your energy partner®
OPPD
Omaha Public Power District
444 South 16th Street Mall
Omaha NE 68102-2247

November 8, 2005
LIC-05-0119

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

References:
1. Docket Nos. 50-285 and 72-054
2. USNRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," March, 23, 2005

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request (LAR) 05-013, "Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby requests the following amendment to the Fort Calhoun Station Unit No. 1 (FCS) operating license.

OPPD proposes to revise the FCS Technical Specifications (TS) to add a new Limiting Condition for Operation (LCO) 2.8.3(6) and to modify Table 3-4, Table 3-5, and Design Features 4.3.1 to address criticality control during spent fuel cask loading operations in the spent fuel pool. This request applies only to spent fuel cask loading in the spent fuel pool and does not affect the licensing basis or invalidate our existing exemption from the criticality monitoring requirements of Title 10, Code of Federal Regulations (CFR) 70.24 for new and spent fuel storage as discussed further in Section 5.0 of Attachment 1.

Regulatory Issue Summary (RIS) 2005-05, Reference 2, highlights differences in the Nuclear Regulatory Commission (NRC) 10 CFR 50 criticality requirements for the spent fuel pool and 10 CFR 72 requirements for spent fuel casks, and emphasizes that licensees are expected to comply with both Part 50 and Part 72 during cask loading operations in the spent fuel pool. This LAR is consistent with the regulatory direction provided in RIS 2005-05 and provides appropriate controls to ensure that an accidental or inadvertent criticality during spent fuel cask loading operations at FCS is highly unlikely.

Attachment 1 provides the technical bases and the No Significant Hazards Evaluation for these requested changes to the FCS TS. Attachment 2 contains a marked-up version of the TS that shows proposed new LCO 2.8.3(6) and its Bases, and the proposed amendments to the TS tables and design features. Attachment 3 contains a clean version of the TS that incorporates the proposed changes to the TS provided in Attachment 2. Attachment 4 contains a listing of formal OPPD commitments associated with this amendment. Enclosure 1 contains the supporting criticality analysis.

OPPD requests approval of the proposed amendment by March 1, 2006, with a maximum of 60 days for implementation to support scheduled dry spent fuel cask loading operations shortly thereafter. In accordance with 10 CFR 50.91, a copy of this application, with attachments and the enclosure, is being provided to the designated State of Nebraska official.

If you have any questions or require additional information, please contact Mr. Thomas C. Matthews at 402-533-6938.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 8, 2005.

Sincerely,

R.T. Ridenoure
Vice President

RTR/bg

Attachments: 1. OPPD's Evaluation for Amendment of Operating License
2. Mark-up of Technical Specifications and Bases pages
3. Proposed Technical Specifications and Bases pages
4. List of Regulatory Commitments

Enclosure: 1. Framatome ANP Criticality Analysis

cc: Division Administrator – Public Health Assurance, State of Nebraska

Omaha Public Power District's Evaluation for Amendment of Operating License

"Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool"

1.0 DESCRIPTION

The Omaha Public Power District (OPPD) is requesting to amend Operating License DPR-40 for Fort Calhoun Station (FCS), Unit No. 1. The proposed changes would revise the FCS Technical Specifications (TS) to add limits and controls for spent fuel cask loading and unloading[1] operations in the spent fuel pool. This License Amendment Request (LAR) is being submitted in response to NRC Regulatory Issue Summary (RIS) 2005-05 (Reference 7.1). An amendment to the FCS operating licensing is required to support dry storage cask loading operations, scheduled to begin on or about March 1, 2006.

Currently, the FCS TS include limits and controls for storage of unirradiated (fresh) fuel and spent fuel in the FCS spent fuel pool storage racks. Spent fuel cask loading in the spent fuel pool in support of dry cask storage has not previously been performed at FCS. OPPD plans to implement dry spent fuel storage under the general license provisions of 10 CFR 72, Subpart K (Docket No. 72-054) utilizing the Transnuclear Standard NUHOMS® System (10 CFR 72 Certificate of Compliance No. 1004). As a result of our review of RIS 2005-05 and discussions with NRC staff, OPPD has determined that a Part 50 operating license amendment is necessary to support cask loading operations. Because the fuel basket inside the Transnuclear 32PT dry storage canister (DSC) has a different geometric spacing and neutron poison plate design than the FCS spent fuel storage racks, separate criticality analyses were required to demonstrate compliance with the Part 50 regulations, and corresponding new and revised Part 50 TS were deemed necessary.

Spent fuel cask loading will be performed in the northwest corner of the FCS spent fuel pool in a cask loading area adjacent to existing Region II spent fuel storage racks designated "D," "G2," and "E." The proposed TS changes are consistent with the assumptions and inputs used in the supporting criticality analysis (Enclosure 1). The criticality analysis is consistent with previously accepted methodologies used in licensing actions for the FCS plant and at other nuclear power plants.

Recent similar license amendment requests submitted by Southern Nuclear Operating Company and Entergy Operations (References 7.2 and 7.3), and associated responses to NRC Requests for Additional Information (RAI) (References 7.4 and 7.5) have been reviewed. OPPD has taken into consideration the content of those amendment applications and the issues discussed in the RAIs in developing this LAR, to the extent the information is applicable to FCS, in an attempt to reduce or eliminate any RAIs for this license amendment request.

The following sections include detailed information regarding the proposed changes, background, technical basis, regulatory requirements, no significant hazards, and environmental considerations associated with this license amendment request.

---

[1] This LAR addresses any time the spent fuel cask is submerged in the spent fuel pool with one or more fuel assemblies in the cask during loading or unloading operations. Hereafter in this LAR, only loading operations will be discussed for simplicity. The proposed TS are written appropriately to govern both loading and unloading operations.

2.0 PROPOSED CHANGES

The specific proposed changes to the FCS Technical Specifications (as shown in Attachment 2) are as follows:

New LCO 2.8.3(6), "Spent Fuel Cask Loading"

This new LCO adds: 1) a new minimum boron concentration limit (800 ppm) for the spent fuel pool during spent fuel cask loading operations and 2) a new burnup versus enrichment curve for fuel assemblies located in a spent fuel cask in the spent fuel pool. New bases for new LCO 2.8.3(6) have been created and are included for information.

Table 3-4, "Minimum Frequencies for Sampling Tests"

Revised Footnote (4) to this table addresses boron concentration sampling test frequency prior to, and during the time that spent fuel assemblies are located in a spent fuel cask in the spent fuel pool.

Table 3-5, "Minimum Frequencies for Equipment Tests"

New Item 24 added to Table 3-5 addresses spent fuel cask loading operations.

Design Features Section 4.3.1

New Subsection 4.3.1.3 addresses spent fuel cask design features.

In summary, this request is consistent with the regulatory direction provided in RIS 2005-05 and provides appropriate administrative controls to ensure that an accidental or inadvertent criticality during spent fuel cask loading operations at Fort Calhoun Station is highly unlikely.

3.0 BACKGROUND

NRC Regulatory Issue Summary (RIS) 2005-05

NRC RIS 2005-05 (Reference 7.1) describes overlapping regulatory requirements between 10 CFR Part 72 and 10 CFR Part 50 pertaining to the loading of spent fuel casks in the spent fuel pool. Specifically, it states that the requirements of 10 CFR 50.68 (Reference 7.6) apply during spent fuel cask loading in the spent fuel pool. 10 CFR 50.68 establishes requirements that apply to the storage of unirradiated (fresh) and spent fuel at the facility, including wet storage of spent and fresh fuel, and dry storage of fresh fuel in vaults or racks. The RIS highlights the NRC's expectation that licensees comply with all applicable requirements in 10 CFR 72 and 10 CFR 50 during cask loading in the spent fuel pool.

The regulatory requirement that forms the basis for this LAR is 10 CFR 50.68(b)(1), which states:

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

The NRC criteria for criticality control during spent fuel cask loading operations have been historically governed solely by the requirements of 10 CFR 72. Likewise, the criteria for criticality control of spent fuel stored in the spent fuel pool storage racks are governed by the requirements of 10 CFR 50. Part 50 and Part 72 have different acceptance criteria for the criticality analyses that independently provide adequate assurance that the spent fuel will remain subcritical in their respective storage configurations. Parts 50 and 72 also require unique Technical Specifications to be established that are applicable to spent fuel storage in the spent fuel storage racks and in the spent fuel cask during loading operations, respectively.

10 CFR 50.68(b)(1) requires that fuel assemblies handled or stored together in any quantity remain safely subcritical under the most adverse moderator conditions feasible by unborated water. "Safely subcritical" is defined more explicitly in 10 CFR 50.68(b)(4) for spent fuel stored in the spent fuel storage racks. For purposes of spent fuel cask loading in the spent fuel pool, these "rack requirements" are assumed to apply to the cask if at least one fuel assembly is located in the cask while it is in the spent fuel pool. 10 CFR 50.68(b)(4) requires the effective neutron multiplication factor (k_{eff}) for fuel in the cask to meet one of the following two criteria, as demonstrated by analysis, while in the spent fuel pool:

1. If no credit for soluble boron is taken, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

2. If credit is taken for soluble boron, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k_{eff} must

remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The spent fuel to be loaded into the NUHOMS® System 32PT Dry Storage Canister (DSC) at FCS cannot be shown by analysis to meet $k_{eff} \leq 0.95$ with unborated water. Therefore, the second set of criteria from 10 CFR 50.68(b)(4) has been used to demonstrate compliance with 10 CFR 50.68(b)(1) for spent fuel cask loading at FCS for this LAR.

In order to demonstrate $k_{eff} < 1.0$ for the spent fuel when flooded with unborated water, the NRC Office of Nuclear Reactor Regulation (NRR) has historically permitted licensees to credit the reduced reactivity of the spent fuel associated with burnup during operation in the Part 50 criticality analysis. The NRC Spent Fuel Project Office (SFPO) has historically required a maximum value of k_{eff} (≤ 0.95) to be demonstrated with all fuel in the spent fuel cask assumed to be fresh fuel at the maximum enrichment allowed by the cask Certificate of Compliance (CoC) as described in NUREG-1536 (Reference 7.9). (Current SFPO review guidance does permit a limited amount of burnup credit to be considered.)

To date, no spent fuel storage systems have been licensed under 10 CFR 72 with burnup credit considered in the criticality analysis. Instead, the criticality analysts have taken credit for the negative reactivity of soluble boron in the spent fuel pool during loading operations for PWR fuel. Thus, Part 72 CoCs require soluble boron credit for certain PWR fuel storage systems to maintain spent fuel in the cask sufficiently subcritical during cask loading operations in the spent fuel pool. In addition, certain Part 50 criticality analyses also incorporate credit soluble boron in the spent fuel pool. However, the minimum soluble boron concentrations in the spent fuel pool required by the Part 50 and Part 72 Technical Specifications are also dependent upon differences in the storage system geometries and the amount of credit taken for neutron poison in the fixed neutron absorber in the spent fuel storage racks and spent fuel cask in the respective criticality analyses. These differences in criticality methodology and acceptance criteria, and the need to comply with both Part 50 and Part 72 during cask loading operations, are described in detail in RIS 2005-05.

FCS Dry Spent Fuel Storage

As part of the long-term spent fuel management strategy at FCS, OPPD has decided to move some of its spent fuel assemblies currently in the spent fuel pool into dry storage at an on-site Independent Spent Fuel Storage Installation (ISFSI) under the general license provisions of 10 CFR 72, Subpart K. ISFSI operations are expected to begin in the first quarter of 2006 and proceed with periodic loading campaigns into the future. OPPD has chosen the Transnuclear Standard NUHOMS® System using the 32PT DSC for dry spent fuel storage. OPPD will load the 32PT DSC under Amendment 8 to the CoC, which is expected to be effective on December 5, 2005 (FR Notice dated 9/20/05). Depending on the type of fuel basket in the 32PT DSC and the enrichment of the fuel to be stored, the Technical Specifications in the NUHOMS® System 10 CFR 72 CoC require anywhere from 1800 to 2500 ppm soluble boron in the DSC for criticality control during wet loading of FCS spent fuel to preserve the assumptions made in the storage system Part 72 design basis criticality analyses.

New criticality analyses, using 10 CFR 50 methods and assumptions (i.e., burnup credit) have been performed for the 32PT DSC with bounding FCS fuel parameters to demonstrate compliance with 10 CFR 50.68(b)(1) during DSC loading operations in the FCS spent fuel pool. This LAR proposes appropriate administrative controls, consistent with these new analyses, for the FCS Technical Specifications to be implemented during wet loading of spent fuel casks in the spent fuel pool.

System Description

The Transnuclear Standard NUHOMS® System is a dry spent fuel storage system certified under 10 CFR 72, Subpart L (CoC No. 1004). The Standard NUHOMS® System incorporates a number of DSC models for dry storage of spent fuel, which differ by capacity for BWR and PWR fuel. Each DSC model also 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 (primarily initial enrichment) and the level of spent fuel pool soluble boron desired. OPPD chose to use the 32PT PWR DSC for dry storage of FCS fuel. FCS fuel may be stored in the Type A, B, or C fuel baskets. The 32PT DSC fuel basket types are differentiated by maximum permitted fuel enrichment, poison plate configuration, number of poison rod assemblies (PRAs), and soluble boron as shown in the table below for the Type A, B, and C baskets for FCS fuel:

Table 3.0-1

NUHOMS® 32PT DSC Basket Design and Soluble Boron Requirements for FCS Fuel[2]

	Type A Basket – No PRAs			**Type B Basket – 4 PRAs**		**Type C Basket – 8 PRAs**		**Minimum Soluble Boron (ppm)**
	Poison Plate Configuration			**Poison Plate Configuration**		**Poison Plate Configuration**		
	16	**20**	**24**	**20**	**24**	**20**	**24**	
Initial Fuel Enrichment (w/o ^{235}U)	3.35	3.40	3.50	3.90	4.00	4.35	4.35	1800
	3.50	3.60	3.70	4.10	4.20	4.55	4.55	2000
	3.60	3.65	3.80	4.20	4.30	4.70	4.70	2100
	3.70	3.75	3.90	4.30	4.40	4.80	4.80	2200
	3.75	3.85	4.00	4.40	4.50	4.90	4.90	2300
	3.80	3.90	4.05	4.50	4.60	5.00	5.00	2400
	3.90	4.05	4.15	4.55	4.70	-	-	2500

The maximum enrichment of fuel analyzed for FCS spent fuel cask loading is 4.55 w/o ^{235}U. This bounds the maximum enrichment currently licensed for use in the FCS reactor in the most reactive fuel basket design – the Type A basket with 16 poison plates. This analysis and the resulting administrative limits in the proposed TS, therefore, bound the loading of FCS fuel enriched up to 4.5 w/o ^{235}U in the Type A, B, or C fuel basket designs. The details of the criticality analyses are provided in Section 4.0. The poison plate configurations permit the cask user to 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. All neutron poison plates have the same ^{10}B

[2] Data taken from Standard NUHOMS® 10 CFR 72 Certificate of Compliance, Amendment 8, Table 1-1g.

areal density of 0.007 grams per square centimeter[3]. Poison Rod Assemblies are inserts required by the Standard NUHOMS® Part 72 CoC to be installed in certain fuel storage locations in the Type B and C 32PT DSC fuel basket designs for criticality control. For conservatism, no credit for PRAs is taken in the Part 50 criticality analyses performed in support of this LAR. The Type A fuel basket with the 16-poison plate configuration was modeled in all cases because the fewer poison plates make it the most reactive fuel basket type among the three under consideration.

Operationally, an empty DSC with integral fuel basket is inserted into a Transnuclear OS197L transfer cask and the assemblage is placed in the cask loading area of the FCS spent fuel pool. The cask loading area is located in the northwest corner of the spent fuel pool adjacent to Region II spent fuel racks designated "D," "G2," and "E." There is no physical barrier (i.e., cask loading pit) between the cask loading area and the spent fuel racks; however, the floor of the cask loading area is approximately two feet below the floor of the spent fuel pool proper.

Up to 32 FCS fuel assemblies meeting the limits specified in the NUHOMS® System CoC and the enrichment limit in the Part 50 FCS Technical Specifications are moved from the spent fuel storage racks into the 32PT DSC. While in the spent fuel pool, both Part 50 and Part 72 requirements pertaining to criticality control apply to spent fuel cask loading operations. During wet loading operations, the NUHOMS® System CoC requires a minimum concentration of soluble boron in the water inside the DSC for criticality control (see table above), which is verified before wet loading operations begin and periodically thereafter in accordance with the Part 72 Technical Specifications. This LAR proposes additional Part 50 Technical Specification controls on fuel enrichment, boron concentration and minimum fuel assembly burnup that will also apply during loading operations while the DSC is in the spent fuel pool. Upon completion of fuel movement, a shield plug is installed into the 32PT DSC under water and the DSC/transfer cask assemblage is removed from the spent fuel pool for completion of DSC preparation for deployment at the ISFSI.

Safety Analysis Report References

Appendix M.8 of the NUHOMS® System 10 CFR 72 Final Safety Analysis Report (FSAR), "Operating Systems" provides additional detail regarding cask loading and unloading operations for the 32PT DSC. Appendix M.6 of the NUHOMS® System FSAR provides additional detail regarding the 10 CFR 72 criticality evaluation for the 32PT DSC. Section 9.5 of the FCS Updated Safety Analysis Report (USAR), "Auxiliary Systems-Refueling Systems" provides additional detail regarding storage of new and spent fuel in the Part 50 facility spent fuel pool and new fuel storage racks.

Existing Operating Condition

No Part 50 Technical Specifications or other administrative controls pertaining to criticality control currently exist to govern spent fuel cask loading in the FCS spent fuel pool. No spent fuel storage casks have been loaded in the FCS spent fuel pool to date. Loading of the NUHOMS® System will not commence without approval of this license amendment request.

[3] Data taken from Standard NUHOMS® 10 CFR 72 Certificate of Compliance, Amendment 8, Table 1-1h.

Proposed Operating Conditions

To provide reasonable assurance that an inadvertent or accidental criticality will not occur during spent fuel cask loading, new FCS LCO 2.8.3(6) is proposed to add a minimum boron concentration in the spent fuel pool during cask loading operations and a minimum fuel burnup versus enrichment curve for assemblies to be loaded into a spent fuel cask in the spent fuel pool. These new operating conditions are consistent with the Part 50-based criticality analyses (Enclosure 1) that demonstrate compliance with 10 CFR 50.68(b)(1). In addition, other portions of the existing Technical Specifications are proposed to be amended as discussed in Section 2.0 above as conforming changes to address spent fuel cask loading in the spent fuel pool.

4.0 TECHNICAL ANALYSIS

4.1 Design Basis

4.1.1 Spent Fuel Racks

The FCS spent fuel pool is designed to prevent criticality by use of neutron absorbing material in the spent fuel racks and by establishing restrictions on the minimum burnup and placement of spent fuel assemblies. Existing TS LCO 2.8.3(1) and accompanying Figure 2-10 ensure that spent fuel stored in the spent fuel racks have the minimum burnup required for storage in the Region 2 racks to maintain k_{eff} less than 0.95 assuming the pool to be flooded with unborated water. The Region 1 racks can be used to store unirradiated (fresh) or spent fuel of any authorized burnup with an initial enrichment up to 4.5 w/o ^{235}U, while meeting the same criticality acceptance criterion. When unirradiated fuel assemblies are to be stored in the spent fuel pool, existing TS LCO 2.8.3(3) requires a minimum soluble boron concentration of 500 ppm to ensure criticality control in the event a fuel assembly not meeting the applicable minimum burnup versus enrichment requirement (up to and including a fresh fuel assembly) is mis-loaded into Region 2.

4.1.2 Dry Spent Fuel Storage System

The spent fuel storage system in general, and the 32PT DSC in particular, are designed to remain safely subcritical during all normal, off-normal, and credible accident conditions. Generic, bounding criticality analyses were performed by the Part 72 CoC holder as part of the NRC certification process under 10 CFR 72, Subpart L. The Part 72 criticality analyses for the 32PT DSC assume 32 fresh fuel assemblies loaded into the canister at the maximum initial enrichment authorized by the CoC and other conservative assumptions consistent with the guidance in NUREG-1536 (Reference 7.9). In order to meet the 10 CFR 72 criticality acceptance criterion of $k_{eff} \leq 0.95$ for the 32PT DSC, soluble boron was assumed in the water in the fuel cavity during fuel loading. The amount of soluble boron required varies based on the type of fuel basket design and the enrichment of the fuel assumed in the analysis (see Table 3.0-1 above).

4.1.3 Dry Spent Fuel Storage System §50.68 Criticality Analysis for Wet Loading Operations

The 10 CFR 50.68 criticality analysis supporting this LAR (Enclosure 1) was performed by Framatome Advanced Nuclear Power (FANP) for a loaded 32PT DSC, including any applicable design changes through CoC Amendment 8. The 32PT DSC was assumed to be inside the OS197L transfer cask and submerged in the FCS spent fuel pool, consistent with the fuel loading operating requirements for the Standard NUHOMS® System. Normal and credible accident conditions were analyzed to demonstrate subcriticality will be maintained with appropriate safety margins, at a 95% probability/95% confidence level (95/95) for a sufficient number of cases to bound all 32PT DSC fuel basket types and fuel permitted by the Part 72 CoC to be loaded into the storage system at FCS. The conditions and acceptance criteria are as follows:

a) Normal conditions with unborated water: $k_{eff} < 1.0$

b) Normal conditions with borated water: $k_{eff} \leq 0.95$

c) Mis-loaded fresh fuel assembly accident condition with borated water: $k_{eff} \leq 0.95$

d) Dropped fresh fuel assembly accident condition with borated water: $k_{eff} \leq 0.95$

4.1.3.1 Applicable Accident Conditions and the Double Contingency Principle

The two accident conditions analyzed, namely mis-loading a fresh fuel assembly and dropping of a fresh fuel assembly, are consistent with the FCS current design and licensing basis for the spent fuel racks. The fresh fuel assembly and dropped fuel assembly are assumed to have an initial enrichment equal to the maximum value currently permitted to be used in the FCS reactor, or 4.5 w/o ^{235}U. Soluble boron is usually present in the spent fuel pool at a concentration of approximately 1900 ppm. A boron dilution event in the spent fuel pool is not a design basis accident for FCS and is, therefore, not postulated to occur as an accident event, either individually or concurrently with the above-mentioned accidents. Nevertheless, complete dilution of the boron in the spent fuel pool would not cause a criticality in the 32PT DSC based on the normal condition, unborated water criticality analysis performed to demonstrate compliance with 10 CFR 50.68(b)(1) and (b)(4).

A boron dilution event is also not postulated to occur coincident with the dropped or mis-loaded fuel assembly accident based on the double contingency principle. The double contingency principle is stated as follows (per ANSI/ANS 8.1):

> *At least two unlikely, independent, and concurrent or sequential changes must be postulated to occur in the conditions essential to nuclear criticality safety before a nuclear criticality accident is possible.*

The fuel misloading or fuel drop events themselves each constitute unlikely, independent events. A fuel misloading would require human error either through mis-identifying an ineligible assembly for loading into the storage system or retrieving the wrong assembly from the fuel storage racks. A fuel assembly drop event requires a mechanical malfunction of the fuel grapple or human error in order to release the assembly at any location other than where the operator wishes it to be released. A simultaneous boron dilution event concurrent with either of these accidents in the spent fuel pool would require unrelated manual operator actions to occur simultaneously that are unrelated to fuel assembly movement. Therefore, credit for the presence of soluble boron is assumed in evaluating the mis-loading and dropped fuel assembly accident conditions. This approach is consistent with the guidance in Section 3 of Reference 7.11.

4.1.3.2 Computer Codes, Methodology, and Prior NRC Review

The computer codes used in the criticality analyses are:

KENO V.a was used for the criticality evaluation. It is a three-dimensional Monte Carlo code developed by the Oak Ridge National Laboratory for the specific purpose of performing criticality safety analyses. The code uses a multi-group library for the energy dependent solution.

CASMO3 was used for the fuel assembly isotopic distribution as a function of burnup. It is a two-dimensional assembly depletion code based on transport theory modeling of the fuel, including the pellets and cell structure within the fuel assembly. The code uses a multi-group library with a first order Legendre expansion for neutron scattering.

The methodology applied in the criticality analysis for the NUHOMS® System 32PT DSC and OS197L transfer cask is the same methodology that was previously used for the Fort Calhoun Station spent fuel pool criticality analysis. This analysis is also consistent with previous FANP analyses through the use of methods and benchmarks that have been previously reviewed and approved by the NRC in other licensing actions. The most recent instances where the FANP methodologies used for this LAR have been submitted for review include:

- FANP Document No. 77-506974-00, "Shearon Harris Criticality Evaluation," Docket 50-400, August, 2005.
- USNRC Docket No. 50-305, "Kewaunee Fresh Storage and Spent Fuel Storage Pool."
- USNRC Docket No. 50-346, "Davis-Besse Fresh Storage and Spent Fuel Storage Pool."
- USNRC Docket No. 50-302, "Crystal River 3 Spent Fuel Storage Pool."
- USNRC Docket No. 50-244, "Ginna Spent Fuel Storage Pool."

Consistency with previous 10 CFR Part 50 analyses for the Fort Calhoun Station spent fuel pool is ensured by benchmark comparisons of the existing Part 50 Technical Specifications associated with spent fuel storage in the spent fuel racks. This comparison reproduced part of the criticality safety analysis results for the spent fuel pool. The benchmark, with bounding uncertainties, is equal to or more reactive than the safety analysis results supporting the existing TS. In addition to the spent fuel pool benchmark, benchmark comparisons of the NRC-certified Transnuclear 32PT DSC were used to ensure consistency with the bounding uncertainties associated with the cask. The bounding cask uncertainties are incorporated into the 10 CFR Part 50 criticality safety analyses supporting this LAR. Additional discussion of benchmarking for these computer codes is provided in Enclosure 1 to this LAR.

4.1.3.3 Assumptions and Conservatisms

The significant assumptions and conservatisms used in this analysis are as follows:

- The criticality analysis was performed for the Standard NUHOMS® 32PT DSC with a Type A fuel basket and all fuel having the maximum enrichment of 4.55 w/o ^{235}U.

- The FCS spent fuel rack fuel cells in the first row adjacent to the cask loading area are assumed to be empty during cask loading operations.

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

- CASMO cases assumed a control rod was inserted for part of the depletion to maximize k_∞

- Water density was at optimum moderator density of 1.00 gram/cc corresponding to 4°C.

- All fuel rods were assumed to be filled with fresh water in the pellet/clad gap for both normal and accident conditions.

- All cases assume full DSC reflection in the radial direction.

- Only 90% credit is taken for the B-10 in the neutron poison plates.

- Only a minor change to the Transnuclear Type A cask model was applied. The Transnuclear model assumes reflection on all sides. The model used here has been changed to place 20 cm of water at the top and bottom of the Type 'A' basket.

- Poison rod assemblies (PRAs) were not modeled. Moderator was assumed where the PRAs would be located. This is conservative because the Part 72 requires the use of PRAs in the Type B and C baskets and PRAs add negative reactivity to the system.

4.1.3.4 Cask Manufacturing and Assembly Tolerances

The bounding geometrical conditions for the NUHOMS®-32PT DSC and fuel assemblies were determined by Transnuclear for 10 CFR 72 licensing, based on various design and manufacturing tolerances for the components. The most reactive system configuration was used for this criticality evaluation. However, the Part 72 evaluation was performed assuming soluble boron and no burnup credit. Because the Part 50 criticality evaluation for normal conditions was performed with fresh water, the moderator density and fuel assembly spacing assumptions were re-evaluated to confirm the most reactive configuration was being used.

4.1.3.5 Fuel Assembly Position

The fuel assembly position evaluation for the Type A basket was performed by considering the most reactive Type A basket configuration and performing three cases; namely, an off-set case, the centered case, and a symmetric offset case. The off-set toward the center configuration proved to be slightly more reactive than the other two. The models and results are shown in the calculation in Enclosure 1.

The most reactive system water density changed from an interspersed moderator density (IMD) of about 0.8 in soluble boron to an IMD of 1.0 in fresh water. The most reactive fuel assembly position is the same as in the soluble boron case. The position corresponds to an off-set toward the center of the cask. These two conditions were combined to perform the criticality analysis.

4.1.3.6 DSC Position in the Spent Fuel Pool Cavity

The 32PT DSC and OS197L transfer cask assemblage is assumed to be placed diagonally in the FCS spent fuel pool cask loading pit. That is, the fuel basket walls are oriented at 45-degree angles with respect to the adjacent spent fuel pool rack walls (see Figure 6-2 in Enclosure 1). This orientation is required in order to provide clearance between the transfer cask lifting trunnions and the spent fuel racks. In this orientation, the closest the transfer cask can approach the spent fuel racks is approximately three inches. This value is used in the criticality analyses. This proximity is achieved assuming the cask crane lifting hooks make physical contact with the spent fuel racks. In reality, the lifting hooks will not contact the racks and the transfer cask-to-rack distance will be greater than three inches.

Neutronic coupling between the fuel in the DSC and the fuel in the spent fuel racks was evaluated assuming the first row of fuel cells in each rack adjacent to the cask loading area is empty and a reflective boundary condition was applied to the DSC in the radial direction. This empty row condition will be ensured via procedural controls for spent fuel cask loading (see Attachment 4). The full reflection assumption is equivalent to assuming a second DSC located where the Region 2 racks are located. This is a conservative approach because the DSC geometry is more reactive than the Region 2 spent fuel racks.

4.1.3.7 Accident Condition – Mis-Loaded Fuel Assembly

The mis-loaded fuel assembly accident analysis was performed by assuming the DSC is loaded with the most reactive spent fuel combination of burnup and enrichment in borated water when the single remaining empty position is loaded with a fresh fuel assembly of 4.5 w/o ^{235}U enrichment. Multiple empty fuel storage locations were evaluated to locate the most reactive empty cell. The soluble boron concentration values evaluated ranged from 500 to 800 ppm. All analysis assumptions from the normal case were also applied to this analysis; namely, all fuel rod gaps are flooded with pure water and the cask is fully reflected. The sensitivity evaluations indicate that the most reactive fuel cell location into which a fresh fuel assembly should be mis-loaded to establish the bounding case for this event are on the periphery of the fuel basket. This is expected because there are no neutron absorber plates on the outside walls of the peripheral fuel storage cells.

4.1.3.8 Accident Condition – Fuel Assembly Drop

The fuel assembly drop accident assumes that a fresh fuel assembly of 4.5 w/o ^{235}U is dropped in the space between the transfer cask and the spent fuel racks during loading operations. This event was evaluated assuming the cask is filled with the most reactive spent fuel combination of burnup and enrichment and over the same range of soluble boron concentration as the mis-loading event. The dropped fuel assembly was modeled in the upright position with the active fuel regions of the assemblies in the DSC and the dropped assembly exactly matched in the axial direction. In other words, the dropped assembly was modeled analogous to locating the assembly in an extra peripheral fuel cell location (see Enclosure 1, Figure 6-4). The dropped assembly was evaluated at various azimuthal locations around the transfer cask to determine the most reactive position. The zero degree position (cask radial centerline) was determined to be the most reactive position.

4.1.3.9 Criticality Analysis Results

The results of the criticality analysis show that the spent fuel cask system remains sufficiently subcritical under all normal and applicable accident conditions in the FCS licensing basis as shown in the table below. The criticality analysis results are shown in Table 4.1-1 for the most reactive cases, including uncertainty and bias. Results from all cases analyzed may be found in Enclosure 1.

Table 4.1-1
Criticality Analysis Results Summary*

Condition	Reactivity Acceptance Criterion (k_{eff})	Maximum Calculated k_{eff}	Minimum Required Soluble Boron Concentration (ppm)
Normal	1.0	0.99713	0.0
Normal	0.95	0.92575	500
Accident – Fuel Assembly Mis-loading	0.95	0.94035	800
Accident – Fuel assembly drop between cask and spent fuel racks	0.95	0.91255	800

* Fuel assemblies must have burnup greater than or equal to that shown in Table 4.1-2 for the applicable initial average enrichment. The mis-loaded and dropped assemblies are assumed to be fresh fuel at 4.5 w/o enrichment.

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, all FCS fuel assemblies loaded into the 32PT DSC must have a minimum average assembly burnup greater than or equal to the value shown in Table 4.1-2.

Table 4.1-2
Analysis of Burnup and Enrichment for 32PT DSC Wet Loading of FCS Fuel

Maximum Initial Fuel Enrichment (w/o ^{235}U)	Minimum Average Assembly Burnup (MWD/MTU)
1.65	0
2.5	12,180
3.0	18,340
3.5	24,110
3.9	28,670
4.55	38,220

5.0 REGULATORY ANALYSIS

FCS Current Licensing Basis for Criticality Monitoring (10 CFR 70.24)

OPPD requested an exemption from the requirements of 10 CFR 70.24 for Fort Calhoun Station (FCS) on August 29, 1997. In response, the NRC granted the exemption by letter dated February 6, 1998 (Reference 7.8). Section III of the exemption states, "The basis for the staff to determine that inadvertent or accidental criticality is extremely unlikely can be established through compliance with the FCS Technical Specifications, the geometric spacing of the fuel assemblies in the new fuel storage racks and spent fuel storage pool, and administrative controls imposed on fuel handling procedures." The exemption addresses both dry storage of new fuel assemblies in the new fuel storage racks as well as wet storage of new and spent fuel in the spent fuel storage pool racks. New fuel enrichment up to the current permitted value of 4.5 w/o ^{235}U is addressed. The last paragraph of Section III of the exemption states "The low probability of an inadvertent criticality, together with the licensee's adherence to General Design Criterion 63, constitutes good cause for granting an exemption to the requirements of 10 CFR 70.24."

By letter dated July 30, 1996 the NRC issued Amendment 174 to the FCS operating license, which included changes to the Technical Specifications for Refueling Operations to incorporate operating controls and limits associated with increasing the enrichment of fuel to be used in the FCS reactor in the plant. In December, 1988, Amendment 188 to the FCS operating license was issued to re-format several LCOs, including those revised under Amendment 174. Current TS LCOs 2.8.3(1) requires minimum burnup for spent fuel assemblies of certain enrichments stored in the Region 2 racks. Additionally, TS LCO 2.8.3(3) requires that spent fuel pool boron concentration be maintained greater than or equal to 500 ppmb when unirradiated fuel assemblies are stored in the pool (i.e., when new fuel is received and stored prior to a refueling outage) both of which were approved in Amendment 174. Since Amendment 188 was granted, there have been no spent fuel storage-related license amendment requests for FCS, nor has OPPD voluntarily chosen to internally modify its licensing basis to comply with 10 CFR 50.68 rather than 10 CFR 70.24.

Based on the above, the need for an exemption from §70.24 continued following the issuance of Amendments 174 and 188 to the FCS operating license. The bases for granting the exemption to 10 CFR 70.24 continue to be met by the design features and administrative controls discussed in the Safety Evaluation Report (SER) for the exemption. The exemption does not include an expiration date or any other "sunset" clause, and the 10 CFR 50.68 rule did not invalidate existing exemptions to §70.24 when it became effective. Furthermore, neither the NRC SERs associated with Amendments 174 and 188, nor subsequent docketed correspondence from the NRC, notified OPPD that the FCS exemption from the requirements of §70.24 has been withdrawn or otherwise invalidated.

OPPD continues to maintain the Technical Specifications, geometric spacing of fuel assemblies, and administrative controls imposed on fuel handling procedures to preclude inadvertent or accidental criticality in accordance with the exemption granted from the requirements of 10 CFR 70.24. Therefore, there is no commitment to 10 CFR 50.68 in the FCS current licensing basis for new and spent fuel storage and the exemption from 10 CFR 70.24 is still effective.

This LAR pertains exclusively to criticality control during spent fuel cask loading operations in the FCS spent fuel pool. Appropriate consideration has been given to neutronic coupling between the fuel in the cask and the fuel in the adjacent spent fuel storage racks. However, the design and licensing basis for the spent fuel and new fuel storage racks remain unchanged by this LAR. That is, the existing exemption to §70.24 is necessary and still applies to the spent fuel pool and new fuel storage racks.

The acceptance criteria in 10 CFR 50.68 were chosen for use in the safety analyses supporting this LAR based on the recommendations in RIS 2005-05, in lieu of using acceptance criteria in the FCS current licensing basis. This does not mean that OPPD now commits to, an across-the-board change in its licensing basis from §70.24 to §50.68 for all (new and spent) fuel storage. Such a choice remains optional in accordance with 10 CFR 50.68(a) and would involve a detailed evaluation of the costs and benefits by OPPD before such a commitment could be made. That effort is not part of this licensing action.

<u>FCS Licensing Basis for this LAR</u>

The technical analyses described in Section 4.0 satisfy all applicable 10 CFR 50 regulatory requirements and guidance concerning criticality control during spent fuel cask loading operations in the spent fuel pool within the existing FCS licensing basis. Normal conditions and appropriate accident conditions have been evaluated within the context of the double contingency principle. The spent fuel pool soluble boron and burnup versus enrichment limits applicable to spent fuel cask loading in proposed new LCO 2.8.3(6) must be verified to be met prior to loading fuel assemblies into the spent fuel cask in the spent fuel pool. Any change to these limits would require a license amendment. Boron concentration is verified before loading operations begin and every 48 hours thereafter as long as a spent fuel cask containing fuel is submerged in the spent fuel pool. The minimum burnup versus enrichment is verified to be met for each and every fuel assembly prior to loading into the spent fuel cask.

The proposed changes are limited in scope to refueling operations involving a spent fuel cask in the spent fuel pool. They do not apply to refueling operations in containment or those evolutions involving only the fuel in the spent fuel storage racks or new fuel storage racks.

5.1 No Significant Hazards Consideration

OPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

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

   Response: No.

   These proposed changes affect only operations in the spent fuel pool during spent fuel cask loading operations. Plant power operations and other spent fuel pool operations are not affected. There are no changes to the design or operation of the power plant that could affect system, component or accident functions resulting from these changes.

   Fuel loading into the spent fuel casks in the spent fuel pool will not require any significant changes to spent fuel pool structures, systems, or components, nor will their performance requirements be altered. The potential to handle a spent fuel cask was considered in the original design of the plant. Therefore, the response of the plant to previously analyzed Part 50 accidents and related radiological releases will not be adversely impacted, and will bound those postulated during cask loading activities in the cask loading area.

   Accordingly, the proposed changes do 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.

   These proposed changes affect only operations in the spent fuel pool during spent fuel cask loading operations. Plant power operations and other spent fuel pool operations are not affected. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for mitigation of an event remain capable of fulfilling their intended design function with these changes to the TS.

   Fuel handling procedures and associated administrative controls for movement of spent fuel in the spent fuel pool remain applicable and are being appropriately augmented to accommodate spent fuel cask loading operations. Additionally, the soluble boron concentration required to maintain $k_{eff} \leq 0.95$ for postulated accidents associated with cask loading operations was also evaluated. The results of the analyses, using a methodology previously approved by the NRC, demonstrate that the amount of soluble boron assumed to be in the pool water during these postulated accidents (800 ppm) is much less than the value at which the spent fuel pool is normally maintained (approximately 1900 ppm).

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   Response: No.

   An NRC-approved methodology was used to perform the criticality analyses that provide the basis to incorporate a boron concentration and a new burnup versus enrichment curve into the plant Technical Specifications to ensure criticality safety margins are maintained during spent fuel cask loading. Spent fuel casks at FCS are loaded in the spent fuel pool in an area adjacent to the spent fuel racks. No physical segregation such as a wall or gate exists between the spent fuel racks and spent fuel cask loading area. The cask loading area floor is approximately two feet lower than the floor on which the spent fuel racks are located. Therefore, the spent fuel pool water flows in and around the spent fuel racks and spent fuel casks being loaded in a common pool. Neutronic coupling between fuel in the spent fuel racks and fuel in the spent fuel cask has been appropriately considered in the criticality analysis, including accident events that postulate mis-loading of a fresh fuel assembly into the cask and dropping a fuel assembly between the spent fuel racks and spent fuel cask during loading.

   The normal condition criticality analysis was performed assuming no soluble boron in the spent fuel pool water and credit for fuel burnup. The proposed new Technical Specification requirement to permit only fuel assemblies with the minimum required burnup versus enrichment to be loaded into the spent fuel cask preserves this analysis basis. The accident condition criticality analysis was performed assuming a minimum of 800 ppm boron in the spent fuel pool during cask loading operations. All analyses account for uncertainties at a 95 percent probability/95-percent confidence level. The proposed new Technical Specification requirement to maintain a minimum boron concentration of 800 ppm in the spent fuel pool during spent fuel cask loading operations preserves this analysis basis. For defense-in-depth, the spent fuel pool boron concentration is typically maintained at approximately 1900 ppm during normal operations and would not be expected to be reduced during spent fuel cask loading operations.

   Therefore, there is no significant reduction in a margin of safety as a result of this change.

<u>Conclusion</u>

Operation of FCS in accordance with the proposed amendment will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident than previously analyzed; and will not result in a significant reduction in a margin of safety.

Based on the considerations discussed above, OPPD concludes that the proposed license amendment adding controls for spent fuel cask loading in the spent fuel pool presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c). Accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements /Criteria

5.2.1 Regulations

The proposed changes to the FCS TS described in this request comply with 10 CFR 50.68 and 10 CFR 50 Appendix A, General Design Criterion 62. As discussed in Section 5.0 above, this licensing action does not invalidate FCS's exemption from 10 CFR 70.24 for new and spent fuel storage.

The spent fuel in the in the spent fuel storage cask has been shown by analysis to be in compliance with 10 CFR 50.68(b)(1) while submerged in the spent fuel pool by analysis demonstrating that a TN 32PT DSC containing up to 32 spent fuel assemblies in the spent fuel pool will be safely subcritical under the most adverse moderator conditions considering the acceptance criteria in 10 CFR 50.68(b)(4). Accident events have been analyzed involving (1) the mis-loading of a fuel assembly not meeting the minimum burnup limit for its enrichment into the cask (up to and including a fresh fuel assembly), and (2) dropping a fresh fuel assembly between the cask and spent fuel racks; and the cask system was found to be safely subcritical ($k_{eff} < 0.95$) assuming a minimum of 800 ppm boron in the spent fuel pool water. This analysis basis is preserved by proposed new TS 2.8.3(6), "Spent Fuel Cask Loading," which requires a minimum boron concentration in the spent fuel pool during cask loading and requires a minimum burnup for each fuel assembly loaded into the cask.

5.2.2 Design Basis (USAR)

There are no accident analyses in USAR Chapter 14 that are applicable to this LAR. The accident events evaluated in support of this LAR are consistent with the "double contingency principle" for criticality events and with Spent Fuel Storage and Fuel Pool Cooling design basis as described in Reference 7.10.

5.2.3 Approved Methodologies

The methodologies used in the criticality analyses supporting this LAR are consistent with the licensing and design bases for Fort Calhoun Station as described in the USAR and plant design basis documents. These methodologies were also used on similar licensing actions from other nuclear plant operators (see Section 4.1.4.2 of this attachment). The methodological approach is also consistent with the guidance in RIS 2005-05 and the internal NRC memorandum from Laurence Kopp dated August 19, 1998 (Reference 7.11). The computer codes used (KENO and

CASMO) are well-recognized, commonly used for these types of nuclear analyses, and their use has been previously reviewed and approved by the NRC.

5.2.4 Analysis

The criticality analysis supporting this LAR demonstrates that a loaded spent fuel cask in the FCS spent fuel pool will remain safely subcritical under all normal and credible accident conditions in accordance with 10 CFR 50.68(b)(1) in accordance with the acceptance criteria of 10 CFR 50.68(b)(4).

5.2.5 Conclusion

Based on the considerations discussed above, OPPD concludes the following: (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 of the United States.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment does not involve and will not result in a condition which significantly alters the impact of Fort Calhoun Station on the environment.

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51.22(b), no environmental assessment needs to be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 NRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," March, 2005.

7.2 Letter from Southern Company (L.M. Stinson) to NRC, "Joseph M. Farley Technical Specifications Revision Spent Fuel Cask Loading Requirements," Docket Nos. 50-348 and 50-364, dated May 17, 2005.

7.3 Letter from Entergy Operations (J.S. Forbes) to NRC, "License Amendment Request To Add Cask Loading Restrictions, Arkansas Nuclear One, Unit 2," Docket No. 50-368, dated July 21, 2005.

7.4 Letter from Southern Nuclear (L.M. Stinson) to NRC, "Response to Draft Request for Additional Information," Docket Nos. 50-348 and 50-364, dated June 13, 2005.

7.5 Letter from Entergy Operations to NRC, "Supplement to License Amendment Request for Cask Loading Restrictions, Arkansas Nuclear One, Unit 2, Docket 50-368, dated August 26, 2005.

7.6 10 CFR 50.68, "Criticality Accident Requirements."

7.7 For Calhoun Station Design Basis Document SDBD-AC-SFP-102, "Spent Fuel Storage and Fuel Pool Cooling," Revision 14.

7.8 Letter from NRC (L.R. Wharton) to OPPD (S.K. Gambhir), "Issuance of Exemption from the Requirements of 10 CFR 70.24 Concerning Criticality Monitors," dated February 6, 1998.

7.9 NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January, 1997.

7.10 OPPD Design Basis Document SBDB-AC-SFP-102, "Spent Fuel Pool Storage and Fuel Pool Cooling," Revision 14.

7.11 U.S. NRC Memorandum from Laurence Kopp to Timothy Collins "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," August 19, 1998.

Attachment 2

Markup of
Technical Specification Pages

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(6) Spent Fuel Cask Loading

Applicability

Applies to storage of spent fuel assemblies whenever any fuel assembly is located in a spent fuel cask in the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

To minimize the possibility of an accident occurring during REFUELING OPERATIONS that could affect public health and safety.

Specification

(1) The spent fuel pool boron concentration shall be ≥800 ppm, and

(2) The combination of initial enrichment and burnup of each spent fuel assembly located in a spent fuel storage cask in the spent fuel pool shall be within the acceptable burnup domain of Figure 2-11.

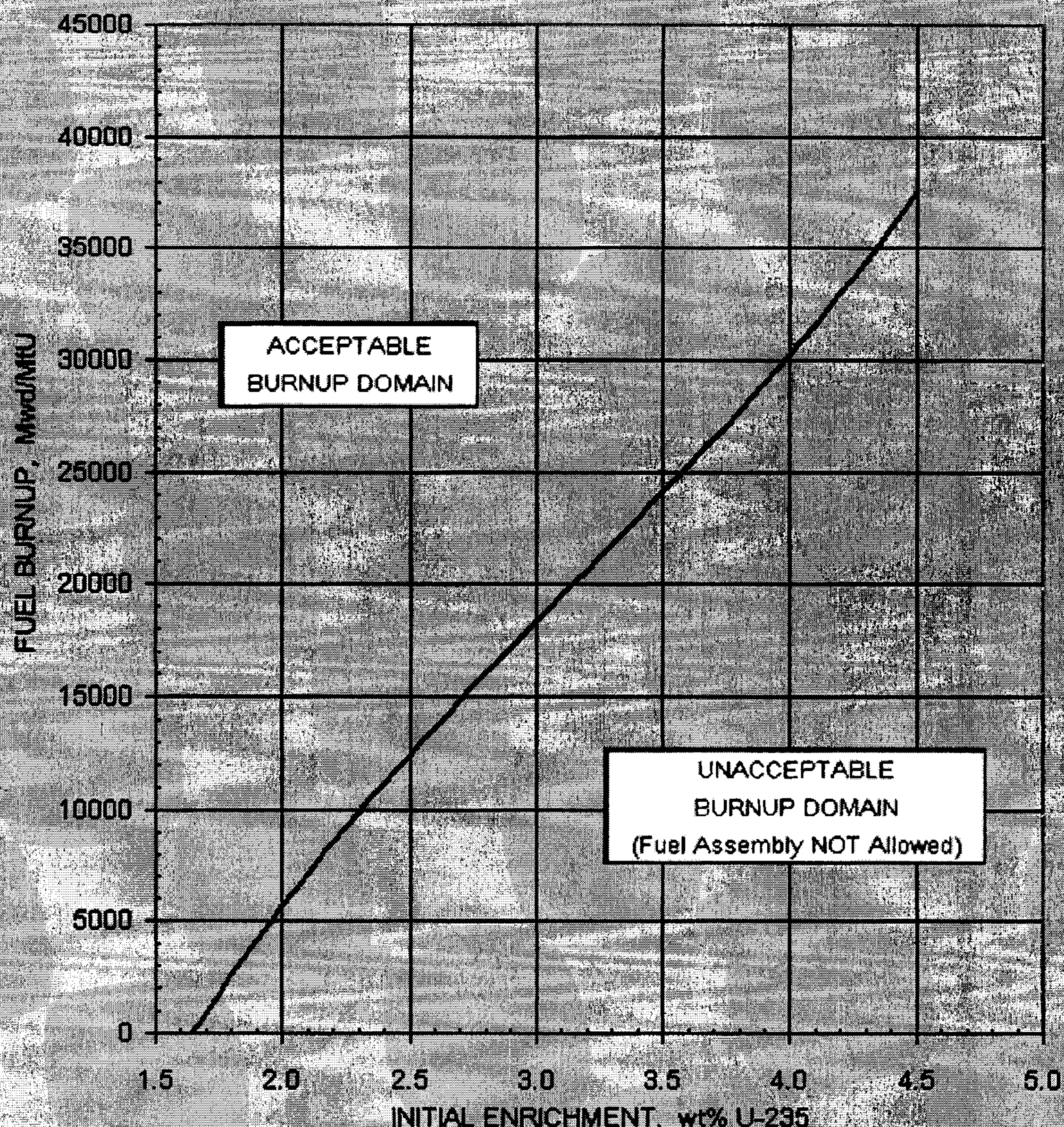
Required Actions

(1) With the spent fuel pool boron concentration <800 ppm, suspend REFUELING OPERATIONS involving spent fuel cask loading immediately, and

(2) Restore spent fuel pool boron concentration to ≥800 ppm immediately.

(3) With the requirements of the LCO 2.8.3(6)(2) not met, initiate action to remove the noncomplying fuel assembly from the spent fuel cask immediately.

Figure 2-11



LIMITING BURNUP CRITERIA
FOR
ACCEPTABLE STORAGE IN
SPENT FUEL CASK

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling

Bases (Continued)

2.8.3(6) Spent Fuel Cask Loading

(1) Soluble Boron

The basis for the 800 ppm minimum boron concentration requirement during spent fuel cask loading operations is to maintain the k_{eff} in the cask system less than or equal to 0.95 in the event a mis-loaded unirradiated fuel assembly is located anywhere in the cask with up to 31 other fuel assemblies meeting the burnup and enrichment requirements of LCO 2.8.3(6)(2). This boron concentration also ensures the k_{eff} in the cask system will be less than or equal to 0.95 if an unirradiated fuel assembly is dropped in the space between the spent fuel racks and the cask loading area during cask loading operations next to a spent fuel assembly. A mis-loaded or dropped unirradiated fuel assembly at maximum enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. Soluble boron in the spent fuel pool water, for which credit is permitted during spent fuel cask loading operations, assures that the effective multiplication factor is maintained substantially less than the design basis limit.

This LCO applies whenever a fuel assembly is located in a spent fuel cask submerged in the spent fuel pool. The boron concentration is periodically sampled in accordance with Specification 3.2. Sampling is performed prior to movement of fuel into the spent fuel cask and periodically thereafter during cask loading operations, until the cask is removed from the spent fuel pool.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of refueling operations shall not preclude completion of movement of a component to a safe, conservative position.

2.0 LIMITING CONDITIONS FOR OPERATION
2.8 Refueling

Bases (Continued)

2.8.3(6) Spent Fuel Cask Loading (Continued)

(2) Burnup vs. Enrichment

The spent fuel cask is designed for subcriticality by use of neutron absorbing material. The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 2-11, and the accompanying LCO, ensure that the k_{eff} of the spent fuel pool always remains ≤ 0.95 assuming the pool to be flooded with borated water and <1.0 assuming the pool is flooded with unborated water, in accordance with 10 CFR 50.68(b)(4).

A spent fuel assembly may be transferred directly from the spent fuel racks to the spent fuel cask provided an independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Figure 2-11. If any fuel assembly located in the spent fuel cask is not in accordance with Figure 2-11, immediate action must be taken to make the remove the non-complying fuel assembly from the spent fuel cask and return it to the spent fuel rack.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

TABLE 3-4 (Continued)

MINIMUM FREQUENCIES FOR SAMPLING TESTS

		Type of Measurement and Analysis	Sample and Analysis Frequency
1.	Reactor Coolant (Continued)		
	(c) Cold Shutdown (Operating Mode 4)	(1) Chloride	1 per 3 days
	(d) Refueling Shutdown (Operating Mode 5)	(1) Chloride (2) Boron Concentration	1 per 3 days(3) 1 per 3 days(3)
	(e) Refueling Operation	(1) Chloride (2) Boron Concentration	1 per 3 days(3) 1 per 3 days(3)
2.	SIRW Tank	Boron Concentration	M
3.	Concentrated Boric Acid Tanks	Boron Concentration	W
4.	SI Tanks	Boron Concentration	M
5.	Spent Fuel Pool	Boron Concentration	See Footnote 4 below
6.	Steam Generator Blowdown (Operating Modes 1 and 2)	Isotopic Analysis for Dose Equivalent I-131	W(5)

(1) Until the radioactivity of the reactor coolant is restored to ≤ 1 μCi/gm DOSE EQUIVALENT I-131.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was subcritical for 48 hours or longer.

(3) Boron and chloride sampling/analyses are not required when the core has been off-loaded. Reinitiate boron and chloride sampling/analyses prior to reloading fuel into the cavity to assure adequate shutdown margin and allowable chloride levels are met.

(4) Prior to placing unirradiated fuel assemblies in the spent fuel pool or placing fuel assemblies in a spent fuel cask in the spent fuel pool, and weekly when unirradiated fuel assemblies are stored in the spent fuel pool, or every 48 hours when fuel assemblies are in a spent fuel storage cask in the spent fuel pool.

(5) When Steam Generator Dose Equivalent I-131 exceeds 50 percent of the limits in Specification 2.20, the sampling and analysis frequency shall be increased to a minimum of 5 times per week. When Steam Generator Dose Equivalent I-131 exceeds 75 percent of this limit, the sampling and analysis frequency shall be increased to a minimum of once per day.

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR Section Reference
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).	
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.	
24.	Spent Fuel Cask Loading	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11.	Prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool.	

4.0 **DESIGN FEATURES**

4.1 Site

The site for Fort Calhoun Station Unit No. 1 is in Washington County, Nebraska, on the west bank of the Missouri River and approximately nineteen miles north, northwest of the city of Omaha, Nebraska. The exclusion area, as defined in 10 CFR Part 100, Section 100.3(a), consists of approximately 1242 acres. The exclusion area boundary extent includes approximately 660 acres in Washington County, Nebraska, owned by the Omaha Public Power District (OPPD), and 582 acres in Harrison County, Iowa, on the east bank of the river directly opposite the facility, on which the District retains perpetual easement rights. The minimum exclusion area boundary point is located approximately at the 187.0 degree radial from the outer wall of the containment building and at a distance of 910 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 133 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Element Assemblies

The reactor core shall contain 49 control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, or hafnium metal as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

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

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

4.0 **DESIGN FEATURES** (Continued)

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North South) center to center distances between fuel assemblies placed in Region 1, the low density fuel storage racks,

e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" may be allowed unrestricted storage in any of the Region 2 fuel storage racks in compliance with Reference (1),

f. Partially spent fuel assemblies with a discharge burnup between the "acceptable domain" and "Peripheral Cells" of Figure 2-10 may be allowed unrestricted storage in the peripheral cells of the Region 2 fuel storage racks in compliance with Reference (1),

g. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 will be stored in Region 1 in compliance with Reference (1).

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference (2).

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Reference (2),

d. A nominal 16 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.3 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,

c. $k_{eff} \leq 0.95$ 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.

Attachment 3

Clean-Typed Technical Specification Pages

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(6) Spent Fuel Cask Loading

Applicability

Applies to storage of spent fuel assemblies whenever any fuel assembly is located in a spent fuel cask in the spent fuel pool. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

To minimize the possibility of an accident occurring during REFUELING OPERATIONS that could affect public health and safety.

Specification

(1) The spent fuel pool boron concentration shall be ≥800 ppm, and

(2) The combination of initial enrichment and burnup of each spent fuel assembly located in a spent fuel storage cask in the spent fuel pool shall be within the acceptable burnup domain of Figure 2-11.

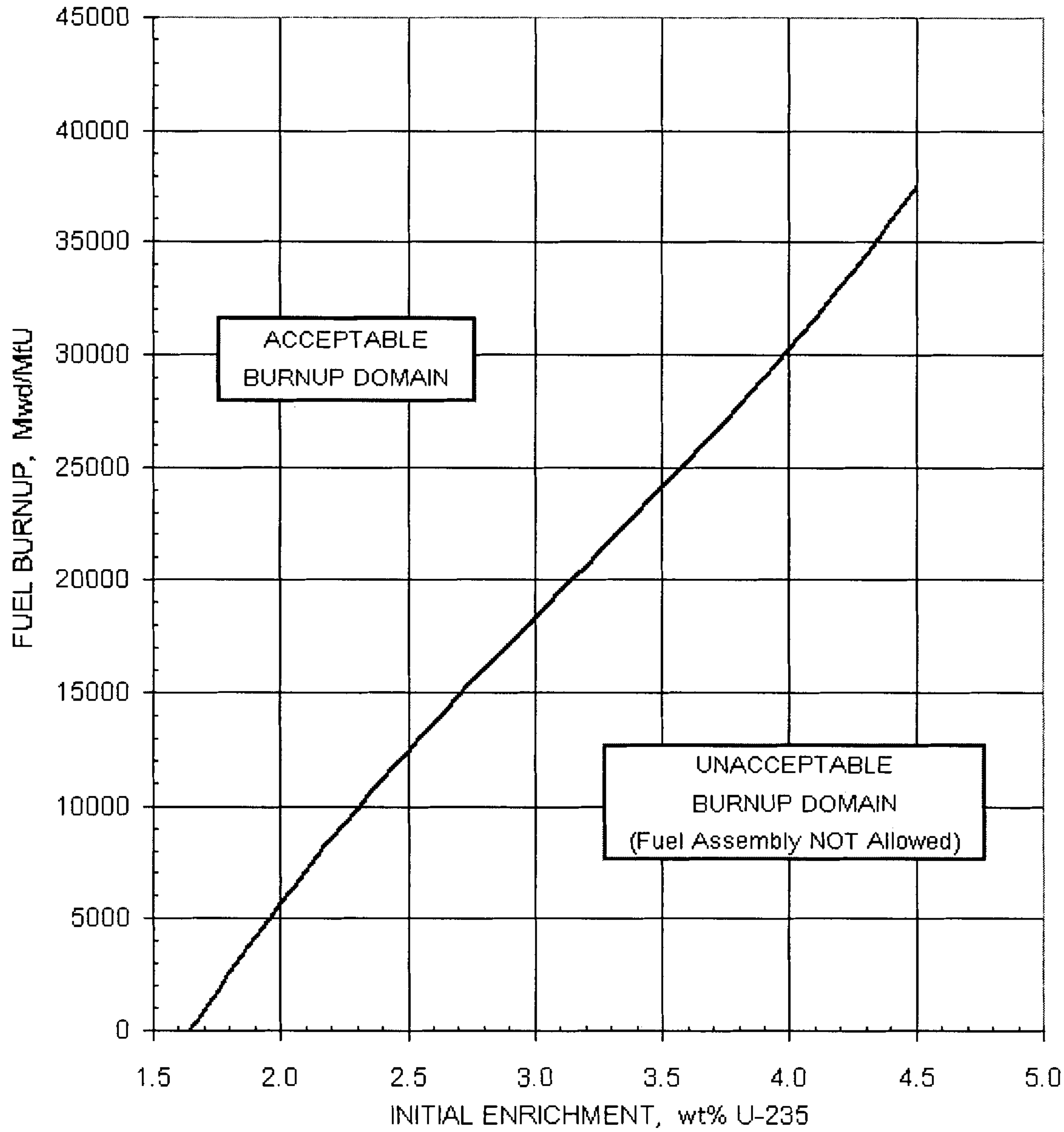
Required Actions

(1) With the spent fuel pool boron concentration <800 ppm, suspend REFUELING OPERATIONS involving spent fuel cask loading immediately, and

(2) Restore spent fuel pool boron concentration to ≥800 ppm immediately.

(3) With the requirements of the LCO 2.8.3(6)(2) not met, initiate action to remove the noncomplying fuel assembly from the spent fuel cask immediately.

Figure 2-11



LIMITING BURNUP CRITERIA
FOR
ACCEPTABLE STORAGE IN
SPENT FUEL CASK

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling

Bases (Continued)

2.8.3(6) Spent Fuel Cask Loading

(1) Soluble Boron

The basis for the 800 ppm minimum boron concentration requirement during spent fuel cask loading operations is to maintain the k_{eff} in the cask system less than or equal to 0.95 in the event a mis-loaded unirradiated fuel assembly is located anywhere in the cask with up to 31 other fuel assemblies meeting the burnup and enrichment requirements of LCO 2.8.3(6)(2). This boron concentration also ensures the k_{eff} in the cask system will be less than or equal to 0.95 if an unirradiated fuel assembly is dropped in the space between the spent fuel racks and the cask loading area during cask loading operations next to a spent fuel assembly. A mis-loaded or dropped unirradiated fuel assembly at maximum enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. Soluble boron in the spent fuel pool water, for which credit is permitted during spent fuel cask loading operations, assures that the effective multiplication factor is maintained substantially less than the design basis limit.

This LCO applies whenever a fuel assembly is located in a spent fuel cask submerged in the spent fuel pool. The boron concentration is periodically sampled in accordance with Specification 3.2. Sampling is performed prior to movement of fuel into the spent fuel cask and periodically thereafter during cask loading operations, until the cask is removed from the spent fuel pool.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of refueling operations shall not preclude completion of movement of a component to a safe, conservative position.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling

Bases (Continued)

2.8.3(6) Spent Fuel Cask Loading (Continued)

(2) Burnup vs. Enrichment

The spent fuel cask is designed for subcriticality by use of neutron absorbing material. The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 2-11, and the accompanying LCO, ensure that the k_{eff} of the spent fuel pool always remains < 0.95 assuming the pool to be flooded with borated water and < 1.0 assuming the pool is flooded with unborated water, in accordance with 10 CFR 50.68(b)(4).

A spent fuel assembly may be transferred directly from the spent fuel racks to the spent fuel cask provided an independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Figure 2-11. If any fuel assembly located in the spent fuel cask is not in accordance with Figure 2-11, immediate action must be taken to make the remove the non-complying fuel assembly from the spent fuel cask and return it to the spent fuel rack.

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

TABLE 3-4 (Continued)

MINIMUM FREQUENCIES FOR SAMPLING TESTS

		Type of Measurement and Analysis	Sample and Analysis Frequency
1.	Reactor Coolant (Continued)		
	(c) Cold Shutdown (Operating Mode 4)	(1) Chloride	1 per 3 days
	(d) Refueling Shutdown (Operating Mode 5)	(1) Chloride (2) Boron Concentration	1 per 3 days[(3)] 1 per 3 days[(3)]
	(e) Refueling Operation	(1) Chloride (2) Boron Concentration	1 per 3 days[(3)] 1 per 3 days[(3)]
2.	SIRW Tank	Boron Concentration	M
3.	Concentrated Boric Acid Tanks	Boron Concentration	W
4.	SI Tanks	Boron Concentration	M
5.	Spent Fuel Pool	Boron Concentration	See Footnote 4 below
6.	Steam Generator Blowdown (Operating Modes 1 and 2)	Isotopic Analysis for Dose Equivalent I-131	W[(5)]

(1) Until the radioactivity of the reactor coolant is restored to ≤1 μCi/gm DOSE EQUIVALENT I-131.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was subcritical for 48 hours or longer.

(3) Boron and chloride sampling/analyses are not required when the core has been off-loaded. Reinitiate boron and chloride sampling/analyses prior to reloading fuel into the cavity to assure adequate shutdown margin and allowable chloride levels are met.

(4) Prior to placing unirradiated fuel assemblies in the spent fuel pool or placing fuel assemblies in a spent fuel cask in the spent fuel pool, and weekly when unirradiated fuel assemblies are stored in the spent fuel pool, or every 48 hours when fuel assemblies are in a spent fuel storage cask in the spent fuel pool.

(5) When Steam Generator Dose Equivalent I-131 exceeds 50 percent of the limits in Specification 2.20, the sampling and analysis frequency shall be increased to a minimum of 5 times per week. When Steam Generator Dose Equivalent I-131 exceeds 75 percent of this limit, the sampling and analysis frequency shall be increased to a minimum of once per day.

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR Section Reference
22.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).	
23.	P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.	
24.	Spent Fuel Cask Loading	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11.	Prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool.	

4.0 DESIGN FEATURES

4.1 Site

The site for Fort Calhoun Station Unit No. 1 is in Washington County, Nebraska, on the west bank of the Missouri River and approximately nineteen miles north, northwest of the city of Omaha, Nebraska. The exclusion area, as defined in 10 CFR Part 100, Section 100.3(a), consists of approximately 1242 acres. The exclusion area boundary extent includes approximately 660 acres in Washington County, Nebraska, owned by the Omaha Public Power District (OPPD), and 582 acres in Harrison County, Iowa, on the east bank of the river directly opposite the facility, on which the District retains perpetual easement rights. The minimum exclusion area boundary point is located approximately at the 187.0 degree radial from the outer wall of the containment building and at a distance of 910 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 133 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Element Assemblies

The reactor core shall contain 49 control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, or hafnium metal as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,
b. k_{eff} $\leq$ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North South) center to center distances between fuel assemblies placed in Region 1, the low density fuel storage racks,

e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" may be allowed unrestricted storage in any of the Region 2 fuel storage racks in compliance with Reference (1),

f. Partially spent fuel assemblies with a discharge burnup between the "acceptable domain" and "Peripheral Cells" of Figure 2-10 may be allowed unrestricted storage in the peripheral cells of the Region 2 fuel storage racks in compliance with Reference (1),

g. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 will be stored in Region 1 in compliance with Reference (1).

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference (2).

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Reference (2),

d. A nominal 16 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.3 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. $k_{eff} \leq 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

c. $k_{eff} \leq 0.95$ if fully flooded with borated water $\geq$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.

Attachment 4

Commitments

List of Regulatory Commitments

The following table identifies the action committed to by OPPD in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding this commitment to Mr. Thomas C. Matthews at 402-533-6938.

REGULATORY COMMITMENT	DUE DATE
Revise procedures to prohibit fuel storage in spent fuel cell locations in the first row of the spent fuel racks adjacent to the spent fuel cask loading area during spent fuel cask loading or unloading operations. (Action Request 38033)	Prior to initial spent fuel cask loading operations in the spent fuel pool

Enclosure 1

Framatome ANP Criticality Analysis

CALCULATION COVER SHEET

Calculation Number: FC07257 ~~gw 10/28/05~~	Page No: 1
QA Category: [ ] CQE [X] Non-CQE [X] LCQE	Total Pages: 42
Calculation Title: NUHOMS -32PT 50.68 Criticality Analysis for Fort Calhoun	Short Term Calc: [ ] Yes [x] No Vendor Calc. No.: 1121-0600 Associated Project::
Software Tracking No.: (from PED-MEI-23, if applicable)	Responsible NED Dept No.: 860
Owner Assignment (by Dept Head): (Required only if there are affected documents to be changed)	
OPPD Engineer Assignment (by Dept Head): (Required only for verification of vendor/contractor calculations)	MATHEW M. PANICKER

Verification of Vendor/Contractor Calc. assumptions, inputs and conclusions complete:

OPPD Engineer: [signature] Date: 10/28/2005

APPROVALS - SIGNATURE AND DATE
(Multiple preparers shall identify section prepared per PED-QP-3, Section 4.3.)

Rev. No.	Preparer(s)	Reviewer(s)	Required for CQE Independent Reviewer(s)	Supersedes Calc No.	Confirmation Required? Yes	No
A	Areva [signature] for	Transnuclear [signature] for	[signature]	0		x

CALCULATION COVER SHEET

Calculation Number: FC07257	Page No.: 2
Applicable System(s) / Tag Number(s)	
EA's and/or Calculations Used as input in this Calculation	

External Organization Distribution (Groups affected by this calculation)

Name and Location	Copy Sent (✓)	Name and Location	Copy Sent (✓)

CALCULATION REVISION SHEET

Calculation No.: FC07257	Page No.: 3
Rev. #	Description/Reason for Change
~~A~~ 0 JW 10/28/05	Initial Issue for ~~EC 32306~~ JW 10/28/05

Calculation No. FC07257

Page No. 4

CALCULATION AFFECTED DOCUMENTS

The Calculation Preparer is to identify documents affected by this Calculation. Markups are to be provided in an Attachment to the Calculation except those noted with an *. Changes not involving procedures should follow the associated change process. The preparer is to indicate below how the Calculation is to be processed by Document Control.	
	Not Required, Calculation supports EC#____ or is used to support EA-FC- - this form can be signed off by the Calculation Preparer. Calculation "As Built" follows direction given for modifications.
X	EC, FLC, Preapproved NRC commitment change, or Condition Report need identified. Calculation is closed on receipt of the completed PED-QP-3.8 form.
	Change to a DBD, USAR, etc., without a change to plant procedures identified. Calculation is "As Built" on receipt of the completed PED-QP-3.8 form.
	Change to a DBD, USAR, etc., and plant procedures (no hardware) identified. Calculation is "As Built" on receipt of the completed PED-QP-3.8 form.
	No document changes or other changes are required. Calculation "As Built" on receipt of the completed PED-QP-3.8 form.
NOTE: Markups are to include any inputs or assumptions which define plant configuration and/or operating practices that must be implemented to make the results of the Calculation valid. The Calculation may provide the basis for a 10CFR50.59 analysis or substantiate a 10CFR50.59 analysis.	

Affected Documents

Document Type	Document Number (N/A = not applicable)	Procedure Change No., FLC No., etc.
Emergency Operating Procedure*	N/A	
Abnormal Operating Procedure*	N/A	
Annunciator Response Procedure	N/A	
Technical Data Book	3	LAR 05-013
Surveillance Test Procedure	5	LAR 05-013
Calibration Procedure	N/A	
Operating Procedure	N/A	
Maintenance Procedure	N/A	

Calculation No. FC07257

Page No. 5

Affected Documents		
Document Type	Document Number (N/A = not applicable)	Procedure Change No., FLC No., etc.
PM Procedure	N/A	
EP/EPIP/RERP*	N/A	
Security Procedures * (Safeguards)*	N/A	
Operating Instructions	N/A	
System Training Manuals	N/A	
Technical Specification*	MHP ~~N/A~~ (1)	LAR 05-13
USAR	MHP ~~N/A~~ (2)	LAR 05-13
Licensing Commitments	N/A	
Standing Order	N/A	
Security Plan (Safeguards)	N/A	
CQE List	N/A	
Vendor Manual Changes	N/A	
Design Basis Documents	~~N/A~~ 4	LAR-05-13
Equipment Database	N/A	
Oil Spill Prevention, Control and Countermeasure (SPCC) Plan	N/A	
EEQ Manual	N/A	
SE-PM-EX-0600	N/A	
Updated Fire Hazard Analysis	N/A	
EPIX	N/A	
Electrical Load Distribution Listing (ELDL)	N/A	
Station Equipment Labeling	N/A	
Engineering Analysis	N/A	

Calculation No. FC07257

Page No. 6

Affected Documents		
Document Type	Document Number (N/A = not applicable)	Procedure Change No., FLC No., etc.
Calculations	N/A	
Drawing Number	N/A	
Drawing Number	N/A	
Other	N/A	
Completed by Owner (if Plant Procedure Changes Required or N/A): ~~N/A~~		Date: 10/28/2008
Completed by Preparer: for AREVA		Date: 10/28/08

(1) TS LCO 2.8.3(b), Table 3-4, Table 3-5, Section 4.3.1

(2) USAR Section 9.5

3 A new TDB Figure may be developed per LAR plan

4 SBDB - AC-SFP-102

5 A new ST will be developed per LAR plan

Control Copy #2
Steve Andersen

TRANSNUCLEAR AN AREVA COMPANY	Form 3.2-1 Calculation Cover Sheet	Calc. No.:	1121-0600		
		Rev. No.:	0		
Calculation Title:		Page:	1	of	36
NUHOMS® -32PT 50.68 Criticality Analysis for Fort Calhoun		Project No.:	1121		
		DCR No.:	N/A		

Project Name: NUHOMS® -32PT for OPPD

Number of CDs attached: 0

If original issue, Is Licensing Review per TIP 3.5 required?

☒ No (explain)	☐ Yes	Licensing Review No._________

This calculation directly incorporates, with minor editorial corrections, the Framatome-ANP calculation that supports a 50.68 criticality analysis of the NUHOMS® -32PT for OPPD. This is a PART 50 calculation and will be utilized to support a site specific license amendment for OPPD. Therefore, a 72.48 review by Transnuclear is not applicable/required.

Software utilized: N/A since this is a documentation of editorial corrections / clarifications.

Calculation is complete

Originator's Signature: A. Prakash	Date: 10/28/2005

Calculation has been checked for consistency, completeness, and correctness

Checker Signature:	Date: 10/28/05

Calculation is approved for use

Project Engineer Signature: for J. Axline	Date: 10/28/05

Form 3 2-1, Revision 0

Revision Summary

Rev. 0	This calculation is prepared to incorporate OPPD comments and clarifications to the Framatome-ANP calculation, 86-90003453-000, "Fort Calhoun NUHOMS -32PT Criticality Analysis." The TN file number for the Framatome-ANP calculation is 1121-0090. Framatome-ANP is an authorized supplier/vendor under the Transnuclear QA (TIPS) program and therefore an independent review of the Framatome calculation is not performed here. The main body of this calculation is exactly the same as the Framatome ANP-calculation except for the changes as noted below. These changes would be incorporated with revision bars such that the nature of the change is clear and unambiguous. 1) Title of Table 6-1 changed to "Maximum Enrichment and Burn-up Results for Type "A" and Type "B" Transfer Cask" in the List of Tables to be consistent with the actual table title. 2) Adjust the table (page 9) after Table 4-1 (before Section 4.2) such that the last row does not spill over to the next page. 3) Table 6-4 (page 26), added "0°" to the last entry in column 1 as it was missing. 4) Section 7.0 (page 29), third bullet, 4.5 w/o was replaced with 4.55 w/o. 5) Section 7.0 (page 29), added an additional bullet after the third bullet that describes a polynomial fit of the fuel burnup as a function of initial enrichment based on the results shown in Table 6-1 and Figure 6-5.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 OBJECTIVE

The purpose of this report is to summarize the results from the criticality analysis that Framatome ANP (Advanced Nuclear Power) performed for loading operations with the Transnuclear, Inc. (TN) NUHOMS®-32PT Dry Shielded Canister (DSC) and transfer cask OS197L designs in the Fort Calhoun spent fuel pool. The license to operate the reactor includes a criticality safety evaluation of the spent fuel pool. The appropriate limits are noted in the plant Technical Specifications [8.1]. However, the Omaha Public Power District (OPPD) plans to improve the operation of Fort Calhoun with an Independent Spent Fuel Storage Installation (ISFSI). This ISFSI will incorporate Transnuclear Inc. (TN) NUHOMS®-32PT DSC to store the spent fuel. TN has an NRC Safety Evaluation Report (SER) to support the licensing of the NUHOMS®-32PT DSC in the Independent Spent Fuel Storage Installation[8.2]. While the Fort Calhoun Technical Specifications and the Transnuclear SER were considered sufficient licensing documents, in March of 2005 the NRC issued Regulatory Issue Summary (RIS) 2005-05 [8.3]. To comply with this RIS, the criticality safety analysis summarized in this report was required.

2.0 SUMMARY OF RESULTS

The TN criticality safety calculations of the NUHOMS®-32PT DSC have been based on bounding "worst case" conditions [8.4, 8.5]. That is, every parameter whose uncertainty could result in an increase in reactivity of the DSC was combined in the criticality model to represent the maximum increase. These bounding modeling conditions were carried over into the Framatome ANP (FANP) analysis of the DSC and transfer cask in the Fort Calhoun spent fuel pool. Moreover, the FANP calculations were benchmarked to previous TN results to show that FANP modeling would produce TN results for the cask criticality evaluations.

The existing Technical Specifications (TS) for the Fort Calhoun spent fuel pool (SFP) allow burned uranium fuel in Region 2 as shown by TS Figure 2-10 [8.1]. FANP used a bounding modeling approach to duplicate the TS Figure 2-10 results. This benchmark demonstrated that the FANP modeling approach was consistent with the criticality safety analysis results that are the basis for the Technical Specifications. That is, using bounding conditions for every parameter whose uncertainty could result in a reactivity increase, and bounding the uncertainty bias from the model benchmarks, the FANP criticality safety analysis results confirmed the validity of the Fort Calhoun spent fuel pool Technical Specifications. Consequently, the FANP modeling approach, with bounding parameters that produce the maximum k_{eff} (effective neutron multiplication factor), is consistent with the licensing of the spent fuel pool, and the licensing of the TN NUHOMS®-32PT DSC.

The FANP modeling of the NUHOMS®-32PT DSC in the spent fuel pool with bounding conditions was used to perform the criticality safety analysis for the canister loading operation[8.6]. The analysis considered normal and accident conditions. The normal condition analysis was performed with burnup credit in lieu of credit for the soluble boron in the spent fuel pool. The analysis determined the required fuel assembly burnup as a function of initial enrichment and equilibrium decay that maintains $k_{eff} < 1$ for the NUHOMS®-32PT Transportable DSC assembly. The accident events considered were the misloading and dropping of a fresh fuel assembly of the highest enrichment permitted by facility Technical Specifications. The accident analysis determined the soluble boron concentration in the spent fuel pool of 800 ppm is required to maintain $k_{eff} < 0.95$ for the DSC assembly. All the cases apply burnup credit.

3.0 BACKGROUND

The Nuclear Regulatory Commission issued Regulatory Issue Summary (RIS) 2005-05 on March 23, 2005 regarding criticality analyses for spent fuel pools and independent spent fuel storage installations[8.3]. The NRC identified regulatory inconsistencies in licensee methodologies for criticality analyses and concluded that dry cask operations performed in the spent fuel pool (SFP) must meet both 10 CFR Part 72 and Part 50 requirements.

Transnuclear, Inc., an AREVA and Siemens Company, contracted with Framatome ANP, also an AREVA and Siemens company to perform the 10 CFR Part 50 criticality analyses for the NUHOMS®-32PT transportable dry shielded canister (DSC) and transfer cask OS197L in the OPPD Fort Calhoun SFP. The criticality analysis for the NUHOMS®-32PT transportable DSC and transfer cask OS197L is documented in Framatome-ANP calculations 32-9003495-000 [8.6] and 32-9001685-00 [8.7]. This report summarizes the reference calculations.

4.0 METHODOLOGY

The methodology applied in the criticality analysis for the NUHOMS®-32PT transportable DSC and transfer cask OS197L documented in References 8.6 and 8.7 is the same methodology that was previously applied for the Fort Calhoun spent fuel pool criticality analysis [8.8, 8.9] and is based on the CASMO-3 and KENO V.a codes. This analysis is also consistent with previous Framatome-ANP analyses through the use of methods and benchmarks that have been previously reviewed and approved by the NRC for other utilities. The most recent instances where the Framatome ANP methodologies have been submitted for review include:

ADAMS Accession ID	Date	Plant/Docket	Description
ML052510504	08/31/05	Shearon Harris/ 50-400	Framatome-ANP Report 77-5069740-NP-00
ML052510502	09/01/05		License Amendment Request

Other previous submittals include the following:

USNRC Docket No. 50-305, "Kewaunee Fresh Storage and Spent Fuel Storage Pool".
USNRC Docket No. 50-346, "Davis Besse Fresh Storage and Spent Fuel Storage Pool".
USNRC Docket No. 50-302, "Crystal River 3 Spent Fuel Storage Pool"
USNRC Docket No. 50-244. 'Ginna Spent Fuel Storage Pool."

The NUHOMS®-32PT DSC assembly KENO V.a criticality analysis is documented in Reference 8.6. The CASMO-3 calculations are documented in Reference 8.7. The following sections of this report summarize these calculations.

4.1 CASMO-3 Calculations

The CASMO-3 (CASMO) calculations provided the isotopic atom densities for the burned 14x14 fuel assembly used in the Fort Calhoun reactor [8.7]. The fuel enrichment evaluated ranged from 2.5 wt% U-235 to 4.75 wt%. The lumped fission product number densities in the CASMO depletion cases can not be used directly by the KENOV.a code because they are not available in the KENOV.a materials library. The lumped fission products 401 and 402 in CASMO have been appropriately modeled in the KENOV.a calculation so that reactivity effects of the lumped fission products are preserved. The method used to convert the CASMO generated isotopic inventory to number densities for use in the KENOV.a calculations was previously developed in Reference 8.8 and demonstrated to be conservative in Reference 8.7.

The CASMO model for the Fort Calhoun fuel was developed using the fuel assembly geometrical information and sample code listings from Reference 8.8. This information is summarized in Table 4-1. The CE type 14x14 fuel assembly was loaded with 96% theoretical density fuel that was reduced to 10.3171 g/cm^3 to account for dishing of the pellets. For conservatism, the fuel assemblies had no axial blankets. CASMO hot full power (HFP) depletions were performed and included the effects of non-uniform axial burnup as well as control rod insertion during operation. The depletion covered specific burnup points where the fuel isotopic inventory was needed for the subsequent KENOV.a calculations.

The fuel assemblies contained sixteen gadolinia bearing fuel rods with 4.0 wt% Gd for the 2.5 and 3.0 wt% enriched fuel. All the higher enrichment fuel assemblies contained sixteen Gadolinia bearing fuel rods with 8.0 wt% Gad rods.

Restart files were saved at the different burnup points where the isotopic inventory was to be calculated. The restart CASMO cases were calculated at the different burnup points of interest at cold conditions with zero ppm soluble boron and no control rods inserted. The control assembly and gadolinia assumptions conservatively overestimate the reactivity associated with operation in the spent fuel assembly.

Calculations were performed to effectively model the reactivity effects of the fission products. Also, the short-lived isotopes were appropriately decayed.

Table 4-1

CE14x14 Fuel Dimensions for CASMO Runs

Dimension	inches	cm
Pellet Diameter	0.377	0.95758
Clad Inner Diameter	0.384	0.97540
Clad Outer Diameter	0.440	1.11760
Pitch	0.580	1.4732
GT/IT Inner Diameter	1.0350	2.62890
GT/IT Outer Diameter	1.1150	2.83210
Array Width based upon pitch	8.12	20.6248
Assembly pitch with Water Channel	8.18	20.7772
Total Fuel Length	128	325.12
Total Fuel Rods in Assembly	176	----
%TD of Stack with Dishing Factor	94.1341 – 10.3171 g/cm^3	
Temperature Information		
HFP Moderator Temperature.	566.35 °F	570 °K
HFP Fuel Temperature	1077.53 °F	854 °K
HFP Moderator Pressure	2100 psia	144.79 bars
HFP Boron	500	---
Cold Moderator Temperature	38.95 °F	277 °K
Cold Fuel Temperature	38.95 °F	277 °K
Cold Moderator Pressure	14.696 psia	1.01325 bars
Cold Boron	0	---

The cold restart case sets several of the short lived isotopes to zero to account for total decay using the CASMO multiplication option on the "CNU" card. The isotopes that are set to zero are the following:

Isotope	CASMO ID
Rh-105	45105
I-135	53135
Xe-135	54135
Pm-148	61148
Pm-148m	61248

4.2 KENO V.a Calculations

4.2.1 KENO V.a Benchmark Cases

The initial KENO models for the Type "A" DSC were provided by TN to FANP along with results from TN calculations [8.4, 8.5]. The FANP KENO is part of the SCALE version 4.4a code package operating on the Linux operating system platform. The TN KENO cases were from a SCALE 4.4 application on a Windows based PC platform. The differences between SCALE 4.4 and SCALE 4.4a have been previously documented in the open literature such as the SCALE newsletter. Nevertheless, a benchmark exercise was performed in order to qualitatively assess the difference, if any, between the two KENO applications. The exact same cases were run at FANP and the resulting $k_{eff} \pm 2\sigma$ was compared to the TN $k_{eff} \pm 2\sigma$. The minor differences observed is due to a combination of the differences in histories and the differences between SCALE version 4.4 and 4.4a. Results of the benchmark are presented in Table 4-2.

Table 4-2
Comparison of TN SCALE 4.4 results with FANP SCALE 4.4a

DESCRIPTION	k_{Keno}^{TN}	$\pm\ \sigma_{Keno}^{TN}$	k_{Keno}^{FANP}	$\pm\ \sigma_{Keno}^{FANP}$
3.8 wt% U-235 0.40% IMD	0.91459	0.00093	0.91131	0.00089
3.8 wt% U-235 0.50% IMD	0.92611	0.00081	0.92333	0.0008
3.8 wt% U-235 0.55% IMD	0.92694	0.00081	0.92518	0.00082
3.8 wt% U-235 0.60% IMD	0.92766	0.00083	0.92529	0.00078
3.8 wt% U-235 0.65% IMD	0.92629	0.00093	0.92373	0.00107
3.8 wt% U-235 0.70% IMD	0.92276	0.00091	0.92106	0.00082
3.8 wt% U-235 0.75% IMD	0.91469	0.00088	0.91569	0.00084
3.8 wt% U-235 0.80% IMD	0.90956	0.00087	0.90726	0.00085
3.8 wt% U-235 0.85% IMD	0.90205	0.00090	0.90048	0.00095
3.8 wt% U-235 0.90% IMD	0.89505	0.00084	0.89488	0.0009
3.8 wt% U-235 0.95% IMD	0.88713	0.00087	0.88886	0.00088
3.8 wt% U-235 1.00% IMD	0.87992	0.00093	0.87915	0.00085
3.9 wt% U-235 0.40% IMD	0.92068	0.00087	0.91855	0.0009
3.9 wt% U-235 0.50% IMD	0.93149	0.00083	0.93124	0.00084
3.9 wt% U-235 0.55% IMD	0.93556	0.00092	0.93239	0.00101
3.9 wt% U-235 0.60% IMD	0.93569	0.00089	0.93202	0.00085
3.9 wt% U-235 0.65% IMD	0.93180	0.00090	0.93178	0.00082
3.9 wt% U-235 0.70% IMD	0.92844	0.00091	0.92741	0.00082
3.9 wt% U-235 0.75% IMD	0.92429	0.00074	0.92308	0.00091
3.9 wt% U-235 0.80% IMD	0.91900	0.00081	0.91682	0.00089
3.9 wt% U-235 0.85% IMD	0.91043	0.00091	0.90821	0.00093
3.9 wt% U-235 0.90% IMD	0.90370	0.00103	0.90325	0.0008
3.9 wt% U-235 0.95% IMD	0.89521	0.00081	0.89543	0.00097
3.9 wt% U-235 1.00% IMD	0.88938	0.00095	0.88645	0.0009

4.2.2 Comparison of FANP SCALE versions 4.4a, 5.0, and 5.0.2

The use of SCALE 4.4a was determined by a number of factors, including consistency with the previous calculation [8.8]. The KENO calculation [8.6] showed that no significant change would have been introduced by the use of either SCALE version 5.0 or 5.0.2. A major difference in SCALE 5.0.2 is the code fix for an error related to cylindrical holes. The error may occur where the boundary of a cylindrical hole in a KENO model overlays the surrounding boundary. Because this is fixed in SCALE version 5.0.2, and there are no significant changes in results then it can be inferred that, even though holes are used in the TN model the usage does not encounter this error. It is also noted that no cylindrical holes occurred in the TN input files. The results from the comparison is presented in Table 4-3.

Table 4-3
Comparison of FANP KENO Output at 3.9 w/o Enrichment

IMD	SCALE4.4a k_{KENO}	σ		SCALE 5.0 k_{KENO}	σ		SCALE 5.0.2 k_{KENO}	σ
100%	0.88645	0.00090		0.88761	0.00078		0.88632	0.00091
95%	0.89543	0.00097		0.89618	0.00085		0.89603	0.00076
90%	0.90325	0.00080		0.90276	0.00089		0.90252	0.00079
85%	0.90821	0.00093		0.90849	0.00086		0.91102	0.00090
80%	0.91682	0.00089		0.91648	0.00080		0.91609	0.00089
75%	0.92308	0.00091		0.92235	0.00123		0.92080	0.00081
70%	0.92741	0.00082		0.92682	0.00089		0.92758	0.00080
65%	0.93178	0.00082		0.93019	0.00082		0.93235	0.00082
60%	0.93202	0.00085		0.93459	0.00095		0.93181	0.00099
55%	0.93239	0.00101		0.93407	0.00083		0.93216	0.00079
50%	0.93124	0.00084		0.93081	0.00087		0.93085	0.00103
40%	0.91855	0.00090		0.91776	0.00091		0.91766	0.00084

4.2.3 Changes to the TN KENO Model

The changes made to the KENO model involved the number of histories selected, the replacement of the lead shield in the transfer cask with stainless steel and removing the soluble boron. These changes are discussed in the following sections.

4.2.3.1 Number of Histories

Illustrated in Table 4-4 are results with 500k, 1000k, 2000k, and 4000k neutron histories. The aforementioned table shows that results are statistically equal for all cases, except for a couple of cases that slightly exceed 2σ. Therefore, cases with ~1000k histories are sufficient to provide acceptable results. For conservatism, FANP elected to use 2500k histories (eg., gen =2600, npg=1000, and nsk=100) in the continuing input decks.

Table 4-4
Comparison of FANP KENO Results for Different Histories

Enrichment	IMD	Original $keff_{KENO}$	σ	1 million $keff_{KENO}$	σ	2 million $keff_{KENO}$	σ	4 million $keff_{KENO}$	σ
3.8 w/o	40%	0.91131	0.00093	0.91149	0.00056	0.91123	0.00040	0.91154	0.00033
3.8 w/o	50%	0.92333	0.00084	0.92445	0.00061	0.92497	0.00044	0.92437	0.00034
3.8 w/o	55%	0.92518	0.00081	0.92646	0.00064	0.92591	0.00042	0.92603	0.00031
3.8 w/o	60%	0.92529	0.00091	0.92554	0.00056	0.92588	0.00041	0.92566	0.00033
3.8 w/o	65%	0.92373	0.00079	0.92349	0.00069	0.92352	0.00041	0.92319	0.00030
3.8 w/o	70%	0.92106	0.00093	0.91981	0.00062	0.91945	0.00043	0.91979	0.00034
3.8 w/o	75%	0.91569	0.00077	0.91584	0.00067	0.91519	0.00044	0.91520	0.00031
3.8 w/o	80%	0.90726	0.00076	0.90861	0.00064	0.90888	0.00043	0.90869	0.00033
3.8 w/o	85%	0.90048	0.00077	0.90159	0.00072	0.90175	0.00048	0.90250	0.00034
3.8 w/o	90%	0.89488	0.00095	0.89456	0.00062	0.89472	0.00042	0.89500	0.00030
3.8 w/o	95%	0.88886	0.00079	0.88736	0.00069	0.88768	0.00047	0.88679	0.00032
3.8 w/o	100%	0.87915	0.00096	0.87932	0.00068	0.87901	0.00047	0.87948	0.00035
3.9 w/o	40%	0.91855	0.00091	0.91865	0.00059	0.91867	0.00043	0.91821	0.00034
3.9 w/o	50%	0.93124	0.00087	0.93149	0.00062	0.93201	0.00043	0.93133	0.00030
3.9 w/o	55%	0.93239	0.00083	0.93314	0.00066	0.93367	0.00045	0.93310	0.00032
3.9 w/o	60%	0.93202	0.00095	0.93225	0.00062	0.93215	0.00043	0.93320	0.00032
3.9 w/o	65%	0.93178	0.00082	0.93148	0.00065	0.93096	0.00043	0.93139	0.00033
3.9 w/o	70%	0.92741	0.00089	0.92719	0.00066	0.92706	0.00045	0.92729	0.00031
3.9 w/o	75%	0.92308	0.00123	0.92271	0.00063	0.92246	0.00042	0.92244	0.00030
3.9 w/o	80%	0.91682	0.00080	0.91727	0.00064	0.91720	0.00043	0.91734	0.00032
3.9 w/o	85%	0.90821	0.00086	0.90951	0.00063	0.90992	0.00044	0.91053	0.00031
3.9 w/o	90%	0.90325	0.00089	0.90319	0.00061	0.90302	0.00043	0.90292	0.00033
3.9 w/o	95%	0.89543	0.00085	0.89585	0.00064	0.89596	0.00042	0.89535	0.00031
3.9 w/o	100%	0.88645	0.00078	0.88610	0.00062	0.88577	0.00046	0.88709	0.00031

4.2.3.2 Removal of Soluble Boron and Gamma Shield

The KENO criticality calculation [8.6] utilized burnup credit in lieu of boron credit to satisfy the criticality safety criterion. This required that with no soluble boron the system remains subcritical ($k_{eff} < 1$). The TN KENO model was altered to remove the soluble boron and the interspersed moderator density (IMD) was adjusted from 5% to 100% to determine the most reactive case. All calculations used 2.5 million histories as previously discussed. Based on the Fort Calhoun Technical Specifications, a fresh fuel assembly with an enrichment of $1.65^{w}/_{o}$ was used as a starting point.

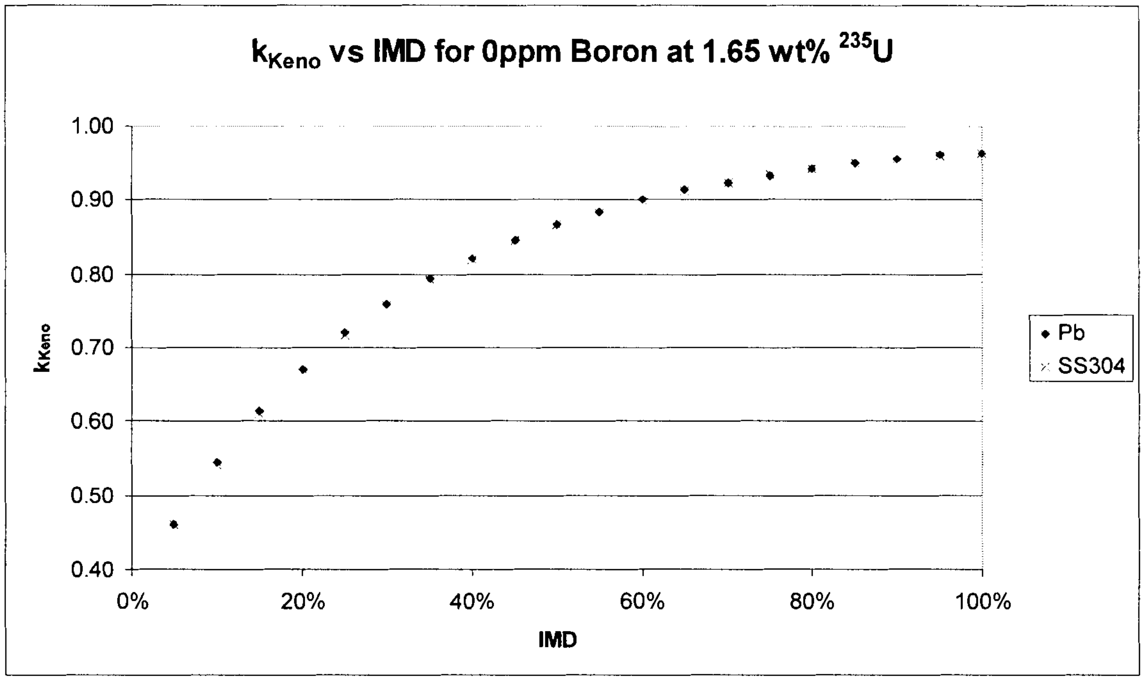
The other modification to the model was the replacement of the lead (Pb) gamma shield in the transfer cask with stainless steel 304 (SS304) as presented by TN for the 'light' transfer cask OS197L [8.10]. Table 4-5 summarizes the changes in results due to the change in shield materials. Figure 4-1 shows the KENO k_{eff} as a function of IMD values from 5% to 100% to determine the most reactive case.

Table 4-5
1.65 w/o Enrichment, Fresh Fuel, Pb and SS304 Shield

IMD	k_{Keno} w/ Pb	σ	k_{Keno} w/ SS304	σ
5%	0.46079	0.00024	0.45935	0.00022
10%	0.54373	0.00028	0.54150	0.00028
15%	0.61306	0.00033	0.61092	0.00030
20%	0.67031	0.00032	0.66978	0.00035
25%	0.71969	0.00035	0.71736	0.00039
30%	0.75922	0.00035	0.75836	0.00035
35%	0.79308	0.00037	0.79205	0.00038
40%	0.82207	0.00039	0.82088	0.00038
45%	0.84652	0.00038	0.84579	0.00040
50%	0.86772	0.00038	0.86672	0.00042
55%	0.88447	0.00044	0.88488	0.00042
60%	0.90022	0.00038	0.89965	0.00041
65%	0.91340	0.00039	0.91278	0.00040
70%	0.92400	0.00042	0.92382	0.00040
75%	0.93297	0.00041	0.93457	0.00038
80%	0.94205	0.00038	0.94159	0.00040
85%	0.94868	0.00042	0.94859	0.00042
90%	0.95430	0.00039	0.95463	0.00040
95%	0.95997	0.00040	0.95947	0.00041
100%	0.96240	0.00038	0.96259	0.00040

Figure 4-1

4.2.4 Benchmark Uncertainties



There are four independent benchmarks that FANP has used to establish the appropriate uncertainties for the criticality safety analysis of the Transnuclear NUHOMS®-32PT DSC in the Fort Calhoun spent fuel pool.

(1) The FANP KENO V.a calculations of the NUHOMS®-32PT DSC were compared to those from Transnuclear. The Transnuclear calculations were independently benchmarked to a set of critical experiments; but more importantly, the Transnuclear calculations contained the worst case uncertainties to produce the maximum k_{eff} value.

(2) The FANP KENO V.a calculations were used to model the Fort Calhoun spent fuel pool. While the criticality safety modeling used to support the Technical Specifications incorporated an independent set of uncertainties, the FANP modeling should meet the licensing criteria for the Technical Specifications in order to demonstrate a consistent set of uncertainties.

(3) The FANP KENO V.a calculations were used for benchmark comparisons to a set of critical measurements from cold experiments with the appropriate canister and spent fuel configurations. The cold experiments included plutonium buildup effects.

(4) The FANP CASMO calculations were used for benchmark comparisons of the Fort Calhoun reactor operation during several reload cycles.

The results from the benchmarks demonstrated the maximum reactivity that is associated with the uncertainties in the methods and models provide bounding results for the criticality safety modeling of the NUHOMS®-32PT DSC. The results from the FANP benchmark comparisons to the Transnuclear KENO V.a calculations of the NUHOMS®-32PT DSC demonstrated that the results were equivalent within the statistical uncertainty associated with 500,000 neutron histories.

The transition from the criticality safety modeling of the NUHOMS®-32PT DSC in the Independent Spent Fuel Storage Installation to the loading of the DSC in the spent fuel pool required changing the boron concentration in the water as well as modeling the burned fuel. The DSC in the Independent Spent Fuel Storage Installation had boron concentrations around 2000 ppm (parts per million boron). However, the spent fuel pool criticality safety requirements are based on no boron. Therefore, the Transnuclear optimization of the highest reactivity conditions were repeated for the 0 ppm cases. The results followed those for the spent fuel pool. The highest density (one gram per cubic centimeter) was the most reactive. This included both fresh and burned fuel. In addition, the assembly location in the canister with high boron concentrations continued to have the most reactive conditions with burned fuel and no boron. The other parameters that Transnuclear considered in the modeling of the canister, such as the minimum borated metal loading in the poison plates, remained optimized for the highest reactivity.

The results from the FANP benchmark comparisons with KENO V.a calculations to the Fort Calhoun Technical Specifications for the spent fuel pool demonstrated that the bounding modeling uncertainties produced the Region 2 Technical Specification results in Figure 2-10.[8.1] The bounding modeling uncertainties not only included the uncertainties associated with the spent fuel pool parameters, but also included the uncertainties associated with critical experiment benchmark comparisons and the modeling of plant operation and burnup for several cycles. Thus, while the FANP burned fuel modeling uncertainties in the spent fuel pool would not be expected to be equivalent to those in the safety analysis for the Technical Specifications, they do produce equivalent criticality safety results.

The FANP modeling approach, with bounding parameters that produce the maximum k_{eff}, is consequently consistent with the licensing of the spent fuel pool, and the licensing of the Transnuclear NUHOMS®-32PT DSC. The KENO V.a results discussed in this document include the bounding uncertainties for the canister and spent fuel pool. The only additional uncertainties that need to be considered are those associated with the methods and those associated with the burned fuel isotopic concentrations.

An important part of the guidance that the NRC and ANSI standards provide concerns the benchmark of the methods used for calculating k_{eff}. The information stresses the importance of having the experimental conditions in the benchmarks essentially the same as those for the fuel and fuel cell models. Moreover, the standards note the requirement that the methods used to analyze the spent fuel pool models must be the same as those used to benchmark the experiments. The FANP criticality analysis to model the loading of the NUHOMS®-32PT DSC in the spent fuel pool includes two sets of independent benchmarks in addition to the

CASMO benchmarks. One set of benchmarks includes experiments of specific configurations that are comparable to the Fort Calhoun spent fuel pool and the DSC.[8.11] The second set includes experiments of fuel assembly configurations that are comparable to the Fort Calhoun fuel assemblies.[8.12]

The KENO V.a benchmark calculations [8.15] modeled one hundred critical experiments that are representative of spent fuel pool and canister configurations. Twenty-one of the one hundred experiments were performed by FANP - B & W and are recommended by the NRC in its "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" [8.14]. This set included the effects of burnup by modeling plutonium fuel in addition to uranium fuel. While the standard recommends taking advantage of the statistical nature of the random uncertainties, the approach utilized for the NUHOMS®-32PT DSC criticality safety analysis was to bound all uncertainties associated with the benchmark results. The bounding uncertainty is:

$$\text{Bounding Benchmark Uncertainty} = 0.01549\ \Delta k_{eff}$$

No calculation of the spent fuel pool - canister experiments gave a higher uncertainty than the above value. Consequently, this value is bounding for subsequent criticality safety evaluations of loading the DSC in the Fort Calhoun spent fuel pool.

The second set of benchmarks is referenced in the "Reactor Analysis System for PWRs" [8.12] report. The fuel design in the first set of benchmarks was not particularly representative of Fort Calhoun, but the second set specifically represented the fuel. Moreover, the first set validated the NITAWL - KENO methods while the second set validated the CASMO methods. The results of the calculations from the second set of benchmarks indicated that the bias and random uncertainties associated with CASMO were smaller than those associated with NITAWL - KENO. That is, no statistically 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 unique uncertainty was assigned to the CASMO methods with fuel assemblies of the Fort Calhoun type. Any k_{eff} that is predicted by CASMO includes the bounding uncertainties from the first set of benchmark results (0.01549 Δk_{eff}).

The FANP CASMO calculations were used for benchmark comparisons of the Fort Calhoun reactor operation during several reload cycles. The uncertainties determined by this benchmark comparison are independent of the bounding conditions that are applied to the burned fuel based on the various tolerances for the components. Thus, the overloading of the uranium, replacing the grids with water, decreasing the burnup, increasing the axial burnup effects by modeling the insertion of control rods, overloading the burnable poisons, etc, are not included in this benchmark comparison.

The CASMO benchmark discussed in Reference 8.12 above was based on cold-clean critical experiments. However, the burnup effects of the fuel need to be benchmarked if burnup credit is to be utilized in the spent fuel pool loading of the canister. This set of benchmarks, in the "Ft. Calhoun PRISM Benchmarking Cycles 17-20" [8.13] document, was used to assess CASMO burnup uncertainties. The calculations modeled the fuel in Cycles 17 through 20 and followed the core operation throughout Cycles 17, 18 and 19. The results show a calculation bias that averages -40 ppm boron at the beginning of cycles 17 through 19 and -26 ppm boron at the end. The calculations are more reactive; that is, the calculations require more boron to be critical than is measured. Thus, for applications to criticality safety the CASMO burnup results are considered to be approximately $+.0025\ \Delta\rho$ (delta-rho, $\{\text{delta-}k\}/k$) too reactive. Consequently, they are conservative.

While the uncertainty associated with CASMO burnup benchmarks is conservative, there are uncertainty biases that are applied to ensure bounding criticality safety predictions. The dominant bias is from the axial burnup effects. The burnup credit calculations are performed assuming a uniform burnup profile throughout the active length of the fuel assemblies. The burnup profile for the burned fuel assemblies is not generally uniform due to the axial flux distribution in the core and the neutron leakage from the ends of the fuel assembly. This typically results in a burnup profile that looks resembles a "flattened cosine". The uniform burnup profile assumption 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 difference between the k_{eff} values based on the axial burnup profile and the uniform burnup assumption is what is termed as "axial end-effect" whose magnitude depends on the actual burnup value and the axial burnup profile selected. For the burnup credit calculations, the expected burnup values are such that the assumption of a uniform burnup profile may not be conservative. In other words, an axial end-effect bias needs to be applied to the burnup credit calculations to account for the increase in reactivity due to axial end-effects.

Generic analyses confirm the minor and generally negative reactivity effect of the axially distributed burn-up at values less than about 30,000 MWD/MTU[8.16]. As a result, KENO calculations with less than 30,000 MWD/MTU do not contain an axial bias. The highest burnup evaluated in this effort was 38,200 MWD/MTU. The axial bias uncertainty applied at this burn-up is $+0.013\ \Delta k$[8.16].

Another major contributor to the bounding uncertainty is the bias in the assembly burnup. This bias results from inaccurate predictions of fuel assembly power, core power, and cycle lifetime. The modeling follows NRC guidance with a 5% uncertainty at the lower burnups expressed in terms of MWD/MTU (mega-Watt days per metric ton of uranium). When the fuel has been burned for several cycles, the bounding burnup is represented by 1565 MWD/MTU, or an equivalent of 52 EFPDs (effective full power days) in Fort Calhoun. While the application of these biases ensure conservative results with k_{eff} values that are too high, the benchmark of the burned fuel to the Fort Calhoun spent fuel pool Technical Specifications shows that there is an overall consistency with the existing criticality safety analysis.

Because the approach taken with the criticality safety analysis for loading Transnuclear's NUHOMS®-32PT DSC (canister) in the Fort Calhoun spent fuel pool is to treat uncertainties with bounding – biased values, the uncertainty from the benchmarks confirms that the uncertainties are appropriately bounding.

The additional reactivity value that must be applied to the KENO V.a calculations of the loading model k_{eff} is the bounding bias from the benchmark comparisons. As noted above, this value is 0.01549 Δk_{eff}.

4.2.5 Cask Manufacturing and Assembly Tolerances

The bounding conditions for the NUHOMS®-32PT DSC and fuel assemblies were determined by Transnuclear based on various tolerances for the components [8.4, 8.5]. The most reactive system configuration was used for the present criticality evaluation. However, the Transnuclear Part 72 evaluation was performed assuming soluble boron. Since the Part 50 criticality evaluation is performed with fresh water, the present work re-evaluated the moderator density and fuel assembly spacing assumptions to confirm the most reactive configuration was being used.

4.2.6 Fuel Assembly Position Evaluation

The fuel assembly position evaluation for the Type "A" basket was performed by utilizing the most reactive Type A cask configuration and performing three cases; namely, off-set case, the centered case, and a symmetric offset case. The off-set configuration proved to be slightly more reactive. The models and results are shown on the following pages.

The most reactive system water density changed from an IMD of about 0.8 in soluble boron to an IMD of 1.0 in fresh water. The most reactive fuel assembly position is the same as in the soluble boron case. The position corresponds to an off-set toward the center of the cask. These two conditions were combined to perform the criticality analysis. The results are summarized in Table 4-6.

Table 4-6
System Bias Evaluation Results

Centered with SS304 no Soluble Boron 4.75wt% at 38.96 GWD/MTU		
IMD	k_{KENO}	s_{KENO}
80%	0.93651	0.00039
90%	0.95552	0.00038
100%	0.96911	0.00041
TN Off-set with SS304 no Soluble Boron 4.75wt% at 38.96 GWD/MTU		
80%	0.93913	0.00039
90%	0.95696	0.00039
100%	**0.97105**	**0.00038**
FANP Offset with SS304 no Soluble Boron 4.75 wt% at 38.96 GWD/MTU		
80%	0.93354	0.00041
90%	0.95148	0.00038
100%	0.96522	0.00042

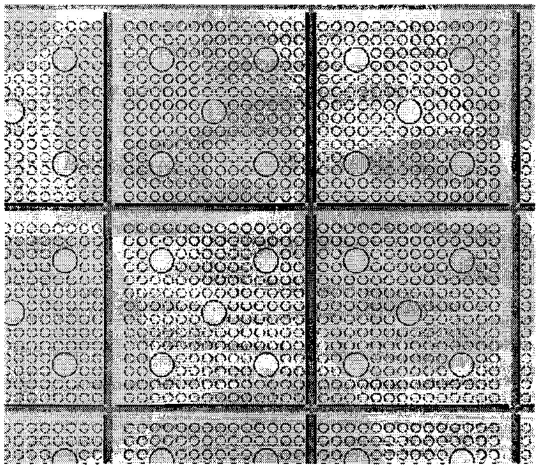


Figure 4-2
Fuel Assembly Position
Off-Set Inward – Model

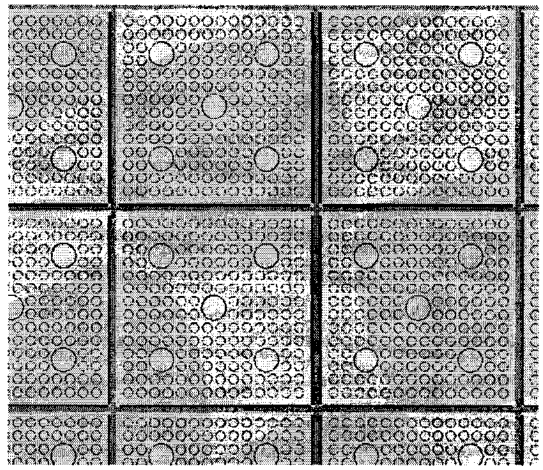


Figure 4-3
Fuel Assembly Position
Centered - Model

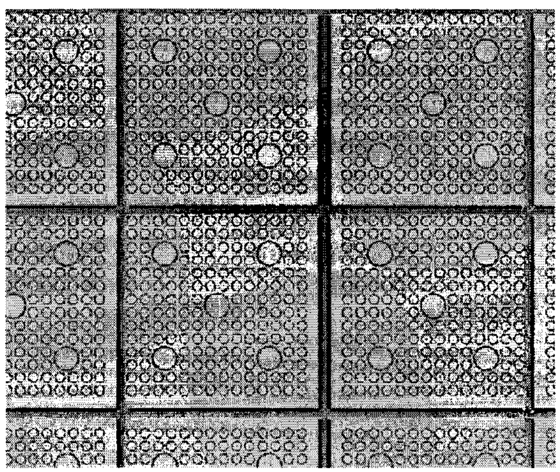


Figure 4-4
Fuel Assembly Position
Off-set Outward Model

5.0 ASSUMPTIONS

5.1 Key Assumptions

A key assumption is any assumption or limitation that must be verified prior to using the results and/or conclusions of a calculation for a safety-related task. There are no key assumptions in the present calculation or the reference calculations.

5.2 Assumptions and Conservatisms

- Criticality analysis was performed for the NUHOMS®-32PT Type "A" DSC as it is the most reactive design.
- NO PRA assemblies are modeled in the DSC.
- The NUHOMS®-32PT Type "A" DSC without PRA bounds the Type "B" DSC.
- Fort Calhoun SFP peripheral cells adjacent to the Cask Pit Area are maintained empty during DSC loading.
- No burnable poisons accounted for in any fuel assembly in the KENO model.
- NO PRA assemblies are modeled in the DSC.
- The transition rails between the basket and the canister shell is modeled as 100% aluminum. Steel and open space in the transition rails reduces reactivity because these materials have much higher absorption cross-sections as compared to the aluminum.
- All stainless steel is modeled as SS304. The small differences in the composition of the various stainless steels have no effect on results of the calculation
- CASMO cases assumed control rod was inserted for part of the depletion to maximize axial effects.
- Water density was at the optimum moderator density of 1.00 gram/cc corresponding to 4°C.
- All fuel rods are filled with fresh water in the pellet/clad gap for both normal and accident conditions.
- All cases assume full DSC reflection in the radial direction and axially with a 20 cm reflector.
- Only 90% credit is taken for the DSC B-10 in the poison plates.

6.0 RESULTS

6.1 CASMO Results

The CASMO results consist of the isotopic inventory used in KENO and these are documented in Reference 8.6 and in Attachment 1.

6.2 Normal Conditions KENO Results - Fresh Water

A series of CASMO / KENO runs has produced a curve to coincide in form with the technical specification of the Fort Calhoun Spent Fuel pool. However, the acceptance criteria are different; namely, the spent fuel pool acceptance criterion is for $k_{eff} < 0.95$ whereas the curves in this report have an acceptance criterion of $k_{eff} < 1$. It is understood that the Type "A" basket is limited to 3.9 $^{w}/_{o}$. Table 6-1 and Figure 6-1 summarize the results.

Table 6-1
Maximum Enrichment and Burn-up Results for Type "A" and "B" Transfer Cask

Enrichment $^{w}/_{o}$ U-235	Burn-Up (MWD/MTU)	KENO $k_{eff}+2\sigma$	$k_{eff} + 2\sigma + \Delta k_{eff}$
1.65	0	0.95759	0.97308
2.50	12,180	0.97825	0.99374
3.00	18,340	0.97968	0.99517
3.50	**24,110**	**0.98164**	**0.99713**
3.90	28,670	0.97949	0.99498
4.55 †	38,220	0.95967	0.98816

† The 4.55 $^{w}/_{o}$ case corresponds to a maximum burn-up of 36, 400 MWD/MTU and did not include burn-up uncertainty. A 5% burnup uncertainty was applied to the burnup. The maximum k_{eff} includes an axial bias uncertainty of 0.013 Δk.

6.3 Normal Conditions KENO Results – Borated Water

The most reactive configuration from the fresh case was analyzed with 500 ppm of soluble boron. The results below demonstrate the cask assembly k_{eff} is below 0.95.

Enrichment $^{w}/_{o}$ U-235	Burn-Up (MWD/MTU)	KENO $k_{eff}+2\sigma$	$k_{eff} + 2\sigma + \Delta k_{eff}$
3.5	24,110	0.89224	0.90773

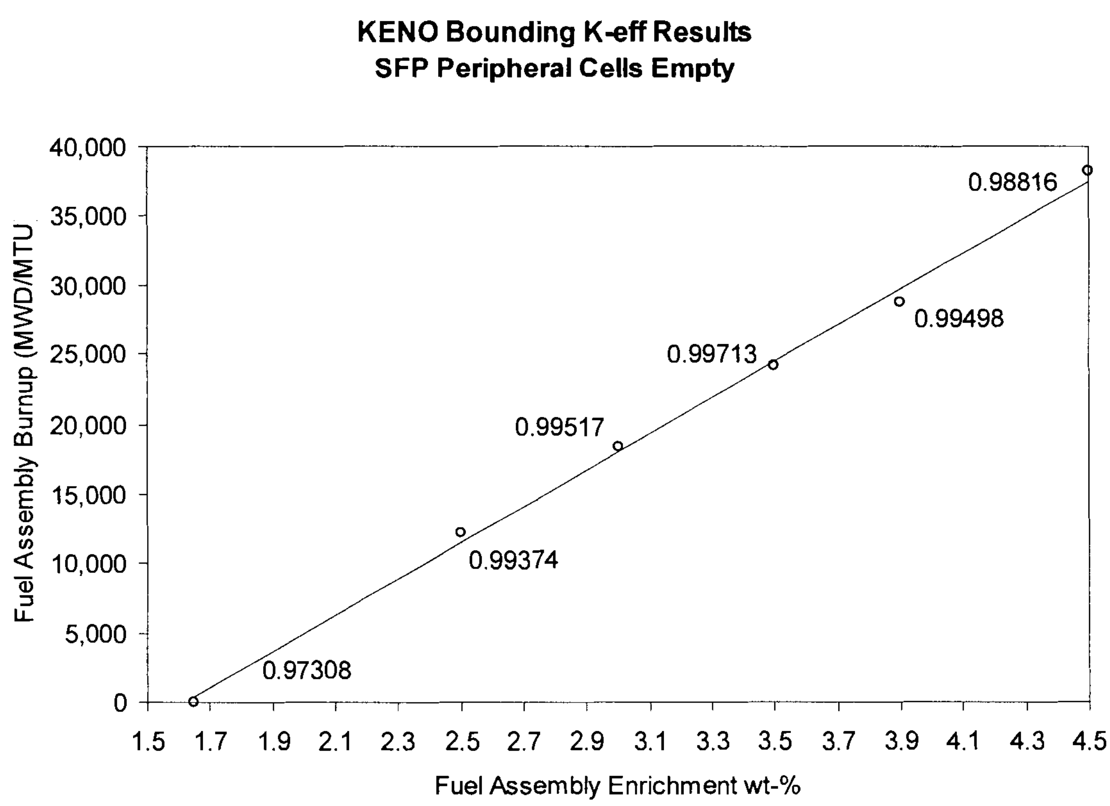
6.4 SFP Region 2 and DSC Type "A" Reactivity Comparison

The reactivity comparison between the DSC and SFP Region 2 was performed by selecting the DSC 3.50 $^{w}/_{0}$ result from Reference 8.6 and the SFP Region 2 3.5 $^{w}/_{0}$ case from Reference 8.16. The point is on Fort Calhoun Technical Specifications Figure 2-1[8.1]. These results are summarized in Table 6-2.

Table 6-2
SFP Region 2 and DSC Reactivity Comparison

Case	Enrichment $^{w}/_{0}$ U-235	Burn-Up (MWD/MTU)	KENO k_{eff}
SFP Region 2	3.5	24,240	0.9148
DSC- Type A	3.5	24,110	0.9809

Figure 6-1



6.5 DSC Position in the SFP Cavity

The DSC and transfer cask assembly is placed diagonally to the SFP fuel racks at approximately 45° due to physical limitations; that is, the cask can not fit in any other configuration. In the configuration shown in Figure 6-2, the closest the DSC transfer cask can approach the SFP rack is proximately 3 centimeters. That proximity is reached when the lifting hooks make physical contact with the SFP rack.

Figure 6-2
DSC and Transfer Cask Position in the SFP Cavity[8.10]

Figure Withheld Under 10 CFR 2.390

The coupling between the SFP and the DSC was evaluated by assuming the SFP peripheral cells are maintained empty and reflective boundary condition applied to the DSC in the radial direction. The full reflection assumption is in effect a second DSC where the SFP Region 2 is located. This is a conservative approach since the DSC is more reactive than the SFP Region No. 2.

6.6 Accident Condition – Misloaded Fuel

The misloaded fuel assembly accident analysis was performed by assuming the transfer cask is loaded with various enrichments as shown in Table 6-3 and the empty position is loaded with a fresh fuel assembly of 4.5 $^{w}/_{o}$ 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. All analysis assumptions from the normal case were also applied to this analysis; namely, all fuel rod gaps are flooded with pure water and the cask is fully reflected. The most reactive fuel in this configuration is the 4.55 $^{w}/_{o}$ enrichment. These results show the minimum required soluble boron concentration to maintain $k_{eff} < 0.95$ is 800 ppm with all uncertainties.

Table 6-3
Misloaded Fuel Assembly KENO k_{eff}

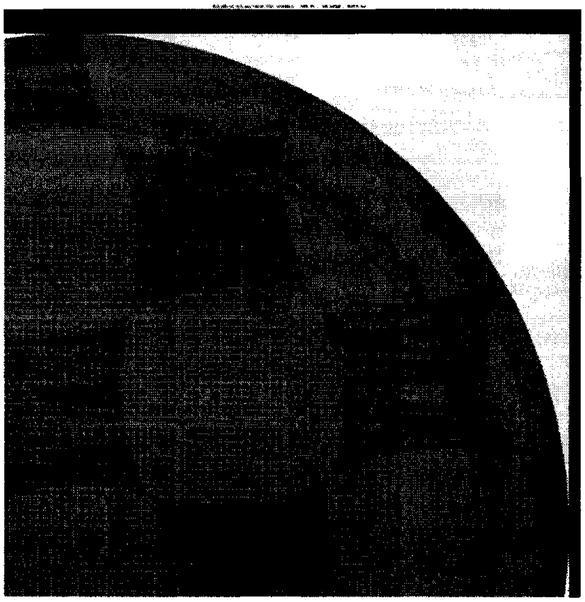
Reflected 0 cm Water DSC 3.5 $^{w}/_{o}$ with 500 ppm boron			
Bundle Position †	k_{eff}	σ	$k_{eff}+2\sigma+\Delta k_{eff}$
M1	0.92134	0.00045	0.93773
M2	0.94132	0.00053	0.95787
M3	**0.95504**	**0.00055**	0.97163
M4	0.93673	0.00049	0.95320
Reflected 0 cm of Water DSC 3.5 $^{w}/_{o}$ with 600 ppm boron			
M3	0.94204	0.00050	0.95853
Reflected 0 cm of Water DSC 3.5 $^{w}/_{o}$ with 700 ppm boron			
M3	0.92854	0.00053	0.94509
Reflected 0 cm of Water DSC 3.5 $^{w}/_{o}$ with 800 ppm boron			
M3	0.91710	0.00047	0.93353
Reflected 0 cm of Water DSC 3.9 $^{w}/_{o}$ with 700 ppm boron			
M3	0.93606	0.00046	0.95247
Reflected 0 cm of Water DSC 3.9 $^{w}/_{o}$ with 800 ppm boron			
M3	0.92384	0.00051	0.94035
Reflected 0 cm of Water DSC 4.55 $^{w}/_{o}$ with 800 ppm boron			
M3	0.91670	0.00052	0.94623

† See next page for the positions

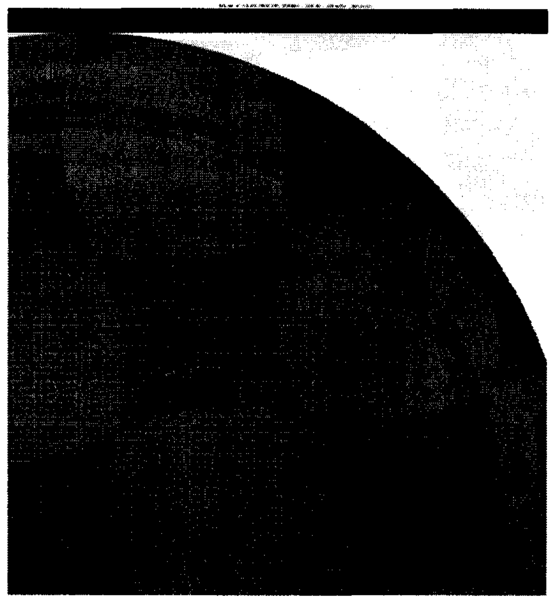
The analysis demonstrated the most reactive cells are toward the periphery of the cask array since these locations do not have boron plates. The misloaded positions are shown in Figure 6-3. The bounding value for the misloading fuel bundle accident at 800 ppm is the following:

$$k_{eff} = k_{eff\text{-}KENO} + 2\sigma_{KENO} + \Delta k_{eff}$$

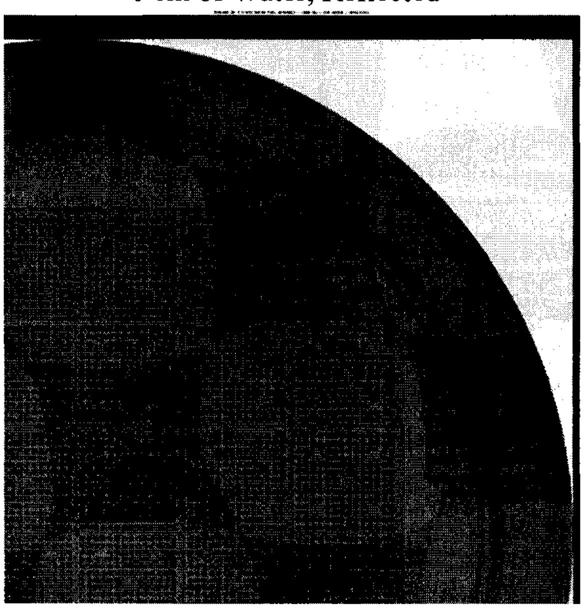
$$k_{eff} = 0.91670 + 0.00104 + 0.01549 + 0.013 = 0.94623$$



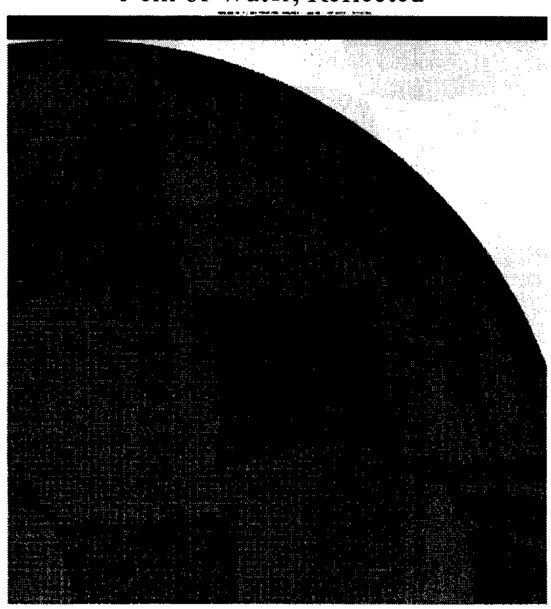
CASE M1
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected



CASE M2
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected



CASE M3
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected



CASE M4
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected

Figure 6-3

Misloaded Fresh Fuel Bundle Locations

6.7 Accident Condition - Fuel Drop

The fuel drop accident was evaluated with the same fuel assembly soluble boron assumptions as in the misloading accident. The dropped assembly is placed along the perimeter of the transfer cask aligned longitudinally and evaluated at different azimuthal locations to find the most reactive position. Table 6-4 summarizes the results.

Table 6-4
Fuel Assembly Drop KENO k_{eff}

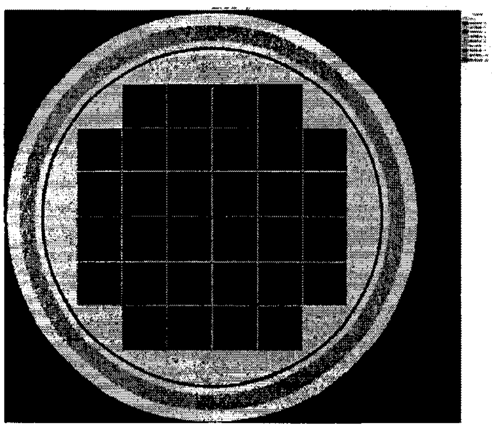
DSC 3.5 $^w/_o$ with Soluble Boron 500 ppm			
Bundle Position †	k_{eff}	σ	$k_{eff}+2\sigma+\Delta k$
0°	0.94282	0.00062	0.95955
15°	0.94246	0.00052	0.95899
30°	0.92996	0.00165	0.94875
45°	0.89168	0.00038	0.90793
DSC 3.5 $^w/_o$ with Soluble Boron 600 ppm			
0°	0.92607	0.00062	0.94280
DSC 3.5 $^w/_o$ with Soluble Boron 700 ppm			
0°	0.91023	0.00057	0.92686
DSC 3.5 $^w/_o$ with Soluble Boron 800 ppm			
0°	0.89391	0.00077	0.91094
DSC 3.9 $^w/_o$ with Soluble Boron 800 ppm			
0°	0.89554	0.00076	0.91255
DSC 4.55 $^w/_o$ with Soluble Boron 800 ppm			
0°	0.89518	0.00066	0.92499

† See next page for the positions

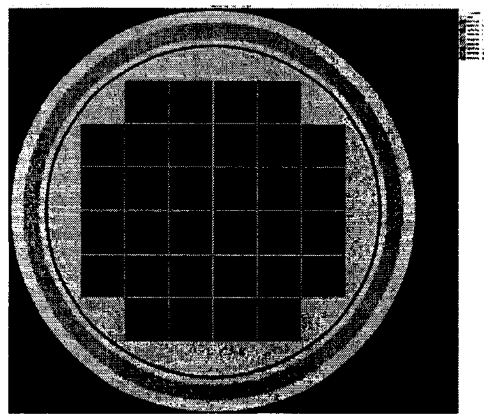
The 0° position is most reactive and the system k_{eff} is bounded by the misloading accident. The bounding k_{eff} value with 800 ppm of soluble boron for the dropped fuel bundle accident is bounded by the misloading accident and is the following:

$$k_{eff} = k_{eff\text{-}KENO} + 2\sigma_{KENO} + \Delta k_{eff}$$

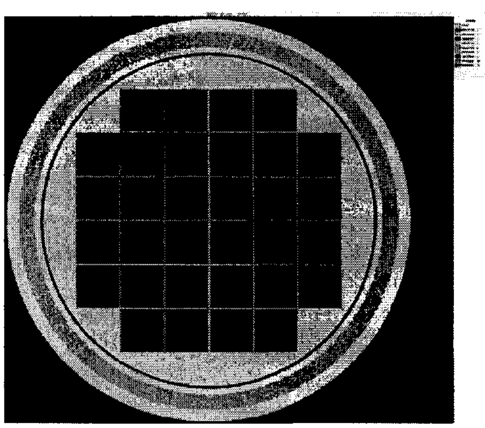
$$k_{eff} = 0.89518 + 0.00132 + 0.01549 + 0.013 = 0.92499$$



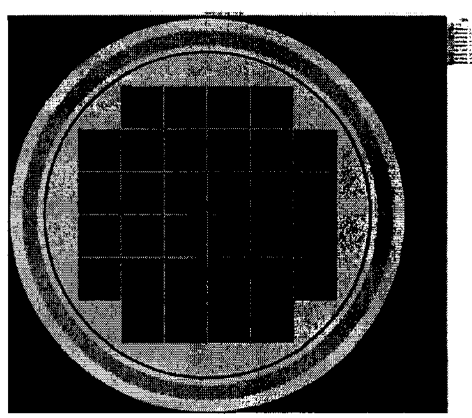
Dropped Assembly at 0°



Dropped Assembly at 15°



Dropped Assembly at 30°



Dropped Assembly at 45°

Figure 6-4

Dropped Fuel Bundle KENO Geometry

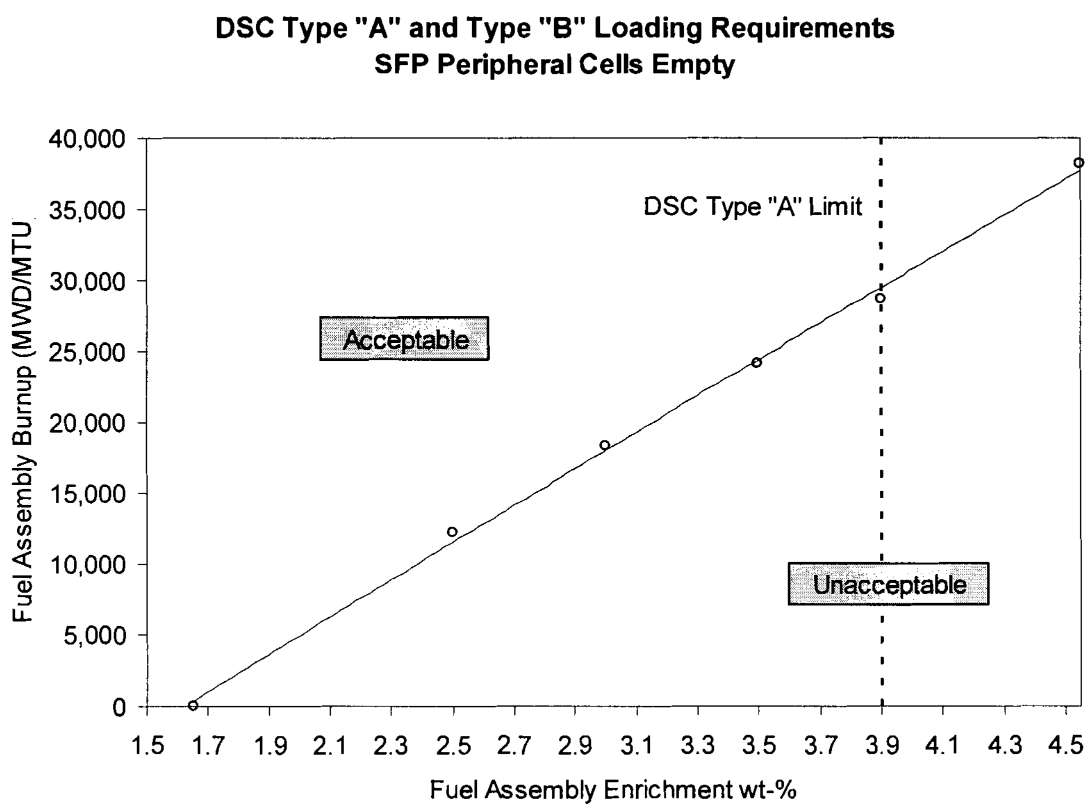
6.8 DSC Type "A" and Type "B"

The reference KENO calculations were all performed in the most reactive DSC design. It corresponds to the Type "A" basket with no PRA and only 16 poison plates. The basket is limited in enrichment to 3.9 $^{w}/_{o}$ by the licensing Certificate of Conformance (CoC) [8.2]. The KENO calculations, however, were extended to 4.55 $^{w}/_{o}$ and the results show the Type "A" DSC with 4.55 $^{w}/_{o}$ meet the acceptance criterion. Therefore, any fuel cask up to 4.55 $^{w}/_{o}$ with a minimum burn-up of 38,220 MWD/MTU can be stored in any of the NUHOMS®32PT DSC basket designs listed in the CoC.

6.9 DSC Type "A" and Type "B" Loading Curve

The KENO results for the fuel enrichment and burn-up combinations listed in Table 6-1 support a loading curve for all the NUHOMS®32PT DSC basket designs listed in the CoC. Figure 6-5 is a composite curve for the Type A and B DSC basket designs.

Figure 6-5



7.0 SUMMARY/CONCLUSIONS

The DSC criticality analysis performed by FANP demonstrates the calculated loading curve complies with §50.68(4) as follows:

- The most reactive DSC cask configuration (normal case) has a $k_{eff} < 1$ (i.e., 0.99713) with unborated water and a keff < 0.95 (i.e., 0.92575) when flooded with borated water at a concentration of 500 ppm. Both cases apply burnup credit.

- The bounding misloaded fuel assembly accident credits soluble boron and the results for 800 ppm is a k_{eff} value of 0.94035. Therefore, the minimum boron concentration required to maintain $k_{eff} < 0.95$ for accident conditions is 800 ppm with burnup credit.

- Any fuel cask up to 4.55 $^{w}/_{o}$ with a minimum burn-up of 38,220 MWD/MTU can be stored in any of the NUHOMS®32PT DSC basket designs listed in the OPPD Certificate of Conformance.

- The minimum required burnup as a function of initial enrichment can be expressed as a third order polynomial as shown below:

Burnup (MWD/MTU) = A + B1*E + B2*E² + B3*E³

$$\text{Burnup (MWD/MTU)} = A + B1 \cdot E + B2 \cdot E^2 + B3 \cdot E^3$$

Where

A	=	-42324
B1	=	36442
B2	=	-7929.3
B3	=	837.1
E	=	Initial Enrichment expressed as $^{w}/_{o}$ U-235

- The misloaded fuel bundle accident bounds the dropped assembly accident.

These results require spent fuel pool peripheral cells adjacent to the Cask Pit Area are maintained empty during DSC loading operation.

8.0 REFERENCES

8.1 AREVA/FANP 38-9004068-000, "OPPD Tech Spec for SFP," 1121-0100-23.

8.2 AREVA/FANP 38-9004083-000, TN Transmittal No. 1121-0011-01, Amendment 8A, Chapter M6, Criticality Evaluation".

8.3 NRC Regulatory Issue Summary 2005-05, Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations, March 23, 2005.

8.4 AREVA/TN Doc. # NUH32PT.0600, "NUHOMS®-32PT Transportable Dry Shielded Canister Criticality Analysis", Transnuclear West Calculational Package, 6/28/2001.

8.5 AREVA/TN Doc. # NUH32PT.0601, "NUHOMS®-32PT with 16 Poison Aluminum Plates", Transnuclear Calculational Package, approved 2/2/2004.

8.6 AREVA/FANP Calculation 32-9003495-000, NUHOMS-32PT Criticality Analysis for Fort Calhoun, October 2005.

8.7 AREVA/FANP Calculation 32-9001685-00, Fort Calhoun Transnuclear CASMO Analysis, October 18, 2005

8.8 AREVA/FANP Calculation 32-5038209-00, Fort Calhoun SFP Criticality Analysis, February 26, 2004.

8.9 AREVA/FANP Calculation 32-5038771-00, Fort Calhoun Spent Fuel Pool CASMO & Uncertainty Analysis, October 19, 2004.

8.10 AREVA/FANP 38-9004097-000, TN Transmittal No. 1121-0011-01, Drawing Sketch 1121-4035, "OS197L Light Yoke Clearences in OPPD Spent Fuel Pool".

8.11 AREVA/FANP Doc. # 32-1245321-00, "KENO V.a SCALE 4.2 Benchmark Calculations," L.A. Hassler, released 3/24/1997.

8.12 AREVA/FANP Doc. # EMF-96-029(P)(A), "Reactor Analysis System for PWRs," S.K. Merk, N.A. Anguiano, C.J. Lewis, R.W. Twitchell, A.H. O'Leary, R.G. Grummer, January, 1997.

8.13 AREVA/FANP Doc. # E-6088-N01-1, Revision 0, "Ft. Calhoun PRISM Benchmarking Cycles 17-20," K.C. Segard and N.A. Anguiano, signed off on 1/25/2002.

8.14 U.S. NRC Memorandum, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," from Laurence Kopp, August 19, 1998.

8.15 AREVA/FANP Doc. # 77-5069740, Revision 0, "Shearon Harris Criticality Evaluation" Appendix A - Bias and Bias Uncertainty Evaluation, August 2005.

8.16 AREVA/FANP 32-9004446-000, "Design Input 1121-0100-26 EA-FC-96-001 Rev 0"

Attachment A
CASMO Isotopic Inventory for Burnup Credit

The number densities for the various isotopes were determined from the running of CASMO in calculation 32-9001685-000 [8.7]. While a significant number of CASMO and KENO runs were performed, the results presented in this attachment are the five sets values that are germane to the technical specification DSC loading curve. The isotopics were calculated in [8.7] with special constraints regarding burnup and burnable poisons, in order to add conservatism to this calculation the burnable poison is not included as a part of the KENO input. The following tables present the isotopic number densities for the specific cases of interest.

Number Densities of Isotopes at 2.5 w/o Enrichment and 11.57 MWd/kg Burnup

Isotope	Number Density
kr-83	1.14978E-06
rh-103	8.77403E-06
xe-131	7.57771E-06
cs-133	1.73247E-05
cs-134	7.74486E-07
cs-135	5.44835E-06
nd-143	1.35117E-05
nd-145	5.60599E-05
pm-147	4.06968E-06
sm-147	6.00616E-07
sm-149	1.05617E-07
sm-150	3.50666E-06
sm-151	2.99131E-07
sm-152	1.74630E-06
eu-153	1.06085E-06
eu-154	1.72340E-07
eu-155	1.35399E-07
u-234	3.78897E-06
u-235	3.50755E-04
u-236	4.17397E-05
u-238	2.21288E-02
np-237	2.55799E-06
pu-238	3.63081E-07
pu-239	9.44123E-05
pu-240	2.16590E-05
pu-241	8.93673E-06
pu-242	1.09121E-06
am-241	1.28453E-07
am-242m	1.38667E-09
am-243	7.29087E-08
cm-242	2.36401E-08
cm-244	6.35366E-09
ag-109	8.57360E-07

Number Densities of Isotopes at 3.0 w/o Enrichment and 17.24 MWd/kg Burnup

Isotope	Number Density
kr-83	1.64262E-06
rh-103	1.25700E-05
xe-131	1.07698E-05
cs-133	2.52894E-05
cs-134	1.50213E-06
cs-135	9.07020E-06
nd-143	1.93952E-05
nd-145	8.26078E-05
pm-147	5.19305E-06
sm-147	1.16650E-06
sm-149	1.21835E-07
sm-150	5.35565E-06
sm-151	3.78887E-07
sm-152	2.46499E-06
eu-153	1.77454E-06
eu-154	3.50312E-07
eu-155	2.21189E-07
u-234	4.21565E-06
u-235	3.63224E-04
u-236	6.07220E-05
u-238	2.19283E-02
np-237	4.31517E-06
pu-238	8.13810E-07
pu-239	1.10581E-04
pu-240	3.07325E-05
pu-241	1.47130E-05
pu-242	2.40265E-06
am-241	3.04944E-07
am-242m	3.73336E-09
am-243	2.31333E-07
cm-242	6.93993E-08
cm-244	2.85659E-08
ag-109	1.34392E-06

Number Densities of Isotopes at 3.5 w/o Enrichment and 22.90 MWd/kg Burnup

Isotope	Number Density
kr-83	2.09699E-06
rh-103	1.60198E-05
xe-131	1.35904E-05
cs-133	3.27658E-05
cs-134	2.39189E-06
cs-135	1.33913E-05
nd-143	2.49432E-05
nd-145	1.08197E-04
pm-147	5.95153E-06
sm-147	1.79783E-06
sm-149	1.35956E-07
sm-150	7.19843E-06
sm-151	4.64319E-07
sm-152	3.07305E-06
eu-153	2.53253E-06
eu-154	5.77729E-07
eu-155	3.29501E-07
u-234	4.54626E-06
u-235	3.78380E-04
u-236	7.94066E-05
u-238	2.16453E-02
np-237	6.46629E-06
pu-238	1.52726E-06
pu-239	1.25355E-04
pu-240	3.89142E-05
pu-241	2.04652E-05
pu-242	4.03443E-06
am-241	5.51327E-07
am-242m	7.32830E-09
am-243	5.06057E-07
cm-242	1.40253E-07
cm-244	8.13660E-08
ag-109	1.83072E-06

Number Densities of Isotopes at 3.9 w/o Enrichment and 27.23 MWd/kg Burnup

Isotope	Number Density
kr-83	2.41887E-06
rh-103	1.82878E-05
xe-131	1.54680E-05
cs-133	3.82667E-05
cs-134	3.18175E-06
cs-135	1.66449E-05
nd-143	2.88363E-05
nd-145	1.27757E-04
pm-147	6.31422E-06
sm-147	2.34653E-06
sm-149	1.42764E-07
sm-150	8.55528E-06
sm-151	5.14663E-07
sm-152	3.45391E-06
eu-153	3.11976E-06
eu-154	7.82885E-07
eu-155	4.25635E-07
u-234	4.85731E-06
u-235	3.93985E-04
u-236	9.29018E-05
u-238	2.15005E-02
np-237	8.15039E-06
pu-238	2.31059E-06
pu-239	1.27526E-04
pu-240	4.21303E-05
pu-241	2.39232E-05
pu-242	5.73574E-06
am-241	7.80913E-07
am-242m	1.10183E-08
am-243	9.00791E-07
cm-242	2.23098E-07
cm-244	1.84831E-07
ag-109	2.17563E-06

Number Densities of Isotopes at 4.55 w/o Enrichment and 34.59 MWd/kg Burnup

Isotope	Number Density
kr-83	3.01079E-06
rh-103	2.24158E-05
xe-131	1.88101E-05
cs-133	4.77601E-05
cs-134	4.45106E-06
cs-135	2.31554E-05
nd-143	3.60785E-05
nd-145	1.60996E-04
pm-147	7.04561E-06
sm-147	3.28527E-06
sm-149	1.57943E-07
sm-150	1.08629E-05
sm-151	6.19866E-07
sm-152	4.14096E-06
eu-153	4.08317E-06
eu-154	1.11946E-06
eu-155	5.80637E-07
u-234	5.20317E-06
u-235	4.04771E-04
u-236	1.19459E-04
u-238	2.12510E-02
np-237	1.14123E-05
pu-238	3.59804E-06
pu-239	1.43440E-04
pu-240	5.17003E-05
pu-241	2.99696E-05
pu-242	7.52720E-06
am-241	1.12934E-06
am-242m	1.66321E-08
am-243	1.32010E-06
cm-242	3.22710E-07
cm-244	2.95961E-07
ag-109	2.69958E-06