April 10, 2006

Mr. R. T. Ridenoure
Vice President - Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: (TAC NO. MC8860)

Dear Mr. Ridenoure:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 239 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to Omaha Public Power District's (OPPD's) application dated November 8, 2005, as supplemented by letters dated March 17 and 27, 2006. In the March 17, 2006, letter, OPPD committed to be in full compliance with Section 50.68 of Title 10 of the *Code of Federal Regulations* 6 months after the fall 2006 refueling outage is complete.

The amendment adds limits and controls for the spent fuel cask loading and unloading operations in the spent fuel pool (SFP). The change modifies the TSs by adding a new Limiting Condition for Operation (LCO) 2.8.3(6) that establishes (1) a boron concentration requirement during cask loading operations in the SFP, and (2) a spent fuel burnup-initial enrichment limit in the spent fuel cask to ensure subcritical conditions are maintained during spent fuel cask loading operations in the SFP. In addition, the change modifies TS Tables 3-4 and 3-5, and adds a new subsection 4.3.1.3 in Design Features 4.3.1 to describe the spent fuel cask design features. The change also conforms pagination.

A copy of the 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. 50-285

Enclosures: 1. Amendment No. 239 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 239
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee) dated November 8, 2005, as supplemented by letters dated March 17 and 27, 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, Renewed Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 239 , are hereby incorporated in the 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 30 days 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 Technical Specifications

Date of Issuance: April 10, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 239

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

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

REMOVE	INSERT
TOC - Page 8	TOC - Page 8
2.8 - Page 14	2.8 - Page 14
2.8 - Page 15	2.8 - Page 15
2.8 - Page 16	2.8 - Page 16
2.8 - Page 17	2.8 - Page 17
2.8 - Page 18	2.8 - Page 18
2.8 - Page 19	2.8 - Page 19
2.8 - Page 20	2.8 - Page 20
2.8 - Page 21	2.8 - Page 21
2.8 - Page 22	2.8 - Page 22
2.8 - Page 23	2.8 - Page 23
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------------	2.8 - Page 25
------------	2.8 - Page 26
------------	2.8 - Page 27
3.2 - Page 5	3.2 - Page 5
3.2 - Page 12	3.2 - Page 12
4.0 - Page 2	4.0 - Page 2
4.0 - Page 3	4.0 - Page 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 239 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

In March 2005, the Nuclear Regulatory Commission (NRC) issued Regulatory Issue Summary (RIS) 2005-05, “Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations,” to (1) alert addressees at pressurized-water reactor (PWR) facilities to findings suggesting that the spent fuel pool (SFP) licensing and design bases and applicable regulatory requirements may not be met during loading, unloading, and handling of dry casks in the SFPs; (2) emphasize the importance of maintaining subcritical conditions for spent fuel storage in moderated environments; and (3) encourage addressees to review the current SFP and independent spent fuel storage installation (ISFSI) licensing and design bases at their facilities to ensure compliance with regulations during dry cask loading, unloading, and handling operations.

As a result of the RIS, Omaha Public Power District (OPPD/the licensee) submitted an application dated November 8, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060930053), as supplemented by letters dated March 17 and 27, 2006 (ADAMS Accession Nos. ML060790081 and ML060870168, respectively), requesting changes to the Technical Specifications (Appendix A to Renewed Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1 (FCS). In the March 17, 2006, letter, OPPD committed to be in full compliance with Section 50.68 of Title 10 of the *Code of Federal Regulations* (10 CFR) 6 months after the fall 2006 refueling outage is completed.

The proposed amendment would revise FCS Technical Specifications (TSs) to add limits and controls for the spent fuel cask loading and unloading operations in the SFP to ensure its continued compliance with NRC regulations governing the safe handling of irradiated fuel in the SFP. Specifically, the proposed changes would revise the FCS TSs by adding a new Limiting Condition for Operation (LCO) 2.8.3(6) that establishes (1) an SFP boron concentration requirement during cask loading operations in the SFP, and (2) a spent fuel burnup-initial enrichment limit in the spent fuel cask to ensure subcritical conditions are maintained during spent fuel cask loading operations in the SFP. In addition, the proposed amendment would also modify TS Tables 3-4 and 3-5, and add a new subsection 4.3.1.3 in Design Features 4.3.1 to describe the spent fuel cask design features, and conform pagination.

The FCS TSs currently permit the licensee to store unirradiated (fresh) fuel and spent fuel assemblies in the FCS SFP storage racks. However, OPPD plans to implement dry cask storage in an ISFSI facility at FCS in accordance with the general license provisions of 10 CFR Part 72, Subpart K, using the Transnuclear (TN) Standard NUHOMS® System. OPPD intends to load spent fuel into the NUHOMS®-32PT dry shielded canister (DSC) in its spent fuel cask loading area for subsequent removal and dry storage on the ISFSI.

The supplemental letters dated March 17 and 27, 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 20, 2005 (70 FR 75494).

2.0 REGULATORY EVALUATION

Because the fuel basket inside the NUHOMS®-32PT DSC has a different geometry spacing and neutron poison plate design than the FCS spent fuel storage racks, separate criticality analyses are required to demonstrate compliance with the Part 50 regulations.

While in the SFP for wet loading operation, both Part 50 and Part 72 requirements pertaining to criticality control apply to spent fuel cask loading operations. General Design Criterion (GDC) 62 in Appendix A to 10 CFR 50 specifies that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The acceptance criteria specified in 10 CFR 50.68 section (b)(1) and (b)(4) for criticality prevention in the SFP that are applicable to the licensee's proposed amendment are, respectively:

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; and

2. The effective neutron multiplication factor (K_{eff}) of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity, including an allowance for uncertainties, must not exceed 0.95 at a 95 percent probability at a 95 percent confidence (95/95) level, if flooded with borated water, and the K_{eff} must not exceed 1.0 (subcritical) at a 95/95 level, if flooded with unborated water.

Under 10 CFR 72.124, "Criteria for nuclear criticality safety," the NRC regulates dry cask storage activities to ensure that subcriticality is maintained during the handling, packaging, transfer, and storage of spent fuel assemblies. The NRC regulations for dry cask criticality prevention rely on favorable geometric configurations and fixed neutron absorbers. However, unlike 10 CFR 50.68, the 10 CFR Part 72 regulations for criticality prevention in dry casks allow licensees to credit the SFP soluble boron for maintaining subcritical conditions during cask loading, unloading, and handling operations in the SFP. Therefore, many cask designs have incorporated soluble boron credit, in lieu, of a burnup credit as a means of increasing day cask storage capacity while maintaining subcritical conditions. OPPD's amendment request proposes to demonstrate that it can satisfy the applicable 10 CFR 50.68 criticality prevention requirements, with a burnup credit, during cask loading, unloading, and handling operations in the SFP.

The NRC defines the acceptable methodologies for performing criticality analyses in the following documents:

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

2. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis;" and

3. NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998.

The NRC staff used these documents to assist in its review of the licensee's amendment request to ensure compliance with GDC 62 and 10 CFR 50.68.

3.0 TECHNICAL EVALUATION

3.1 Spent Fuel Cask System Description

The TN Standard NUHOMS® System incorporates a number of DSCs for dry storage of spent fuel. Each DSC includes a variety of options for the fuel basket design to offer operational flexibility. Different fuel basket designs may be used based on the characteristics of the fuel to be loaded and the level of SFP soluble boron desired. OPPD chose to use the NUHOMS®-32PT PWR DSC for dry storage of the FCS spent fuel. NUHOMS®-32PT DSC has three types of fuel baskets, namely, Type A, B, and C, which are differentiated by the maximum permitted fuel enrichment, neutron poison plate configuration, number of poison rod assemblies (PRA), and soluble boron. The cask user may choose 16, 20, or 24 poison plates in the fuel basket based on the enrichment of the fuel to be loaded and the desired soluble boron limit. The PRAs are inserts required by the Standard NUHOMS® Part 72 Certificate of Compliance. The PRAs are installed in certain fuel storage locations in the NUHOMS®-32PT DSC fuel basket designs for criticality control. There are 0, 4, and 8 PRAs in the Type A, B, and C baskets, respectively.

Operationally, an empty DSC with an integral fuel basket is inserted into a TN OS197L transfer cask (TC) and the assemblage is placed in the cask loading area of the FCS SFP. Except for the floor of the cask loading area being approximately two feet below the floor of the SFP proper, there is no physical barrier between the cask loading area and the spent fuel racks. Up to 32 fuel assemblies, meeting the specified limits, can be moved from the fuel storage racks into the NUHOMS®-32PT DSC. Upon completion of fuel movement, while still under water a shield plug is installed into the NUHOMS®-32PT DSC. The DSC/TC assemblage is then removed from the SFP for completion of DSC preparation for deployment at the ISFSI.

Table 3.0-1 in the Framatome Advanced Nuclear Power (FANP) Criticality Analysis report (Ref. 4) lists the NUHOMS®-32PT DSC Type A, B, and C basket designs and soluble boron requirements for the FCS fuel. A Type A fuel basket with a 16-poison plate configuration is the most reactive fuel basket design because of the fewer poison plates. Although FCS spent fuel may be stored in the Type A, B, or C baskets, for conservatism, the Type A basket with 16 poison plates and no credit for PRAs is used in the criticality analyses performed in support of this amendment request.

3.2 Description of Proposed Technical Specification Changes

Attachment 2 to the November 8, 2005, letter provided a markup of the TSs and corresponding bases pages. The following is the descriptive list of the changes proposed by the amendment request:

(1) Addition of TS 2.8.3(6), “Spent Fuel Cask Loading”

TS 2.8.3(1) through 2.8.3(5) in the existing TS LCO 2.8.3, “Refueling Operations - Spent Fuel Pool,” include the limits and controls for storage of unirradiated fuel and spent fuel in the FCS spent fuel storage racks. The licensee proposed to add TS LCO 2.8.3(6), “Spent Fuel Cask Loading,” specifying (1) a new minimum boron concentration limit of 800 parts per million (ppm) for the SFP during spent fuel cask loading operations, and (2) a new burnup versus enrichment curve (TS Figure 2-11) for fuel assemblies located in the spent fuel cask in the SFP. TS 2.8.3(6) also includes required actions when these LCOs are not met.

The acceptability of the new LCO 2.8.3(6) is demonstrated by the licensee’s criticality analysis for compliance with the subcriticality criteria of 10 CFR 50.68. The staff evaluation of the criticality analysis is described in Section 3.3 of this report.

(2) Revision to TS Table 3-4, “Minimum Frequencies for Sampling Tests”

The existing Footnote (4) for TS Table 3-4 specifies the sampling test frequency of the SFP boron concentration as “prior to placing unirradiated fuel assemblies in the spent fuel pool and weekly when unirradiated fuel assemblies are stored in the spent fuel pool.” The licensee proposed to revise Footnote (4) to “prior to placing unirradiated fuel assemblies in the spent fuel pool <u>or placing fuel assemblies in a spent fuel cask in the spent fuel pool,</u> and weekly when unirradiated fuel assemblies are stored in the spent fuel pool<u>, or every 48 hours when fuel assemblies are in a spent fuel storage cask in the spent fuel pool</u>.” [Underline added]

As a result of this amendment request, this revision is needed to address the boron concentration sampling test frequency for the conditions when spent fuel assemblies are located in a spent fuel cask in the SFP. The proposed change provides a new restriction for the spent fuel cask that is more restrictive than the current TSs and, therefore, the change is acceptable.

(3) Revision to TS Table 3-5, “Minimum Frequencies for Equipment Tests”

The licensee proposed to add new item 24 to TS Table 3-5 to address spent fuel cask loading operations by requiring the licensee to “verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11” at a frequency of “prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool.”

This revision is an added surveillance requirement to assure that LCO 2.8.3(6)(2) regarding TS Figure 2.11, burnup-enrichment load curve, is met. The proposed change provides a new restriction for the spent fuel cask that is more restrictive than the current TSs and, therefore, the change is acceptable.

(4) Revision to TS 4.3.1, "Criticality for Fuel Storage"

The licensee proposed to add TS Subsection 4.3.1.3 to specify that the spent fuel casks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR [updated safety analysis report],

c. k_{eff} # 0.95 if fully flooded with borated water $ 800 ppm, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

d. A nominal 9.075-inch center-to-center distance between fuel assemblies placed in the spent fuel cask,

e. Spent fuel assemblies with a combination of discharge burnup and initial average assembly enrichment in the "acceptable" range of Figure 2-11.

This added Section 4.3.1.3 describes the characteristics of the spent fuel cask loading in the SFP. Items b and c specify the commitment to meet 10 CFR 50.68 reactivity limits. Items a, d, and e, respectively, specify the requirements regarding the enrichment limit, fuel loading spacing, and minimum burnup limit of fuel assemblies loaded in the spent fuel cask to assure compliance with the subcriticality requirements. The proposed change provides new TS restrictions for the spent fuel cask which previously did not exist and, therefore, the change is more restrictive and is acceptable.

(5) Editorial Changes

As result of the proposed TS changes, several editorial changes to the formatting of the TSs were needed. In the Table of Contents, the Figure 2-11 was added and the other figures were placed in numerical order. Also on this page, the spelling of "concentration" in the title of Figure 2-3 was corrected. In Section 2.8, Pages 14, 15, 26 and 27 were added. Because pages 14 and 15 were new, Section 2.8 from pages 14 to 23 were repaginated to 16 to 25. Section 4.3.1.3 was added to Section 4.0 on page 2. As a result some text on page 2 was moved to page 3. The staff has reviewed these changes and concluded that they are editorial in nature and, therefore, are acceptable.

In our review of the proposed TS changes, the NRC staff reviewed the supporting criticality analysis to ensure that it is consistent with the assumptions and inputs used in the supporting safety analysis.

3.3 <u>Criticality Analysis</u>

Attachment 1, "Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool," and Enclosure 1, "Framatome ANP [Framatome Advanced Nuclear Power] Criticality Analysis," to the November 8, 2005, letter, as amended by LIC-06-0023 Enclosure 1, "NUHOMS-32PT 50.68 Criticality Analysis for Fort Calhoun (Revision Number 1)" (Ref. 4), describe the criticality analyses performed by the licensee and its contractor FANP to support the proposed amendment for the spent fuel cask loading in the NUHOMS®-32PT DSC/TC in the FCS SFP.

The FANP modeling approach used bounding parameters to produce the maximum k_{eff}. In determining the acceptability of the amendment, the NRC staff evaluated the appropriateness of the computer codes employed for the analyses, the methodology used to calculate the maximum k_{eff}, and the criticality analyses to demonstrate compliance with the regulatory reactivity limits. The NRC staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins in accordance with NRC acceptance criteria were developed and could be maintained in the FCS SFPs during cask loading, unloading, and handling operations.

3.3.1 Criticality Analysis Codes

The FCS spent fuel cask loading criticality analyses were performed using the KENO V.a code to calculate k_{eff}. The CASMO-3 code was used for the fuel depletion analyses and determination of the isotopic atom densities of the spent fuel assemblies. The FANP KENO V.a code is a part of the SCALE Version 4.4a code package (Ref. 5) operating on the Linux operating system platform. KENO V.a is a three-dimensional Monte Carlo criticality code that was benchmarked against critical experiments that are representative of SFP and canister configurations. CASMO-3 is a multi-group, two-dimensional transport theory program for burnup calculations on light-water reactor assemblies consisting of cylindrical fuel rods of varying composition on the square array. It is typically used to generate cross-sections of the fuel cycles, and is used for reactivity studies to provide depletion data for burnup credit. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. In an NRC memorandum dated August 19, 1998, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," (Ref. 3) providing guidance on fuel storage criticality analysis, the NRC staff stated that KENO V.a and CASMO-3 codes were acceptable computer codes for the analysis of fuel assemblies stored in the SFP.

The licensee performed independent benchmarks to establish the appropriate uncertainties for the criticality safety analysis of the NUHOMS®-32PT DSC in the FCS SFP, including comparisons of the FANP KENO V.a calculations to those from TN calculations that were independently benchmarked to a set of critical experiments. Also, the KENO V.a calculations were used to model the FCS SFP in support of earlier TSs. The CASMO-3 calculations were used for benchmark comparisons of the FCS reactor operation during several reload cycles. Table 4.4 of Enclosure 1 provides a comparison of an FANP KENO calculation result of k_{eff} and σ for different neutronic histories ranging from 500,000 to 4 million. The table shows the results are statistically equal for all cases, except for a couple of cases that slightly exceed 2σ. Therefore, cases with approximately 1 million histories are sufficient to assure convergence of the KENO V.a reactivity calculations. However, FANP elected to use 2.5 million histories for conservatism. The NRC staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the FCS spent fuel casks in the SFP with a high degree of confidence.

3.3.2 Criticality Analysis Methodology

The NRC Criticality analysis guidance (Ref. 3) specifies that the Maximum k_{eff} value for the criticality analysis should be the summation of the KENO V.a-calculated k_{eff}, the bias in criticality analysis methods, manufacturing and calculational uncertainties, and the correction for the effect of the axial distribution in burnup when credit for burnup is taken. Uncertainties were determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations.

The licensee's modeling approach for the NUHOMS®-32PT DSC/TC criticality analysis is the use of bounding parameters in the KENO V.a design-basis model that produces the maximum k_{eff}. The KENO model of the DSC/TC utilized in the criticality analysis is directly obtained from the bounding criticality analysis model utilized in the Part 72 criticality analysis. The design-basis model is constructed utilizing the worst case geometric and material tolerances that result in the most conservative calculation of the system reactivity. A series of sensitivity studies were performed to evaluate the reactivity effects of geometry and material tolerances of various DSC/Cask components and fuel. The FANP criticality analysis report presented the results of some sensitivity studies on the KENO V.a calculation of the k_{eff}, such as the effect of replacing the lead (Pb) gamma shield in the transfer cask with stainless steel 304 of the "light" TC (Table 4-5), and the effect of eccentric positioning of fuel assemblies in the DSC fuel basket (Table 4-6). The design-basis criticality model is then constructed by combining deterministically the worst reactivity effects of these parameters.

For the NUHOMS®-32PT DSC, the design-basis criticality analysis is modeled with the most reactive system configurations, including the moderator density and fuel assembly spacing. The KENO V.a results include the bounding uncertainties for the canister and SFP. Since the uncertainties associated with the geometrical/material tolerances are conservatively built into the criticality analysis model of the DSC/TC, the only additional uncertainties that need to be considered are those associated with the methods and those associated with the burned fuel isotopic concentrations.

KENO V.a Benchmark Bias

The FANP criticality analysis of the spent fuel cask loading in the SFP includes two sets of independent benchmarks: (1) experiments of specific configurations that are comparable to the FCS SFP and the DSC; and (2) experiments of fuel assembly configurations that are comparable to the FCS fuel assemblies.

The KENO V.a benchmark calculations modeled 100 critical experiments that are representative of SFP and canister configurations. Twenty-one of the 100 experiments were performed by FANP-B&W, and were recommended in the NRC guidance for criticality analysis (Ref. 3). This set included the effects of burnup by modeling plutonium fuel in addition to uranium fuel. The approach utilized for the NUHOMS®-32PT DSC criticality safety analysis was to bound all uncertainties associated with the benchmark results. The bounding benchmark uncertainty was determined to be 0.01549 Δk_{eff} for criticality safety evaluations of loading the DSC in the FCS SFP.

The second set of benchmarks validated the CASMO-3 methods. The results of this benchmark indicated that the bias and random uncertainties associated with CASMO-3 were smaller than those associated with KENO, i.e., no statistical significant bias could be observed and the random deviations were the result of the same type of parameters in the KENO V.a benchmarks. Consequently, no additional uncertainty was assigned to the CASMO methods with fuel assemblies.

Spent Fuel Burnup Uncertainties

In the burnup credit analyses, biases are added to the bounding criticality safety predictions to account for the uncertainties associated with (1) the axial burnup distribution in the active fuel length, and (2) the assembly burnup accuracy. The dominant bias is from the axial burnup effects.

The burnup credit calculations were performed assuming a uniform burnup profile throughout the active fuel length that results in the over-prediction of burnup at the ends of the fuel assembly and under-prediction of burnup in the fuel mid-region. The "axial end-effect" bias, i.e., the difference of the k_{eff} values between the axial burnup profile and the uniform burnup assumption, needs to be applied to the burnup credit calculations to account for the increase in reactivity. Generic analyses confirmed minor and generally negative reactivity effect of the axially distributed burnup at values less than 30,000 megawatt days/metric tons Uranium (MWD/MTU). As a result, KENO calculations with less than 30,000 MWD/MTU do not contain an axial bias. The highest burnup evaluated in the analysis was 38,000 MWD/MTU, to which the axial bias uncertainty of +0.013 Δk is applied.

Another major contributor to the bounding uncertainty is the bias in the assembly burnup to account for the uncertainty in the fuel depletion calculations. The NRC guidance document (Ref. 3) stated an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption. The licensee in its response to an NRC staff question (Ref. 2) stated that the reactivity decrement uncertainty due to burnup was incorporated by shifting the burnup curve upwards by 5 percent uniformly for all burnups.

In summary, the NRC staff evaluated the licensee's criticality analysis methodology, and concluded that the use of bounding parameters to produce the maximum k_{eff} is consistent with the NRC criticality analysis guidance and is acceptable.

3.3.3 DSC Loading Criticality Analysis

To demonstrate compliance with the reactivity criteria of 10 CFR 50.68, the licensee performed the criticality analyses of the NUHOMS®-32PT DSC/TC cask loading in the SFP for both normal and accident conditions. The criticality analyses were performed with the NUHOMS®-32PT DSC Type A fuel basket with 16 poison plates and no PRA, which is the most reactive and bounding DSC basket design. Consistent with the methodology of using bounding parameters, the criticality analyses were performed with the conservative assumptions that tend to maximize the reactivity. These assumptions include:

- FCS SFP peripheral cells adjacent to Cask Pit Area maintained empty during DSC loading.

- No burnable poisons accounted for in any fuel assembly in the KENO model.

- The transition rails between the basket and the canister shell modeled as 100 percent aluminum (steel in the transition rails reduces reactivity because of much higher absorption cross-sections).

- Control rods assumed inserted for CASMO depletion calculations to maximize k_{eff}.

- Water density was at the optimum moderator density of 1.00 gram/cc, corresponding to 4 ^{o}C.

- All fuel rods are filled with fresh water in the pellet/cladding gap.

- Full DSC reflection in the radial direction.

- Only 90 percent of credit taken for of B-10 in the DSC neutron poison plates.

OPPD has demonstrated compliance with the subcriticality criteria of 10 CFR 50.68 that k_{eff} # 0.95 if credit is taken for soluble boron in the SFP, and k_{eff} < 1.0 if flooded with fresh water. Two normal spent fuel cask loading operation conditions were analyzed with fresh water and 500 ppm borated water, respectively, in the SFP. For the case of the SFP flooded with fresh water, the licensee performed a series of CASMO/KENO runs with various enrichment-initial burnup combinations to achieve the maximum k_{eff} < 1.0. The results, shown in Table 6-1 and Figure 6-1 of the FANP criticality analysis report, show that the most reactive configuration has k_{eff} = 0.99713, below the acceptance criterion of 1.0. It should be noted that the assembly burnups listed in Table 6-1 and Figure 6-1 have been increased by 5 percent from the burnups used in the analysis to account for burnup uncertainty. This analysis shows that a complete dilution of the boron in the SFP would not cause a criticality in the NUHOMS®-32PT DSC. The most reactive case was again analyzed with 500 ppm soluble boron in the SFP. The results show k_{eff} = 0.92073, below the acceptance criterion of 0.95.

The licensee analyzed two criticality accident conditions, i.e., misloading of a fresh fuel assembly and dropping of a fresh fuel assembly, that are consistent with the FCS current design and licensing basis for the SFP. The accident analyses were analyzed based on the "double contingency" principle that at least two unlikely, independent accidents or upsets have to occur concurrently or sequentially for a criticality event to be possible. Therefore, a boron dilution event in the SFP, that is not a design-basis event for the FCS, was not postulated to occur as an accident event, either individually or concurrently with the two accidents analyzed. The fuel misloading accident was analyzed assuming that the transfer cask was loaded with fuel assemblies having enrichments up to 4.55 wt% U-235, and the empty position loaded with a fresh fuel assembly of 4.5 wt% enrichment. Multiple empty locations were assumed in order to locate the most reactive empty cell. The soluble boron concentration values ranged from 500 to 800 ppm were assumed in the analysis. The results, provided in Table 6.3 of the criticality analysis report, show that the minimum required soluble boron concentration is 800 ppm to maintain k_{eff} # 0.95 with all uncertainties. The bounding value for the misloading fuel bundle accident with 800 ppm soluble boron is 0.94623 for the DSC fuel enrichment of 4.55 wt%.

The fuel drop accident was evaluated assuming the dropped assembly was placed along the perimeter of the transfer cask aligned longitudinally and evaluated at different azimuthal locations to find the most reactive position. The analysis was performed for various soluble boron concentrations. The results, shown in Table 6-4 of the criticality analysis report, show that with the 800 ppm boron concentration, the most reactive case has k_{eff} = 0.92499.

The results of the criticality analyses for normal and accident conditions are summarized in Table 4.1-1 of Attachment 1 to the November 8, 2005, letter. The results show that the spent fuel cask system remains sufficiently subcritical under normal and applicable accident conditions in the FCS licensing basis. Based on the results of the criticality analyses, a minimum boron concentration of 800 ppm is required in the DSC fuel cavity water during fuel loading for accident conditions. In addition, Table 4.1-2 of Attachment 1 adopts from Table 6.1 of the criticality analysis report the burnup-enrichment requirement. All FCS fuel assemblies loaded into the NUHOMS®-32PT DSC must have a minimum average assembly burnup greater than or equal to the value shown in Table 4.1-2.

Since the criticality analyses for normal and accident conditions of the DSC loading with fuel assemblies, with the burnup-enrichment combinations of Table 4.1-2, have shown compliance with the subcritical criteria of 10 CFR 50.68, these burnup/enrichment combinations are used as the burnup credit loading curve. This loading curve is adopted as TS Figure 2-11, "Limiting Burnup Criteria for Acceptable Storage in Spent Fuel Pool."

The NRC staff reviewed the licensee's amendment to add LCO 2.8.3(6) specifying (1) a minimum boron concentration requirement of 800 ppm for the SFP during spent fuel cask loading operations, and (2) a new burnup versus enrichment curve for fuel assemblies located in the spent fuel cask, as well as other changes to TS Tables 3-4 and 3-5, and TS 4.3.1, related to the spent fuel cask loading operations in the FCS SFP. In addition, editorial changes were made mostly to make the TSs consistent with the proposed changes and to conform the pagination. Based on its review of the licensee's criticality analysis, the NRC staff concludes that the proposed changes for the spent fuel cask loading operations in the SFP meet appropriate subcriticality requirements of 10 CFR 50.68 and GDC 62. The NRC staff found that the licensee's amendment request provided reasonable assurance that under both normal and accident/upset conditions, that licensee would be able to safely operate the plant and comply with the NRC regulations. Therefore, the NRC staff finds the amendment acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (70 FR 75494; December 20, 2005). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from R.T. Ridenoure (OPPD) to the U.S. Nuclear Regulatory Commission, “Fort Calhoun Station Unit No. 1 License Amendment Request (LAR) 05-013, ‘Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool,’” November 8, 2005, LIC-05-0119.

2. Letter from D. J. Bannister (OPPD) to the U.S. Nuclear Regulatory Commission, “Response to Requests for Additional Information and Revision of Fort Calhoun Station Unit No. 1 License Amendment Request, ‘Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool,’” March 27, 2006, LIC-06-0023.

3. NRC Memorandum from L. Kopp to T. Collins, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” August 19, 1998.

4. LIC-06-0023 [Reference 2 above] Enclosure 1, “NUHOMS-32PT 50.68 Criticality Analysis for Fort Calhoun (Revision Number 1).”

5. NUREG/CR-0200, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” distributed by the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1998.

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