January 26, 2007

Mr. Jeffrey S. Forbes
Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT FOR USE OF METAMIC® POISON INSERT ASSEMBLIES IN THE SPENT FUEL POOL (TAC NO. MD2674)

Dear Mr. Forbes:

The Commission has issued the enclosed Amendment No. 228 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to Entergy's application dated July 27, 2006, as supplemented by Entergy's letters dated October 4, October 9 (proprietary), and December 14, 2006.

Specifically, the proposed amendment would revise the ANO-1 TS 3.7.14, "Spent Fuel Pool Boron Concentration," TS 3.7.15, "Spent Fuel Pool Storage," and the associated Figure 3.7.15-1, and TS 4.3, "Fuel Storage," and the associated Figure 4.3.1.2-1. In addition, this amendment would add TS 5.5.17, "Metamic Coupon Sampling Program," and Surveillance Requirement 3.7.15.2, which directs the performance of the coupon sampling program. The proposed TS changes support a modification to the ANO-1 spent fuel pool (SFP) that would utilize Metamic® poison insert assemblies. In addition to the proposed plant modification, the licensee would increase the SFP boron concentration and credit boron to ensure that a 5-percent subcriticality margin is maintained during normal and accident conditions. This proposed amendment also would increase the allowable initial fuel assembly uranium-235 (U-235) enrichment from 4.1 weight percent (wt%) to a maximum U-235 enrichment of 4.95 wt%.

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/

Farideh E. Saba, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 228 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

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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/
Farideh E. Saba, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 228 to DPR-51
2. Safety Evaluation

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Package: ML070160040 *See October 25, 2006, and January 17, 2007, SE input.
**See January 17, 2007, SE input.
Accession No.: ML070160038 TSs and License Page: ML070240605

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/SNPB	NRR/EGCA
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OFFICE	NRR/CSGB	NRR/ITSB	OGC	NRR/LPL4/BC
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DATE	12/15/06	1/19/07	1/23/07	1/26/07

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ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 228
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated July 27, 2006, as supplemented by letters dated October 4, October 9 (proprietary), and December 14, 2006, 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. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 228, are hereby incorporated in the renewed license. EOI 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Facility Operating License No. DPR-51 and the Technical Specifications

Date of Issuance: January 26, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 228

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

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

Remove	Insert
3	3

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

Remove	Insert
3.7.14.1	3.7.14.1
3.7.15-1	3.7.15-1
3.7.15-2 (Figure 3.7.15-1)	3.7.15-2 (Table 3.7.15-1)
------	3.7.15-3 (Table 3.7.15-1 continued)
4.0-3	4.0-3
4.0-5	4.0-5
------	4.0-6
5.0-25	5.0-25
------	5.0-25a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 228 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated July 27, 2006 (Reference 1), and supplemented by letters dated October 4 (Reference 2), October 9 (Reference 3 - proprietary), and December 14, 2006 (Reference 4), Entergy Operations, Inc. (Entergy) submitted, for the Nuclear Regulatory Commission (NRC) staff review and approval, a license amendment to the Arkansas Nuclear One, Unit No. 1 (ANO-1) Renewed Facility Operating License and Technical Specifications (TSs). The proposed TS changes support a planned modification to the ANO-1 spent fuel pool (SFP) that will utilize Metamic® poison insert assemblies (PIAs). In addition to the proposed plant modification, the licensee would increase the SFP boron concentration and credit boron to ensure that a 5-percent subcriticality margin is maintained during normal and accident conditions. This proposed amendment also would increase the allowable initial fuel assembly uranium-235 (U-235) enrichment from 4.1 weight percent (wt%) to a maximum U-235 enrichment of 4.95 wt%.

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.

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. According to GDC 62, "Prevention of criticality in fuel storage and handling" (Reference 5), the licensee must prevent criticality in the fuel handling and storage system by physical systems or processes, preferably by the use of geometrically safe configurations.

Section 50.68 of 10 CFR, "Criticality accident requirements" (Reference 6), provides NRC regulatory requirements for maintaining subcritical conditions in the SFP in lieu of meeting the requirements of 10 CFR 70.24 for radiation monitoring. ANO-1 is currently exempt from the requirements of 10 CFR 70.24, "Criticality accident requirements." The exemption was granted on October 6, 1998 (Reference 7).

As set forth in 10 CFR 50.68(b), the acceptance criteria for prevention of criticality in the spent fuel storage racks as they apply to ANO-1 are as follows:

50.68(b)(1) 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.

50.68(b)(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

50.68(b)(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and /or design features prevent such moderation or if fresh fuel storage racks are not used.

50.68(b)(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

50.68(b)(5) The quantity of SNM [special nuclear material], other than nuclear fuel stored on site, is less than the quantity necessary for a critical mass.

50.68(b)(6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

50.68(b)(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

50.68(b)(8) The FSAR [Final Safety Analysis Report] is amended no later than the next update which 50.71(e) of this part requires, indicating that the licensee has chosen to comply with 50.68(b).

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

1. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 9.1.2, "Spent Fuel Storage," Draft Revision 4 (Reference 8),

2. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" (Reference 9), 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 10).

According to the NUREG-0800, Section 9.1.2, "Spent Fuel Storage," the review should ensure 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.

The scope of analysis and design of SFP structure and its components affected by the license amendment request (LAR) is required to meet the requirements of 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena," and Criterion 4, "Environmental and dynamic effects design bases," to demonstrate that structural adequacy of the fuel racks and the SFP structure is maintained. NUREG-0800 provides guidance for performing design and analysis to demonstrate compliance with the regulation. Design requirements for the SFP storage racks and its components are provided in the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NF, and those for the SFP structure are provided in the American Concrete Institute (ACI)-318, "Building Code Requirements for Structural Concrete," and ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures." In addition, "USNRC OT [Office of Technology] Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 1, 1978, provides additional guidance for design of spent fuel storage racks.

3.0 TECHNICAL EVALUATION

3.1 Proposed Change

By letter dated July 27, 2006, Entergy, the licensee for ANO-1, requested a license amendment to revise TS Sections 3.7.14, "Spent Fuel Pool Boron Concentration," 3.7.15, "Spent Fuel Pool Storage," and the associated Figure 3.7.15-1, TS Section 4.3, "Fuel Storage," and the associated Figure 4.3.1.2-1. In addition, this amendment would add Surveillance Requirement (SR) 3.7.15.2 to direct the performance of the coupon sampling program, which will be reflected in proposed TS 5.5.17, "Metamic Coupon Sampling Program." These TS revisions lead to the following specific changes at ANO-1:

- Allow insertion of Metamic® PIAs into the flux traps of a newly defined Region 3 within the current Region 2 of the ANO-1 SFP.

- Redefine the loading pattern in the current Region 1 racks, taking no credit for Boraflex®.
- Redefine the loading pattern in the remaining Region 2 racks.
- Modify the SFP boron concentration from ≥ 1,600 parts per million (ppm) to > 2,000 ppm.
- Modify the applicability of TS 3.7.14 to specify that the TS is applicable any time fuel assemblies are stored in the SFP regardless of whether an SFP verification has been performed or not.
- Allow an increase in the maximum fuel assembly U-235 enrichment from the current U-235 enrichment of 4.1 wt% to a maximum of 4.95 wt%.
- Redefine storage patterns in the new fuel storage racks.
- Add a Metamic® coupon sampling program (SR 3.7.15.2 and TS 5.5.17)

The proposed license amendment would remove reliance on Boraflex® as a neutron absorber in Region 1 of the SFP. To preclude the continued loss of reactivity margin due to the ongoing degradation of Boraflex®, the neutron absorbing function currently performed by Boraflex® would be replaced by new loading restrictions for Region 1; these would also replace the existing Boraflex surveillance program. The Region 2 racks do not contain fixed poison assemblies. The proposed change to Region 2 would modify a portion of the SFP storage racks in Region 2 by the insertion of Metamic® PIAs. The area with the Metamic® PIAs would be defined as Region 3. Loading restrictions would be applied to the remaining Region 2 racks and also to the racks in the newly defined Region 3. Furthermore, an increase in the SFP boron concentration is proposed for boron credit to assure a 5-percent subcriticality margin would be maintained during normal and accident conditions.

To accommodate future reload plans, Entergy proposed to increase the allowable initial fuel assembly U-235 enrichment from 4.1 wt% to a maximum U-235 enrichment of 4.95 wt%. To this end, criticality analyses were performed based on the higher enrichment for the SFP racks as well as the new fuel storage racks. New loading patterns are defined for the new fuel storage racks. The proposed changes include a coupon sampling program to monitor the potential changes in the characteristics of Metamic®.

By letter dated October 4, 2006, Entergy informed the NRC that Holtec International had identified an error in the criticality safety evaluation that was included in the original submittal. The error was associated with the script that transfers the material compositions between the depletion code CASMO and the criticality code MCNP4a. The error affected the oxygen atom density, and, thereby, affected the atom densities of all actinides and fission products in the MCNP4a criticality calculation by about 3 percent. This resulted in a slight underprediction of the k_{eff} in the analysis, thereby reducing the margin.

The error is documented in the ANO corrective action program. The licensee has also submitted to the NRC the relevant corrected pages for insertion into the original submittal of

July 27, 2006. These corrections resulted in a slightly higher minimum burnup requirement for the fuel assemblies that are stored in Region 1 and Region 2 of the ANO SFP. In addition, the required minimum boron concentration to ensure that k_{eff} remains $\leq$ 0.95 is slightly lower than the previously submitted value.

The supplement also made minor weld changes that resulted in slight wording changes in the summary of the structural considerations, and corrected a typographical error. Two more supplements dated October 9 and December 14, 2006, were submitted in response to staff requests for drawings of the poison insert assemblies, and to the staff request for additional information (RAI) related to the Metamic® coupon sampling program and explanation of a few specific areas of the structural/seismic analysis, respectively.

3.2 Evaluation of TS Changes

3.2.1 Criticality Safety and Thermal-Hydraulic Analyses Evaluation

Evaluated Changes to the ANO -1 TSs

A. TS 3.7.14, "Spent Fuel Pool Boron Concentration"

The requested amendment proposed to increase the requirement for the minimum boron concentration to greater that 2,000 ppm. This proposed increase in the boron concentration provides a sufficient margin that assures the maximum neutron multiplication factor, k-effective (k_{eff}), will remain below 0.95 in the unlikely event of a criticality accident. The upper limit on SFP boron concentration is 3,500 ppm per ANO-1 Final Safety Analysis Report (FSAR) Section 9.6.2.4.3.4. 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.

The proposed change also modifies the applicability of TS 3.7.14 to require the designated boron concentration any time fuel assemblies are stored in the SFP, regardless of whether SFP verification has been performed.

B. TS 3.7.15, "Spent Fuel Pool Storage"

Region 1 of the SFP contains Boraflex® poison panels, and currently there are no loading restrictions required by TS for Region 1. The proposed change no longer credits Boraflex® in Region 1, and meets the regulatory requirements with regard to k_{eff} through loading restrictions based on minimum burnup requirements at varying initial U-235 enrichment and cooling times for Regions 1. This results in the creation of a new Table 3.7.15-1 based on new SFP criticality analysis and the concomitant deletion of Figure 3.7.15-1.

Currently, ANO-1 TS 3.7.15 and Figure 3.7.15-1 define loading restrictions for fuel assemblies that are stored in Region 2 of the ANO-1 SFP. Under the proposed changes a portion of the SFP racks in Region 2 are modified by the installation of Metamic® PIAs. This portion of Region 2 is redefined as Region 3 and subject to new loading restrictions for Region 2 specified in Table 3.7.15-1 and two further restrictions: Unrestricted storage is allowed for fuel

assemblies with an initial U-235 enrichment less than or equal to 4.35 wt%; for fuel assemblies with an initial U-235 enrichment greater than 4.35 wt%, the burnup of at least one fuel assembly in each 2 X 2 section of the storage cells is at least 20.1 Giga Watt-Day/metric ton of Uranium.

The remaining SFP racks in Region 2 will continue to be referred to as Region 2 and subject to the new loading restrictions for Region 2 defined in Table 3.7.15-1 with regard to minimum burnup requirements at varying initial U-235 enrichment and cooling time. In addition, rack interface requirements have been evaluated and are included in Table 3.7.15-1.

The proposed TS change applies the new loading restrictions to any fuel assembly that is stored in the SFP and is applicable whenever a fuel assembly is stored in the SFP. The action is modified to require a nonconforming fuel assembly to be placed in an acceptable storage location in accordance with the appropriate loading restrictions. To this end, SR 3.7.15.1 is modified to reflect its applicability to the parameters defined in Table 3.7.15-1. The parameters associated with the fuel assembly must be satisfied prior to storing a fuel assembly in the SFP, as the loading restrictions now apply throughout the pool.

C. TS 4.3.1, “Criticality”

TS 4.3.1.1 addresses the design and constraints of the spent fuel storage racks with regard to criticality as:

- TS 4.3.1.1 a allows fuel assemblies with a maximum U-235 enrichment of 4.95 wt%.

- TS 4.3.1.1 b specifies that k_{eff} is to be maintained at less than or equal to 0.95 if the SFP racks are fully flooded with borated water, which includes an allowance for uncertainties. A criticality analysis demonstrated that a boron concentration of 444 ppm is sufficient to maintain a $k_{eff} \leq 0.95$ during normal operations.

- TS 4.3.1.1 c specifies that k_{eff} is to be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties.

- TS 4.3.1.1 d specifies a nominal 10.65 inch center-to-center distance between fuel assemblies placed in the storage racks.

- TS 4.3.1.1 e specifies that new or partially spent fuel assemblies be stored in accordance with the loading restrictions in Table 3.7.15-1 in the spent fuel storage racks.

- TS 4.3.1.1 f specifies that new or partially spent fuel assemblies with cooling times, U-235 enrichment or discharge burnup in the “unacceptable range” of Table 3.7.15-1 for fuel assemblies stored in Region 1 or Region 2 may be stored in a 2 X 2 checkerboard configuration (i.e., 2 assemblies and 2 empty cells).

- TS 4.3.1.1 g specifies that neutron absorber (Metamic®) be installed between fuel assemblies in the Region 3 racks.

TS 4.3.1.2 addresses the design and constraints of the new fuel storage racks with regard to criticality as:

- TS 4.3.1.2 a allows fuel assemblies with a maximum U-235 enrichment of 4.95 wt%.

- TS 4.3.1.2 b specifies that k_{eff} is to be maintained at less than or equal to 0.95 under normal conditions, which includes an allowance for uncertainties.

- TS 4.3.1.2 c specifies that k_{eff} is to be less than 0.98 with optimum moderation, which includes an allowance for uncertainties.

- TS 4.3.1.2 d specifies a nominal 21-inch center-to-center distance between fuel assemblies placed in the storage racks.

- TS 4.3.1.2 e specifies that fuel assembly loading is prohibited in the interior storage cells as shown in Figures 4.3.1.2-1 or 4.3.1.2-2, based on U-235 fuel enrichment.

Criticality Safety Analyses

The objective of the SFP criticality analysis is to insure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity. In addition, it is demonstrated 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.

A. The specific evaluation performed for the ANO-1 SFP are:

- The Region 1 racks are evaluated for storage of spent fuel assemblies with specific burnup requirements as a function of initial enrichment between 2.0 wt% and 5.0 wt% U-235 and decay times between 0 and 20 years for both of the cases without and with soluble boron credit.

- The Region 1 racks are evaluated for storage of fresh fuel assemblies with a maximum nominal enrichment of 5.0 wt% U-235 in a checkerboard configuration with empty storage cells for both of the cases without and with soluble boron credit.

- The Region 2 racks are evaluated for storage of spent fuel assemblies with specific burup requirements as a function of initial enrichment between 2.0 wt% and 5.0 wt% U-235 and decay times between 0 and 20 years for both of the cases without and with soluble boron credit.

- The Region 2 racks are evaluated for storage of fresh fuel assemblies with a maximum nominal enrichment of 5.0 wt% U-235 in a checkerboard configuration with empty storage cells for both of the cases without and with soluble boron credit.

- The Region 3 racks are evaluated for storage of fresh and spent fuel assemblies in a 3-of-4 configuration of three fresh fuel assemblies with a maximum nominal enrichment of 5.0 wt% U-235 and one spent fuel assembly with a maximum nominal initial enrichment of 5.0 wt% U-235 that has accumulated a minimum specified burnup and for both of the cases without and with soluble boron credit.

- The Region 3 racks are evaluated for unrestricted storage of fresh fuel assemblies with a maximum nominal enrichment of 4.35 wt% U-235 for both the case without and with soluble boron credit.

In addition, the reactivity effects of abnormal and accident conditions are evaluated. The most limiting accident condition involves placing a fresh fuel assembly, enriched to 5.0 wt% U-235, outside the storage rack, adjacent to other fuel assemblies in the rack. A minimum soluble boron concentration of 889 ppm must be maintained in the SFP to ensure that the maximum k_{eff} is less than 0.95 under accident conditions. The required soluble boron concentration of 889 ppm is well with the TS for minimum soluble boron concentration for ANO-1.

Similarly, the possibility of an increased reactivity effect due the rack interfaces within and between racks is 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 1 - Region 1, Region 2 - Region 2, Region 3 - Region 3, Region1 - Region 3, and Region 2 - Region 3. The calculated reactivity from the interface calculation is then compared to the calculated reactivity from reference infinite array calculations.

B. Computational Methodology

The principal method for the criticality analysis of the high-density storage racks is the three-dimensional Monte Carlo code MCNP4a (Reference 11). This code has been extensively verified and used for criticality analysis. For this analysis, MCNP4a calculations use continuous energy cross-section data based on ENDF/B-V and ENDF/B-VI. Benchmark calculations show a bias in k_{eff} of 0.0009 and an uncertainly of +/- 0.0011 evaluated with a 95-percent probability at the 95-percent confidence level.

Fuel depletion analyses during core operation are performed with the code CASMO-4 (Reference 12) (using a 70-group cross-section library), a two-dimensional multigroup 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 rack geometry, yielding the two-dimensional infinite multiplication factor (k_{inf}) for the storage rack to determine the 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-35 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:

- Moderator is borated or unborated water at a temperature in the operating range that results in the highest reactivity, as determined by analysis.
- Neutron absorption in minor structural members is neglected, i.e. , spacer grids are replaced by water.
- 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.
- The B_4C loading in the neutron absorber panels is nominally 25 wt%, with an uncertainty of ±0.5 wt%.
- The Axial Power Shaping Rods (APSRs) and Burnable Poison Rod Assemblies (BPRAs) are assumed to cover the entire active fuel length of the assembly during depletion. No credit is taken in the rack criticality calculations for the APSRs and the BPRAs.

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) using the following formula:

$$\text{Max } k_{eff} = \text{Calculated } k_{eff} + \text{biases} + [\sum_i (\text{Uncertainty}_i)^2]^{1/2}$$

<u>Boron Dilution Evaluation</u>

In order to conform to TS 3.7.14 the required minimum soluble boron concentration is 444 ppm under normal conditions. The soluble boron in the SFP water is conservatively analyzed to contain a minimum of 1,600 ppm under operating conditions. The volume of water in the pool is approximately 268,000 gallons. Thus, large amounts of unborated water would be necessary to reduce the boron concentration from 1,600 ppm to 444 ppm. The analyses assume that the unborated water flowing into the pool mixes instantaneously with the water in the pool.

A. Low-Flow Rate Dilution

Administrative controls require a measurement of the soluble boron concentration in the pool water at least weekly. In this time period, an undetected dilution flow rate of 33.7 gallons per minute (gpm) would be required to reduce the boron concentration to 444 ppm. No known dilution flow rate of this magnitude has been identified. Furthermore, a total of more than 333,000 gallons of unborated water would be associated with the dilution event and such a large flow of unborated water would be readily evident by high-level alarms and by visual inspection on daily walk-downs of the storage pool area.

B. High-Flow Rate Dilution

Under certain accident conditions, it is conceivable that a high flow rate of unborated water could flow onto the top of the pool. Such an accident scenario could result from rupture of an unborated water supply line or possibly the rupture of a fire protection system header, both events potentially allowing unborated water to spray onto the pool. Upon consideration of all related scenarios, a significant dilution of the pool soluble boron concentration in a short period of time without corrective action is not considered a credible event. It is not considered credible that multiple alarms would fail or be ignored or that the spilling of large volumes of water would not be observed. Therefore, such a major failure would be detected in sufficient time for corrective action to avoid violation of an administrative guideline and to assure that the health and safety of the public is protected.

Thermal-Hydraulic Considerations

The requirements for cooling capability of the SFP and its attendant cooling systems are set forth in the NRC Standard Review Plan (Reference 8) and USNRC OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications (Reference 13). The thermal-hydraulic qualification analyses for the modified rack fall into the following four categories:

1. Evaluation of bounding maximum decay heat versus time profiles.

2. Evaluation of loss-of-forced cooling scenarios, to establish the minimum times to perform corrective actions and the associated makeup water flow rate requirements.

3. Determination of the maximum local water temperature, at the instant when the pool decay heat reaches its maximum value, to establish that localized boiling in the spent fuel storage racks is not possible while forced cooling is operating. The bulk pool temperature is postulated to be at the maximum limit.

4. Evaluation of the maximum fuel rod cladding temperature, at the instant when the pool decay heat reaches its maximum value, to establish that nucleate boiling is not possible while forced cooling is operating. The bulk pool temperature is postulated to be at the maximum limit.

A. SFP Decay Heat Loads

The bounding maximum decay heat versus time profiles are based on calculating the total decay heat generation history in the SFP. In the SFP, the total decay heat comes from two different groups of assemblies:

1. Fuel assemblies from previous offloads already stored in the SFP.

2. Fuel assemblies that are offloaded from the reactor to the SFP.

The decay heat contributions of both the previously and recently offloaded fuel are determined using the Holtec QA validated computer program DECOR, which incorporates the Oak Ridge

National Laboratory code ORIGEN2. The use of ORIGEN2 has previously been accepted by the NRC for SFP decay heat calculations (Reference 15 and Reference 16).

The computed maximum decay heat versus time profiles are based on the following two offload scenarios:

1. Partial Core Offload - A refueling batch of 76 assemblies is offloaded from the plant's reactor into the SFP, completely filling all available storage locations. The total SFP inventory prior to the offload is 912 fuel assemblies, for a final post-offload inventory of 988 fuel assemblies. This slightly exceeds the storage capacity of the ANO-1 SFP (and the ANO-1 TS 4.3.3 limit of 968 assemblies) and is used for the calculation of conservative decay heat loads.

2. Full Core Offload - The full core of 177 assemblies is offloaded from the plant's reactor into the SFP, completely filling all available storage locations. The total SFP inventory prior to the offload is 836 fuel assemblies, for a final post-offload inventory of 1013 fuel assemblies. This slightly exceeds the storage capacity of the ANO-1 SFP (and the ANO-1 TS 4.3.3 limit of 968 assemblies) and is used for the calculation of conservative decay heat loads.

Given the conservative assumptions incorporated into the calculations, actual decay heat loads are lower than the calculated values.

B. Minimum Time-to-Boil and Maximum Boiloff Rate under conservative assumptions, such as:

- The initial SFP bulk temperature is equal to the bulk temperature limit of 150 °F for the full core offload and 120 °F for the partial core offload.

- The thermal inertia of the SFP is based on the net water volume only.

- During the loss of forced cooling makeup water is not available.

- The loss of forced cooling coincides with the peak SFP bulk temperature and the maximum pool decay heat.

The results of the loss-of-forced cooling evaluations give a minimum time-to-boil, maximum boiloff rate and minimum time for water to drop to top of racks respectively as 8.67 hours, 46.88 gallons per minute (gpm), and 62.1 hours for partial core offload, and 3.18 hours, 86.28 gpm, and 33.7 hours for a full core offload. Thus, in the unlikely event of a failure of forced cooling to the SFP, there are at least 3.18 hrs. available for corrective actions prior to SFP boiling. The maximum water boiloff rate is less than 87 gpm.

C. Maximum SFP Local Water Temperature

The upper bound on the maximum SFP local water temperature is computed under a series of conservative assumptions, such as:

- The walls and floor of the SFP are all modeled as adiabatic surfaces.
- Heat losses by thermal radiation and natural convection from the hot SFP surface to the environment are neglected.
- No downcomer flow is assumed to exist between the rack modules.
- The hydraulic resistance parameters for the rack cells, permeability and inertial resistance, are conservatively adjusted by 10 percent.
- The bottom plenum heights used in the model are less than the actual heights.
- The hydraulic resistance of every spent fuel storage rack (SFSR) cell includes the effects of blockage due to and assumed dropped fuel assembly lying horizontally on top of the SFSRs.

The results of Computational Fluid Dynamics (CFD) analysis with the commercially available CFD program FLUENT (Reference 17), which has been Benchmarked under Holtec's quality assurance program, gives a peak local water temperature of 168 °F and a peak local fuel cladding temperature of 199.5 °F.

Both the maximum local water temperature and the bounding fuel cladding temperature are substantially lower than the 240 °F local boiling temperature at the top of the SFSRs. Thus, boiling, including nucleate boiling on clad surfaces, cannot occur anywhere within the ANO-1 SFP.

D. Fuel Rod Cladding Temperature

Under a postulated accident scenario of the loss of all cooling, the water temperature will rise. Under the assumption of a temperature of 212 °F at the inlet to the rack cells, and conservative bulk-to-local and local-to-clad temperature differences, the maximum possible cladding temperature is 261.5 °F, which is greater than the saturation temperature at the top of the active fuel length. Due to the low maximum fuel assembly heat flux (approximately 7300 W/m^2) and the critical heat flux required for departure from nucleate boiling (on the order of 10^6 W/m^2), it is concluded that the fuel cladding will not be subjected to departure from nucleate boiling even under the postulated accident scenario of the loss of all SFP cooling and the cladding integrity would be maintained.

Conclusion

Based on the considerations discussed above, the NRC staff concludes that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the ANO-1 power plant with the above described revision to its TSs, and that the

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

The NRC staff has reviewed the licensee's supplements to the original license amendment request. Based on the supporting information submitted to the staff with regard to the corrections made to the relevant computations in the original submittal, the staff finds the changes to the TSs for ANO-1 acceptable with regard to the supporting criticality analyses and thermal-hydraulic analyses.

3.2.2 Coupon Sampling Program Evaluation

Metamic® 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 (B_4C) and metallic (Al) 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. In spite of these corrosion resistant positive properties, Metamic® has not been previously used in SFP applications.

In its submittal dated July 27, 2006, the licensee has provided a 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. By letter dated December 14, 2006, Entergy submitted its response to the staff RAI related to the Metamic® coupon sampling program

Program Description

The purpose of the licensee's Metamic® coupon surveillance program is to ensure the physical and chemical properties of Metamic® behave in a similar manner as that found at the test facilities. The coupon program will monitor how the Metamic® absorber material properties change over time under the radiation, chemical, and thermal environment found in the SFP.

The coupon program will be incorporated in TS 5.5.17, "Metamic Coupon Sampling Program." In addition, the licensee will create a new SR 3.7.15.2, which will direct the performance of the sampling program.

The coupons will be installed on a stainless steel coupon tree that holds 10 or more coupons. Each coupon is approximately 7-inch long, 5-inch wide, and 0.10-inch thick. Coupons are identical in composition and manufacturing process as the Metamic® in the PIAs. Each coupon will be mounted in stainless steel jackets simulating the actual insert design. The coupon tree will be placed in the SFP at a location where localized burn-up is greater than assembly average burn-up. In addition, this location will accurately simulate the flow characteristics, pool chemistry, and differential metal interfaces that the Metamic® PIAs will experience. The coupon samples contain 25 percent B_4C, which is consistent with the B_4C content in the Metamic® used in the new spent fuel racks.

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 February 2009 and thereafter at increasing intervals over the service life of the inserts.

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 observation and photography:

    a. The licensee will observe for physical indications on the surface to detect pitting, swelling, discoloration or any other degradation.

2. Dimensional measurements:

    a. Length
    b. Width
    c. Thickness

3. Weight and density

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

The NRC staff reviewed the information provided in the licensee's letter dated July 27, 2006, and asked the licensee to provide the acceptance criteria for the above parameters and the basis for this criteria. In its response dated December 14, 2006, the licensee stated that the acceptance criteria is based on criteria from existing coupon sampling programs or reasonable limits that assure further evaluation. Regarding the criteria for visual examination, the licensee stated that in addition to physical observation and photography, special attention will be paid to development of any edge or corner defects.

Furthermore, the licensee's acceptance criteria for dimensional measurements and B_{10} 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).

The licensee also stated that regardless of whether the acceptance criteria are met, neutron attenuation testing will be performed on any coupon removed. By performing neutron attenuation testing, the licensee will be able to validate the B_{10} loading in the Metamic® panels and coupons. After all testing is finished, the coupons might 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 #	Duration in SFP (Years)	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
11	Spare	At Any Time
12	Spare	At Any Time

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-1. Since the last two coupons are not needed, they will be removed only if additional testing is required.

Conclusion

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

3.2.3 Seismic and Structural Evaluation

Evaluated Changes to the ANO-1 Technical Specification

By letter dated July 27, 2006 (Reference 1), Entergy submitted an LAR to amend the TS of Facility Operating License No. DPR-51 for ANO-1, in accordance with 10 CFR 50.90. The proposed change revises the following Sections of the TS:

- TS 3.7.14
- TS 3.7.15
- TS 4.3
- SR 3.7.15.2 will be added to direct the performance of the coupon sampling program, which will be reflected in proposed TS 5.5.17

Subsequently, by letter dated October 4, 2006 (Reference 2), Entergy submitted supplement to the LAR to correct an error in the original LAR that had slightly under predicted the k-effective in the analysis. The supplement also made minor weld changes that resulted slight wording changes in the summary of the structural considerations, and corrected a typographical error. Two more supplements dated October 9 (Reference 3) and December 14, 2006 (Reference 4), were submitted in response to staff requests for drawings of the PIAs and explanation of a few specific areas of the structural/seismic analysis, respectively.

The LAR, along with the supplements described above, proposes changes to the ANO-1 TS to support a planned modification to the SFP that will utilize Metamic® PIAs. In addition, Entergy proposes to increase the SFP boron concentration and credit boron to assure that a 5-percent subcriticality margin is maintained during normal and accident conditions. Also, to accommodate future reload plans, Entergy proposes to increase the allowable initial fuel assembly U-235 enrichment from 4.1 wt% to a maximum U-235 enrichment of 4.95 wt%.

The changes to TS 3.7.15 proposed by Entergy in the LAR will modify two of the Region 2 storage racks of the ANO-1 SFP to allow insertion of Metamic® PIAs into the flux traps of the racks. The modified racks will be designated as Region 3 storage racks. This modification will introduce additional loads in the SFP racks and the SFP structure. Therefore, structural integrity of the Metamic® panels, the SFP racks and the SFP structure must be evaluated to ensure that the fuel storage and handling system continue to meet the requirements of 10 CFR Part 50, Appendix A, GDC 61, “Fuel storage and handling and radioactivity control.” Changes to other sections of the TS do not affect structural/seismic considerations for design of SFP racks or the SFP structure.

Sections 5.6, “Structural/Seismic Analysis,” and 5.7, “SFP Structural Integrity for Increased Loads from SFP Racks,” of Attachment 1 to the LAR, Entergy described the structural and seismic evaluations performed for the LAR. Attachment 6 to the LAR provided a summary report of the detailed calculations performed to assess design adequacy of the Metamic® PIAs, the spent fuel storage racks and the SFP structure.

Summary of Entergy's Seismic and Structural Evaluation Results

The ANO-1 SFP contains eight independent rack structures designed to hold the spent fuel assemblies and rod cluster control assemblies in storage for long-term decay. There are three regions of racks. The Region 1 racks employ Boraflex® as the neutron absorbing (poison) material. Region 2 racks do not have any poison material. Region 3 racks are Region 2 racks that are being proposed to be modified by inserting Metamic® poison material strips into the flux traps of the cells. The racks are free standing on 14 feet that rest on the bottom of the pool. The eight racks are self-supporting, and are not connected to each other or to the SFP walls. There are two Region 1 racks, four Region 2 racks and two proposed Region 3 racks (that are modified Region 2 racks).

Entergy performed detailed and complete evaluation of the spent fuel storage racks and the SFP structure to address the structural issues resulting from the proposed modification to the SFP. The evaluation considered the loads from dead-weight-induced loads including rack, fuel assembly and poison insert weights, operating and design-basis earthquake (DBE), and thermal loads including 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 poison inserts. The loads, load combinations, and acceptance criteria used in the analyses were based on the ASME Code, Section III, Subsection NF (Reference 18), and USNRC OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications.

Entergy performed both whole pool multi-rack and single rack analyses for ANO-1 racks. The analysis considered the non-linear behavior of the free-standing rack structure with gaps between various components including hydraulic interaction from pool water. Seismic analyses were based on the simulation of the safe shutdown earthquake and the operating basis earthquake (OBE) in accordance with SRP Section 3.7.1, "Seismic Design Parameters," requirements. Impact loading effects were implicitly included in the model using gap elements that open and close during analysis. The results indicated that the maximum seismic displacements do not pose any potential of 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. The resultant member and weld stresses in the racks are all below the allowable stresses, with a largest interaction ratio of 0.82 for cell-to-cell welds during OBE loading case. Entergy determined that racks will remain functional during and after an OBE and DBE. In addition to the seismic evaluation discussed above, the storage racks were also analyzed for all postulated accident conditions. The results of the analysis indicated that the functionality of the racks would not be affected by any postulated accident conditions. Applicable mechanical loads under accident conditions were also included in the analysis. Entergy determined that the results of the analyses performed for the fuel storage racks including the Metamic® inserts 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.

Entergy also evaluated 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 Entergy in support for the re-rack project for ANO-1 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.

Staff Evaluation

The objective of the NRC staff's structural/seismic review was to determine both the adequacy and the acceptability of the seismic and structural design evaluation provided in the LAR and its attachments. The staff reviewed the engineering analyses submitted by Entergy including:

1. Technical information presented in Sections 5.6, "Structural/Seismic Analysis" and 5.7, "SFP Structural Integrity for Increased Loads from SFP Racks" of the LAR, and

2. Attachment 6 to the LAR, "Structural/Seismic Considerations for Addition of Metamic® Panels to the Flux Traps of Two Spent Fuel Racks at ANO-1".

Review of Sections 5.6 and 5.7 of Attachment 1 to the LAR

In Section 5.6, Entergy stated that a structural analysis of the spent fuel rack with the new poison panel inserts was considered for all loading configurations. The analysis evaluated normal, seismic, and accident conditions. Entergy also stated that the PIA's design was evaluated for normal and seismic conditions. The evaluations demonstrated margins of safety in all cases.

In Section 5.7, Entergy stated that an evaluation of the SFP structural integrity for the effects of the increased loads from the SFP racks was performed, and the evaluation demonstrated that the structural integrity of the pool structure was maintained.

The above Sections 5.6 and 5.7 did not provide any details of the analyses performed. The staff concludes that the evaluation methodology described in the above Sections of the LAR complies with the regulatory requirements and guidance mentioned in Section 2.0 of this safety evaluation, and is acceptable.

Review of Attachment 6 to the LAR

Attachment 6 to the LAR, "Structural/Seismic Considerations for Addition of Metamic® Panels to the Flux Traps of Two Spent Fuel Racks at ANO-1," provides a summary report of the analysis performed by Entergy to evaluate structural adequacy of the fuel storage racks and the SFP structure. The report described in detail rack configuration, material properties, structural analysis methodology, rack model development, loads and load combinations, acceptance criteria, and analysis results.

The staff evaluated the material presented in Attachment 6 including analysis methods, loading and load combinations, the acceptance criteria and the results of the analysis performed by Entergy as summarized above. During the evaluation the staff had requested some additional information from Entergy for explanation of certain Sections of Attachment 6 to the LAR. In its letter to the NRC dated December 14, 2006, Entergy responded to the RAIs. The following paragraphs present the staff's evaluation of the licensee's response.

Evaluation of Entergy Response to RAIs

Referring to the last paragraph of Page 6 of 50 of Attachment 6 of the LAR, the staff requested that the licensee discuss the fluid-structure analysis study performed and the results of the study that form the basis for Entergy's assertion that use of SOLVIA in seismic response modeling of the racks, fuel assemblies and inserts was adequate.

Entergy responded that the fluid-structure study consisted of modeling a 3D square structure (a box on feet) representative in general of one rack using shell elements. This structure model was effectively contained in a pool with fluid elements surrounding it. For comparison, the same methodology as used for the rack analysis was used to develop a beam model of the study structure, with hydrodynamic added mass effects also considered the same as for the rack analysis. The fluid/structure model and the associated beam model were analyzed using the same time-history input. Displacement results were comparable and in good agreement. Hence, it was concluded the beam model representation of the racks for the dynamic analysis provided comparable results for consideration of a more detailed model including fluid elements.

The NRC staff finds the above Entergy's response acceptable because adequate amount of comparative analyses considering both the shell element and beam element representations of the rack were performed with acceptable results obtained to demonstrate the adequacy of the modeling adopted and the Entergy's use of the SOLVIA code.

The NRC staff requested the licensee to discuss the key non-linear attributes that were modeled in the Whole Pool Multi-Rack (WPMR) analysis that reflect the interactions between the modules, and between a module and its adjacent spent fuel wall, including any experimental results to validate the use of Gapped-Truss elements in the SOLVIA code.

Entergy responded that non-linear attributes in the WPMR include gaps or clearances between the racks and between the racks and the pool, the free-sliding and lift-off potential for the racks relative to their support on the pool floor, and the accounting for potential impact effects. Experimental verification was not implemented. Methodology used is consistent with industry practice (Reference NUREG/CR-5912, "Review of the Technical Basis and Verification of Current Analysis Methods Used to Predict Seismic Response of Spent Fuel Storage Racks") for analysis of spent fuel racks, shielding blocks, and dry fuel casks. Use of the non-linear gap element (identified as a Gapped Truss element type) in SOLVIA provided a means to account for the gaps between the model components and impact forces if those gaps closed during the analysis. The "gapped truss" element is a compression-only element when the gap is closed, and transmits no loads when the gap is open.

The staff finds that Entergy's analysis method was adequate and acceptable because it was consistent with the positions presented in NUREG/CR-5912.

Referring to Section 3.8, "Poison Insert Analysis Methodology," (Page 9 of 50 of Attachment 6 of the LAR), the NRC staff requested that the licensee discuss the basis for treating the poison inserts as additional beam elements in the structural model, specifically, the appropriateness of the elastic beam element treatment of the Metamic® inserts accounting for its material characteristics (e.g., material brittleness), rack specific geometric layout, (e.g., gaps between cell walls and the inserts), constraints, and potential differences between the stress-strain relationship of the Metamic® inserts and that of the stainless steel cells.

Entergy responded that the poison inserts were originally designed to be wedged in the flux traps. With redesign of the inserts, nominal gaps or clearances within the flux traps were possible and likely to exist. Because of potential for the inserts to now "rattle" within the flux traps during a seismic loading event, including them explicitly in the model was an appropriate way to obtain these load effects.

The licensee further stated that the Metamic® inserts were modeled based on the full composite of all the components of the inserts. This included the stainless steel wrapper channels, the Metamic® panels, the channel shaped bands which hold the inserts in the wrapper channels, and the stiffener plates which hold the two wrapper channel sections together. While the Metamic® panels were modeled using their material properties, this was done only for completeness since the stiffness (structural) contribution to the overall properties of the assembled inserts is small. Because the Metamic® panels are held in the wrapper channels by the bands without any significant clamping force, no shear transfer occurs between the Metamic® panels and the wrapper channels and hence no composite action occurs. The wrapper channel assemblies control the structural behavior (they are much stronger and stiffer) with no significant contribution from the Metamic® panels. Additionally, the yield stress of the stainless steel wrapper plates is about 70 percent of that for the Metamic® panels. Since the wrapper plate assemblies control the displacement and deflections of the inserts including the Metamic® panels, and because they did not yield, the Metamic® panels are effectively protected by the wrapper panel assemblies and their relatively small ductility range (brittleness) is not a concern. Stress-Strain differences are appropriately considered by use of the proper modulus of elasticity for the different materials since everything remained elastic. Hence, consideration of Metamic® insert assembly as a beam within the flux traps with the gaps modeled is appropriate.

The NRC staff finds that the above response acceptable because it adequately describes the Metamic® insert assembly's seismic modeling as well as its expected composite response to the design-basis seismic input motion.

Section 8.10 of Attachment 6 of the LAR, "Comparison of Analysis Results to Westinghouse Results," states that this comparison is a further validation of the Stevenson and Associates [S&A] evaluation and that the use of Westinghouse results for the Wrapper welds, cell seam weld and cell-to-cell weld is justified. The NRC staff requested that Entergy explain any potential issues that may arise from Entergy's use of the Westinghouse results for the Wrapper welds, cell seam weld, and cell-to-cell weld, and their implication on the Region 3 rack seismic response.

Entergy responded that the Region 3 racks are Westinghouse racks, and the Westinghouse analysis is the basis for their original qualification. Because the S&A results show that the addition of the Metamic® inserts effectively did not change the seismic behavior and response of the racks, the rack welds as qualified by Westinghouse are not subjected to additional forces due to the addition of the Metamic® inserts to the racks, and hence the Westinghouse analysis and qualification of the welds remains valid. The Metamic® insert components and welds were analyzed and qualified in the present calculation using the results of the present analysis.

Since the Metamic® insert components and welds were qualified in the present calculation using the results of the present analysis, and meet the acceptance criteria per ASME Code, Section III, Subsection NF requirements, the NRC staff finds the response acceptable.

Referring to the second paragraph of Page 44 of 50 of Attachment 6 of the LAR, the NRC staff requested Entergy to explain and justify with pertinent references for the method used by Holtec in specifying a conservative hydrodynamic pressure resulting from the seismic displacement of the racks.

Entergy responded that the reference to Holtec's analysis in this paragraph was relative to the initial submittal, and should have been updated to reference the S&A analysis performed subsequent to Holtec's.

The pool structure analysis was updated using load results from the updated rack analyses done by S&A. For consideration of hydrodynamic pressure loads from the rack movements on the walls, it was observed in the fluid/structure that the effective pressure on the pool walls during seismic loading was less than or equal to that for the pool water alone. Original design/qualification of the pool and the re-analysis performed for the re-racking in 1982 both considered the effect of the water due to seismic inertial effects and the magnitude of this loading effectively covered pressure differences potentially caused by the movement of the racks.

The NRC staff finds the above response adequate and acceptable because the original design/qualification of the pool and the re-analysis performed for the re-racking in 1982 both considered the effect of the water due to seismic inertial effects and the magnitude of this loading effectively enveloped pressure differences potentially caused by the movement of the racks.

Conclusion

Based on the above detailed evaluation, the NRC staff concludes that, with respect to the seismic design adequacy and the structural integrity evaluation, Entergy has provided adequate and acceptable technical justification in support of its proposed technical specification change to implement the use of Metamic® poison insert assemblies into the flux traps of ANO-1 SFP racks. Therefore, the NRC staff finds the seismic and structural design evaluation of the ANO-1 spent fuel pool storage racks and the SFP structure to be acceptable.

4.0 REGULATORY COMMITMENT

The following table identifies those actions committed to by Entergy in conjunction with this amendment request in Attachment 7 of the July 27, 2006, letter:

Commitment	Type		Scheduled Completion Date
	One-Time Action	Continuing Compliance	
The surveillance coupons will be approximately 7" x 5" and 0.100" 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).		X	
The coupons will be mounted in stainless steel jackets simulating the actual insert design.		X	
The coupon tree will have ten or more coupons.		X	
The coupon tree will be installed within a flux trap in Region 2.		X	
The coupons will be staggered and placed adjacent to the active fuel region where, based on the bumup profile, the localized burnup is greater than the assembly average bumup.		X	
No welding will be used on the Metamic® as per the PIA design.		X	
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.		X	
Coupons will be examined on a 2-year basis for the first three intervals and thereafter on a 4 to 5 year interval over the service life of the inserts.		X	
During the first 6 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.		X	

Commitment	Type		Scheduled Completion Date
	One-Time Action	Continuing Compliance	
Measurements to be performed at each inspection will be as follows: • Physical observations of the surface appearance to detect pitting, swelling or other degradation, • 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-1 SFP. When the coupon is removed, measurements will be taken in the same locations as the original measurements.) • Weight and density to monitor for material loss, and • Neutron attenuation to confirm the B10 concentration or destructive chemical testing to determine the boron content.		X	
The ANO-1 FSAR will be amended no later than the next required update after the proposed TS change is approved and implemented. This FSAR update will indicate that ANO-1 has chosen to comply with 10 CFR 50.68(b).	X		

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are provided by the licensee's administrative processes, including its commitment management program. Should the licensee choose to incorporate a regulatory commitment into the emergency plan, FSAR, or other documents with established regulatory controls, the associated regulations would define the appropriate change-control and reporting requirements. The NRC staff has determined that the commitments do not warrant the creation of regulatory requirements, which would require prior NRC approval of subsequent changes. 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 (see Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000). The commitments should be controlled in accordance with industry guidance or comparable criteria employed by a specific licensee. The NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration in its July 27, 2006, amendment request. The NRC staff reviewed the licensee's analysis and, based on its review, it appeared that the three standards of 10 CFR 50.92(c) were satisfied. Therefore, the NRC staff proposed to determine that the amendment request involves no significant hazards consideration, and published its proposed determination in the *Federal Register* for public comment on December 26, 2006 (71 FR 77414):

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Fuel Handling Accidents

   The current licensing bases for the dose consequences associate with a fuel handling accident (FHA), which was performed considering a maximum U-235 enrichment of 4.95 wt% and a maximum burnup of 60,000 megawatt-days/ton of uranium, does not exceed 25% of 10 CFR 100 limits. The proposed change does not impact the current analysis and therefore, there is no increase in the dose consequences associated with a[n] FHA.

   The probability of having a[n] FHA has not increased. Although it could be postulated that a Metamic® panel could be dropped during installation, the approximate 50 pound weight of the panel falling on the racks is bounded by the current fuel assembly drop analysis.

   Criticality Accidents associated with a Dropped Fuel Assembly

   The three fuel assembly drop accidents described below can be postulated to increase reactivity. However, for these accident conditions, the double contingency principle of ANS[I] [American National Standards Institute] N-16.1-1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since its absence would be a second unlikely event.

Three types of drop accidents have been considered: a vertical drop accident, a horizontal drop accident, and an inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. The structural damage to the pool liner, the racks, and fuel assembly resulting from a dropped fuel assembly striking the rack, the pool floor, or another assembly located in the racks is primarily dependent on the mass of the falling object and drop height. Since these two parameters are not changed by the proposed modification, the postulated structural damage to these items remains unchanged. In all cases the proposed TS limit for boron concentration ensures that a five percent subcriticality margin is met for the postulated accidents.

Criticality Accidents associated with a Misplaced Fuel Assembly

The fuel assembly misplacement accident was considered for all storage configurations. An assembly with high reactivity is assumed to be placed in a storage location which requires restricted storage based on initial U-235 loading, cooling time, and burnup. The presence of boron in the pool water assumed in the analysis has been shown to offset the worst case reactivity effect of a misplaced fuel assembly for any configuration. This boron requirement is less than the boron concentration required by the ANO-1 TS. Thus, a five percent subcriticality margin is met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

Optimum Moderation Accident

For fuel storage applications in the SFP, water is usually present. An "optimum moderation" accident is not a concern in SFP storage racks because the rack design prevents the preferential reduction of water density between the cells of a rack (e.g., boiling between cells). In addition, the criticality analysis has demonstrated that k_{eff} [k-effective] will remain less than 1.0 when the SFP is fully flooded with unborated water.

An "optimum moderation" accident in the new fuel vault was evaluated and the conclusions of that evaluation confirmed that the reactivity effect is less than the regulatory limit of 0.98 for k_{eff}.

Loss of SFP Cooling

The proposed changes to the ANO-1 SFP racks do not result in changes to the SFP cooling system and therefore the probability of a loss of SFP cooling is not increased.

The consequences of a loss of spent fuel pool cooling were evaluated and found to not involve a significant increase as a result of the proposed changes. A thermal-hydraulic evaluation for the loss of SFP cooling was performed. The analysis determined that the minimum time to boil is more than three hours following a complete loss of forced cooling. This provides sufficient time for the operators to restore cooling or establish an alternate means of cooling before the

water shielding above the top of the racks falls below 10 feet. Therefore, the proposed change represents no increase in the consequences of loss of pool cooling.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The presence of soluble boron in the pool water assumed in the criticality analysis is less than the boron concentration required by the ANO-1 TSs. Thus, a five percent subcriticality margin is met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

No new or different types of fuel assembly drop scenarios are created by the proposed change. During the installation of the Metamic® panels, the possible drop of a panel is bounded by the current fuel assembly drop analysis. No new or different fuel assembly misplacement accidents will be created. Administrative controls currently exist to assist in assuring fuel misplacement does not occur.

No changes are proposed to the spent fuel pool cooling system or makeup systems and therefore no new accidents are considered related to the loss of cooling or makeup capability.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

With the presence of a nominal boron concentration, the SFP storage racks will be designed to assure a subcritical array with a five percent subcritical margin (95% probability at the 95% confidence level). This has been verified by criticality analyses.

Credit for soluble boron in the SFP water is permitted under accident conditions. The proposed modification that will allow insertion of Metamic® poison panels does not result in the potential of any new misplacement scenarios. Criticality analyses have been performed to determine the required boron concentration that would ensure the maximum k_{eff} does not exceed 0.95. The ANO-1 TS for the minimum SFP boron concentration is greater than that required to ensure k_{eff} does not exceed 0.95. Therefore, the margin of safety defined by taking credit for soluble boron will be maintained.

> The structural analysis of the spent fuel racks along with the evaluation of the SFP structure indicated that the integrity of these structures will be maintained with the addition of the PIAs. The structural requirements were shown to be satisfied, thus the safety margins were maintained.
>
> In addition the proposed change includes a coupon sampling program that will monitor the physical properties of the Metamic® absorber material. The monitoring program provides a method of verifying that the assumptions used in the SFP criticality analyses remain valid.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The supplemental letter dated December 14, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 26, 2006 (71 FR 77414).

On the basis of the above analyses, on which there has been no public comment during the 30-day comment period, the NRC staff concludes that the proposed amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

6.0 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 had no comments.

7.0 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 NRC 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 December 26, 2006 (71 FR 77414). 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.

8.0 CONCLUSION

The NRC 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.

9.0 REFERENCES

1. Letter from Jeffery S. Forbes (Entergy) to USNRC, "License Amendment Request to Support the Use of Metamic® Poison Insert Assemblies in the Spent Fuel Pool, Arkansas Nuclear One, Unit 1 (ANO-1)," dated July 27, 2006 (Agencywide Documents Access and Management System (ADAMS) ML062220440).

2. Letter from Thomas A. Marlow, Entergy, to NRC, "Supplement to License Amendment Request to Support the Use of Metamic® Poison Insert Assemblies in the Spent Fuel Pool, Arkansas Nuclear One, Unit 1 (ANO-1)", dated October 4, 2006 (ADAMS No. ML062910244).

3. Letter from Thomas A. Marlow, Entergy, to NRC, "Supplemental Information to License Amendment Request to Support the Use of Metamic® Poison Insert Assemblies in the Spent Fuel Pool, Arkansas Nuclear One, Unit 1 (ANO-1)", dated October 9, 2006 (ADAMS No. ML062910229).

4. Letter from John R. Eichenberger, Entergy, to NRC, "Supplement to License Amendment Request to Support the Use of Metamic® Poison Insert Assemblies in the Spent Fuel Pool, Arkansas Nuclear One, Unit 1", dated December 14, 2006 (ADAMS No. ML063610336).

5. Part 50 of 10 CFR, Appendix A, GDC 62, "Prevention of criticality in fuel storage and handling."

6. Section 50.68 of 10 CFR, "Criticality accident requirements."

7. Letter from William Reckley (NRC) to C. Randy Hutchinson (Entergy), "Issuance of Exemption from the Requirements of 10 CFR 70.24 for Arkansas Nuclear One, Units 1 and 2 (ANO-1, ANO-2)," October 6, 1998 (TAC Nos. MA1278 and MA1279).

8. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 9.1.2, "Spent fuel Storage," Draft Revision 4, April 1996.

9. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," December 1981.

10. 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," dated August 19, 1998.

11. J. F. Briesmeister, Editor, “MCNP - A general Monte Carlo N-Particle Transport Code, Version 4A,” LA-12625, Los Alamos National Laboratory (1993).

12. M. Edenius, K. Ekberg, B. H. Forssen, and D. Knott, “CASMO-4 A Fuel Assembly Burnup Program User’s Manual,” Studsvik/SOA-95/1, Studsvik of America, Inc.

13. USNRC OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications, 14 April 1978.

14. Letter from Kahtan N. Jabour (USNRC) to Christopher M. Crane (AmerGen), “Clinton Power Station, Unit 1 - Issuance of an Amendment - Re: Onsite Spent Fuel Storage Expansion (TAC No. MC4202),” October 31, 2005

15. Letter from Karen R. Cotton (USNRC) to Stephen A. Byrne (South Carolina Electric & Gas Company), “Virgil C. Summer Nuclear Station, Unit No. 1 - Issuance of Amendment - Re: Spent Fuel Pool Expansion (TAC No. MB2475),” August 30, 2002

16. USNRC Standard Review Plan 9.1.3, “Spent Fuel Pool Cooling and Cleanup System,” Revision 1, July 1981.

17. Holtec Report HI-2032998, “Validation of FLUENT Version 6.1.18", Revision 0.

18. ASME Boiler & Pressure Vessel Code, Section III, Subsection NF, 1980, through Winter 1981 Addendum.

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