyses, 8) dose analysis, and 9) occupational radiation exposure, the NRC staff concludes that the proposed use of Metamic™ and the proposed TS changes to support the use of Metamic™ in the SFP are acceptable for ANO-2.

In addition, in Section 3.7, Fuel Type, of the Entergy Analysis of Proposed Technical Specification Change, it is stated that, "no fuel types other than Combustion Engineering or Westinghouse 16 x 16 fuel assemblies, including Next Generation Fuel (NGF), will be stored at the ANO-2 SFP." The NRC staff requested that the term, NGF fuel, be defined explicitly. Entergy replied that Westinghouse NGF as described in WCAP-16500 will be used, and that the subtle changes to the fuel design from those defined in the WCAP may be made as needed. The NRC staff finds this response acceptable, because the licensee stated in Section 3.7 of the submittal that, "If different fuel types are used in the future, changes to the fuel assembly design, key fuel assembly mechanical features, and the changes in operating strategy will be evaluated under 10 CFR 50.59, 'Changes, tests and experiments'."

4.0 LIST OF REGULATORY COMMITMENTS

The licensee made the following list of regulatory commitments with respect to its LAR. These commitments are identified in Attachment 7 to its application.

1. The surveillance coupons will be approximately 4" x 8" and 0.106" thick, identical in composition and manufacturing process as the Metamic™ in the inserts (i.e., created from the same manufacturing lot used to manufacture the Metamic™ PIAs).

2. The coupons will be mounted in stainless steel jackets simulating the actual insert design.

3. The coupon tree will have ten coupons.

4. The coupon tree will be installed within a cell in Region 2.

5. The coupons will be staggered and placed adjacent to the active fuel region where, based on the burnup profile, the localized burnup is greater than the assembly average burnup.

6. No welding will be used on the Metamic™ as per the PIA design.

7. Scratches will be simulated by the mechanical etching or scribing the surface of the coupons. The scratches will be formed using hardened materials made out of carbon steel, stainless steel, and Metamic™. The scratches will not be cleaned

after being applied to ensure an evaluation will be performed of the corrosion affects of leaving the trace material in a scratch.

8. Coupons will be examined on a two year basis for the first three intervals and thereafter on a 4 to 5 year interval over the service life of the inserts.

9. During the first six years, freshly discharged fuel assemblies will be placed on two sides of the coupon tree to ensure that the dose to the coupons is maximized.

10. Upon receipt of a coupon for testing, the exposed coupon should be carefully examined and photographed to document the appearance of the coupon, noting any sign of degradation that may be observed. Special attention will be paid to any edge or corner defects and to any discoloration, swelling, or surface pitting that might exist.

11. Measurements to be performed at each inspection will be as follows:

    A. Physical observations of the surface appearance to detect pitting, swelling or other degradation,

    B. Length, width, and thickness measurements to monitor for bulging and swelling (Measurements will be taken in five procedurally defined locations prior to placing the coupons in the ANO-2 SFP. When the coupon is removed, measurements will be taken in the same locations as the original measurements.) Length and width dimensions shall not exceed $\pm$ 0.125 inches when compared to the initial width or length. Thickness is used to monitor swelling and an increase in thickness at any point shall not exceed $\pm$ 0.01 inches of the initial thickness at that point.

    C. Weight and density to monitor for material loss (the weight of each coupon should be obtained within $\pm$ 5% of the initial coupon weight), and

    D. Neutron attenuation to confirm the B_{10} concentration or destructive chemical testing to determine the boron content.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The state official has 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. 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 published May 8, 2007 (72 FR 26175). 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 to 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) 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

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2. Letter from Entergy to the U.S. NRC, "Use of Metamic™ In Fuel Pool Applications (0CAN080201)," August 8, 2002.

3. Letter from U.S. NRC, "Arkansas Nuclear One, Units 1 and 2 – Safety Evaluation for Holtec Report HI-2022871 Regarding Use of Metamic™ in Fuel Pool Applications" (TAC Nos. MB5862 and MB5863), July 17, 2003.

4. Letter from F.E. Saba, U.S. NRC, to J.S. Forbes, Entergy Operations, Inc., "Arkansas Nuclear One, Unit 1 – Issuance of Amendment for Use of Metamic™ Poison Insert Assemblies in the Spent Fuel Pool," January 26, 2007.

5. Letter from F. E. Saba, NRC, to T.G. Mitchell, Entergy Operations, Inc., "Arkansas Nuclear One, Unit 1 – Correction to Amendment No. 228 for the Use of Metamic™ Poison Insert Assemblies in the Spent Fuel Pool," February 26, 2007.

6. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," 1981.

7. ASME Boiler and Pressure Vessel Code, Section III, subsection NF, 1980 through Winter 1981 Addendum.

8. ACI-318, "Building Code Requirements for Structural Concrete," American Concrete Institute.

9. ACE-349, "Code Requirements for Nuclear Safety Related Concrete Structures."

10. Letter from B.K. Grimes, U.S. NRC, to Power Reactor Licensees dated April 14, 1978, with Enclosure 1, “OT Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications,” April 14, 1978, and Addendum, January 19, 1979.

11. ASME Code, Section III, Appendices, through Winter 1981 Addendum.

12. UCRL Report 52342, “Effective Mass and Damping for Submerged Structures,” April 1, 1978.

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16. J. F. Briesmeister, Editor, “MCNP – A General Monte Carlo N-Particle Transport Code, Version 4A,” LA12625, Los Alamos National Laboratory, 1993.

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