# Kewauuee
# New and Spent Fuel Storage Rack
# Licensiug Report

**Non-Proprietary**
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# Executive Summary

Wisconsin Public Service Corporation (WPSC) contracted FCF to perform a full licensing analysis of the New and Spent Fuel Storage Racks for the Kewaunee Nuclear Power plant. The evaluations contained herein are intended to assess the impact of a new Siemens 14 x 14 fuel assembly design that has a heavier loading than previous designs. All analyses were based on a maximum as-built loading of 56.067 g $^{235}$U/axial-cm and a maximum as-built enrichment of 5.00 wt% $^{235}$U.

The criticality evaluation showed that no burnup credit is required for the Spent Fuel Storage Racks (SFSR). No cell locations are required to be locked out in the New Fuel Storage Racks (NFSR). A maximum *k*-effective of [b., c., e.] was shown to exist for the limiting NFSR accident condition, leaving [b., c., e.] margin to the 0.98 criticality limit for misted conditions. An even greater margin of [b., c., e.] to 0.95 k-effective limit was determined for a flooded NFSR with 100% dense moderator. For the SFSR, there is [b., c., e.] margin to the 0.95 criticality limit.

A thermal evaluation of the spent fuel pool and racks showed that even with a single pump failure the pool bulk temperature will remain [b., c., e.] below the design temperature of 150°F. Complete failure of the system will yield a pool heat-up rate of [b., c., e.]. The air temperature in the spent fuel building should remain below [b., c., e.]. Free convection in the pool will allow [b., c., e.] of margin to bulk boiling and [b., c., e.] margin to local boiling in the racks, even with a full core off-load.

The radiological analysis showed that the increased enrichment and loading of the Siemens 14 x 14 assembly will not significantly increase energy emission nor any dose rates where shielding is required. Fuel handling accident doses were shown to be significantly below the NUREG-0800 limits of 75 rem thyroid and 6 rem whole-body. The performance of the spent fuel pool ventilation system and spent fuel pool cleanup system were also shown to be acceptable.

## 1.0 Introduction

The New Fuel Storage Rack (NFSR) and Spent Fuel Storage Rack (SFSR) require evaluation to address two main objectives. The first is to address a new Siemens heavy fuel assembly design with a higher total fuel loading (from [b., c., e.] for a Westinghouse 14 x 14 assembly and [b., c., e.] for an ANF (Siemens) standard 14x14 assembly to approximately [b., c., e.]for the new Siemens 14x14 heavy assembly). The second is an increase in fuel enrichment for the Siemens design to an as-built maximum enrichment of [b., c., e.]. Three analyses are included: criticality, thermal, and radiological analyses.

The criticality analysis consists of two major parts. The first is an evaluation of the NFSR and the second is an evaluation of the SFSR. The NFSR analysis addresses both flooded and interspersed moderator conditions. No assembly locations are required to be locked out for 5.00 wt% $^{235}U$ fuel in the NFSR. The NFSR analysis determines the most reactive moderator temperature for flooded conditions. Interspersed moderator conditions are also evaluated. The misplaced and T-Bone accident are not evaluated because the previous criticality analysis [1] demonstrated that the .misted. condition is most reactive by a significant margin.

The SFSR analysis assumes no soluble boron in the moderator and is the limiting accident condition. Therefore, no misplaced or T-bone drop accident calculations are required. No burnup credit or soluble boron credit is required as an administrative control to achieve the required enrichment limit. The criticality safety criterion is satisfied by use of a fixed $B_4C$ absorber. A moderator temperature study is performed to determine the most reactive conditions. Off-center fuel placement and soluble boron worth are evaluated. Fuel assembly and rack tolerance effects are also considered.

In addition to the criticality analysis, the Spent Fuel Pool Cooling System (SFPCS) was evaluated to ensure that the thermal load of higher enriched fuel could be accommodated. The SFPCS analysis evaluated the system performance, natural circulation cooling of spent fuel assemblies, and the heating and ventilation system requirements. This analysis is described in Section 5.0.

Finally, the radiological consequences of higher enriched fuel were evaluated for the spent fuel storage pool. Considerations included adequacy of shielding, evaluation of fuel handling accident doses, and the adequacy of the spent fuel pool cleanup system. These results are discussed in Section 6.0.

## 2.0 Criticality Analytical Metbods

This section discusses the criticality analysis. It briefly describes computer programs, licensing requirements, and computer models used for the analysis.

### 2.1 Computer Programs and Standards

The absolute reactivities for the various storage rack configurations were determined with the KENO V.a Monte Carlo program [2]. The basic cross section sets used for the analyses are the [b., c., e.] group sets. Self-shielding of uranium resonances are calculated with the NITAWL-II [3] and BONAMI-S [4] programs. Treatment of spatial effects using the XSDRNPM-S [5] program was not required except for testing purposes since all calculations modeled fuel on a heterogeneous pin-by-pin basis. The CSAS [6] modules were used to link and execute the different codes that make up the SCALE 4.2 code system. The choice of these codes is based on extensive in-house benchmarks [7] against critical configurations directly applicable to this analysis. Brief descriptions of these computer programs follow.

1. The KENO V.a program is a module of the SCALE system and is a multigroup Monte Carlo program that allows a simple description of complex geometries. This code calculates the $k_{eff}$ of the modeled system and its associated statistical uncertainty.

2. The NITAWL-II program is a module of the SCALE system and processes cross sections for use with KENO V.a. It provides resonance self-shielding corrections to the cross sections that were used in this study. The neutron resonance self-shielding calculation employs the Nordheim Integral treatment.

3. The BONAMI-S program is a module of the SCALE system which is used to perform Bondarenko calculations for resonance self-shielding. Cross sections and Bondarenko factor input data are input from an AMPX master library.

4. XSDRNPM-S is a module of the SCALE system and is a discrete-ordinates code which solves the one-dimensional Boltzmann equation in slab, cylindrical, or spherical coordinates. In SCALE, XSDRNPM-S is used for several purposes: eigenvalue (K-effective) determination, cross-section collapsing, shielding analysis, and for producing bias factors for use in Monte Carlo shielding calculations.

5. CSAS modules are used to define control sequences for criticality problems. The control sequences activate cross-section processing codes like BONAMI-S, NITAWL-II, and XSDRNPM-S. The CSAS module can then execute KENO V.a. Search capabilities are also provided.

## 2.2 Analytical Requirements and Assumptions

ANSI/ANS-57.2 addresses storage of nuclear fuel at nuclear power plants and is considered one of the important applicable standards by the USNRC. Section 6.4.2.1.3 [8] requires that consideration be given to credible abnormal occurrences. In response to these requirements, the following occurrences are evaluated:

1. The dropping of the Siemens heavy fuel assembly is bounded by the deborated pool accident. Additionally, evaluation of an assembly lying horizontally across the top of the storage cell was evaluated in Reference 1 for both Westinghouse and ANF (Siemens) 14 x14 assembly designs and was significantly bounded by more severe accident scenarios (i.e., misted interspersed moderation in the NFSR or deboration of the spent fuel pool). The tilting of an assembly in the SFSR is bounded by the off-center assembly analysis in section 3.3.2. The NFSR was analyzed by artificially reducing the rack spacing to account for movement of assemblies within the assembly guides as well as off-center placement of the guides. Therefore, all calculations for the NFSR inherently contain any penalty for tilted or off-center assembly placement. (Section 6.4.2.1.3(a,f))

2. For the case where the assembly falls into the opening in the storage rack and breaks through the hard stops that support the assembly, the additional vertical distance in the rack that the assembly could travel is 3.25 inches in the SFSR. This accident is not analyzed because it is conservatively bounded by the deboration accident. (Section 6.4.2.1.3(a,d))

3. Tipping of the storage rack was not analyzed because the Kewaunee SFSRs are restrained from significant horizontal movement by seismic braces. (Section 6.4.2.1.3(b))

4. Misplacement of a fresh fuel assembly in the NFSR is prohibited by the floor grate. Reference 1 considered the misplacement of a fresh fuel assembly adjacent to the SFSR and determined the penalty to be [b., c., e.]. This penalty would not be significantly different for the Siemens heavy fuel assembly and the misplaced assembly penalty is significantly bounded by the deboration event [b., c., e.] . (Section 6.4.2.1.3(c,g))

5. Misloaded assembly accidents are not possible for either the SFSR or the NFSR since only fresh fuel is considered. (Section 6.4.2.1.3(c))

6. A stuck fuel assembly with a crane providing an uplifting force is construed to mean that the assembly hangs up due to contact between the assembly and the rack structural material. The Kewaunee storage rack design prevents this accident. The assemblies are inserted into a can that is smooth-walled and has a flared top guide on the top of the

rack to channel assemblies into the opening. The NFSR has flared top and bottom guides. (Section 6.4.2.1.3(e))

7. Horizontal movement of the assembly within the SFSR is evaluated by the off-center analysis in section 3.3.2. For the NFSR the assemblies were moved closer together by the maximum amount of off-center movement thus providing a bounding analysis. (Section 6.4.2.1.3(f))

8. The only significant objects that could fall into or on the spent fuel rack other than a fuel assembly are the spent fuel handling cranes and the motor driven platform. Both the cranes and platform under all credible loadings including seismic, will neither fall nor become detached from their rail structure. (Section 6.4.2.1.3(h))

9. Any threat to the storage racks from missiles generated by failure of rotating machinery or from natural phenomena is covered by the facility SAR. There are no missiles dependent on fuel enrichment or loading. (Section 6.4.2.1.3(i))

In addition to abnormal occurrences, ANSI/ANS-57.2 Section 6.4.2.2.5 [8] requires determination of optimal fuel assembly parameters. These are considered by the following:

1. The fuel model employed in the analysis represents the most reactive Zircaloy-4 clad assembly that is allowed to be stored in the rack. This model was based on assembly dimensions, materials, and fissile loadings. (Section 6.4.2.2.5(a,b.c,d,e,f,h))

2. No credit is taken for neutron absorption in intermediate spacer grids, control rods, or poison clusters. (Section 6.4.2.2.5(g))

3. No fission products are applicable since the fuel evaluated is fresh. (Section 6.4.2.2.5(i))

ANSI/ANS-57.2 Section 6.4.2.2.6 [8] requires the fuel assembly arrangement in the rack design shall be the arrangement that results in the highest value of K-effective.

1. Consideration was given to spacing between assemblies. (Section 6.4.2.2.6(a))

2. Moderation between assemblies was considered. (Section 6.4.2.2.6(b))

3. Fixed neutron absorbers between the fuel cells were evaluated. (Section 6.4.2.2.6(c))

ANSI/ANS-57.2 Section 6.4.2.2.7 [8] requires the determination of the fuel assembly arrangement with the highest value of k-effective. The following was considered:

1. Eccentricity or off-center placement of fuel assemblies was evaluated. (Section 6.4.2.2.7(a))

2. Dimensional tolerances were evaluated. (Section 6.4.2.2.7(b))

3. Variation in construction materials was considered. (Section 6.4.2.2.7(c))

4. Parametric studies were performed to determine the optimal moderation of the assemblies in both rack designs are a part of the analysis. (Section 6.4.2.2.7(d))

5. Presence of structural material and fixed neutron absorbers in cell walls was addressed with the most limiting configuration determined. (Section 6.4.2.2.7(e))

6. The variation of neutron absorber in the cell wall was addressed. (Section 6.4.2.2.7(f))

ANSI/ANS-57.2 Section 6.4.2.2.8 indicates that credit may be taken for the inherent neutron absorbing effect of materials of construction if certain requirements are met. This analysis is conservative because:

1. [b., c., e.]

2. The effectiveness of additional poisons is not required (such as control components) since none were credited in this analysis. (Section 6.4.2.2.8(b))

3. [b., c., e.]

## 2.3 Computational Models and Methods

This section describes the models used for the storage racks, the base fuel assembly, and the tolerance effects and accident analyses.

### 2.3.1 Fuel Assembly Description.

This analysis evaluated a Siemens heavy 14x14 fuel assembly design. Table 2.3.1-1 defines the specifications and dimensions for the fuel assembly [9-11]. Also shown are the actual dimensions used in the analysis. Section 7.5 in the Appendix provides further amplification of values used. Bounding guide tube dimensions were used in KENO V.a calculations, since slight differences exist between the upper and lower guide tube dimensions. In KENO V.a calculations, the intermediate spacer grids were not modeled.

Table 2.3.1-1 Siemens Heavy Fuel Assembly Parameters

[b., c., e.]

[b., c., e.]

## 2.3.2 NFSR and SFSR Designs

The NFSR is shown in Figure 2.3.2-1 while the SFSR is shown in Figure 2.3.2-2. Both figures are shown with nominal dimensions. The limiting dimensions used for the NFSR are defined in Appendix Section 7.5.2. The limiting dimensions for the SFSR are defined in Appendix Section 7.5.3. The dimensions chosen for the rack analyses are based on maximizing reactivity.

Fignre 2.3.2-1 NFSR Confignration

[b., c., e.]

Figure 2.3.2-2 SFSR Configuration

[b., c., e.]

### 2.3.3 New Fuel Storage Rack Model

The analytical model of the NFSR is described in this section. The Burnable Poison Rod Assembly (BPRA) racks are not modeled in the NFSR because they can only contain poison rods and no credit is taken for rods or poison cluster assemblies. The model for the NFSR flooded with fully dense moderator has the assembly guides modeled as stainless-steel cans. The modeling of the assembly guides as either solid steel cans or as gridwork makes no difference in the final answer since the fuel near the center of the assembly which has no can establishes the K-effective of the system. This is not the case for low-density interspersed moderator conditions. Therefore, the final models for the NFSR assembly guides use solid steel cans for fully flooded cases and steel gridwork for the low-density moderator cases. The assembly guide gridwork has been modeled with minimum dimensions for low-density moderator cases to reduce the volume of steel (which acts as an absorber).

The analytical model for the NFSR is shown in Figure 2.3.3-1 and 2.3.3-2. The flared top guides above the rack structure are conservatively not modeled. The rack bracing and intermediate fuel assembly grids are also not modeled. The center-to-center spacing of the assemblies in the steel guides has been reduced to account for off-center spacing of both the assemblies and cans. Therefore, off-center movement has been accommodated by artificially making the rack smaller.

Figure 2.3.3-1 NFSR Axial Model

[b., c., e.]

**Fignre 2.3.3-2 NFSR Analytical Model - Top View**

[b., c., e.]

### 2.3.4 Spent Fuel Storage Rack Model

The SFSR was modeled as shown in Figures 2.3.4-1 and 2.3.4-2. Note that the SFSR has unequal spaces between the outer two peripheral rows of assemblies. This detail was modeled because it allows more water in the flux trap region for cell types A-E and G-I (see Figure 2.3.4-1). Cell Type F has the smallest gap, [b., c., e.] inches, with tolerances applied. The other gaps with tolerances are [b., c., e.] inches and [b., c., e.] inches, respectively. The racks in the SFSR consist of a 9x10 array of stainless-steel cans. The analytical model used a 10x10 array of assemblies for conservatism. The SFSR has borated plates next to the sides of each can. The borated plate has a matrix of phenolic resin imbedded with $B_4C$ with a minimum $B^{10}$ areal density of [b., c., e.]. The borated plate is sandwiched against the steel can by a thin steel plate with a nominal thickness of [b., c., e.] inches. The significant axial detail is shown in Figure 2.3.4-2. The lower support frame structure was not modeled since variations in water and steel in this region were shown to have no statistically meaningful effect on the calculated results. A cross-section of a single assembly cell is shown in Figure 2.3.4-3. The limiting dimensions that maximize reactivity are shown in parenthesis. All fuel rods in the assembly were modeled on a pin-by-pin basis with the rod plenum region and end caps modeled. The upper and lower assembly end fittings were modeled as homogeneous steel-water regions.

Figure 2.3.4-1 SFSR Model - Top View

[b., c., e.]

Figure 2.3.4-2 SFSR Model - Side View

[b., c., e.]

Figure 2.3.4-3 SFSR Fuel Assembly Unit With Tolerauces

[b., c., e.]

### 2.3.5 Material Specification

The heterogeneous number densities for $UO_2$ are shown in Tables 2.3.5-1 for the nominal and maximum fuel loadings. The densities for steel, water (fully dense and misted conditions), and smeared densities for the upper and lower end fittings are shown in Tables 2.3.5-2 through 2.3.5-6. Tables 2.3.5-7 and 2.3.5-8 list the number densities for Cured Portland Type I cement and borated phenolic resin, respectively. Tables 2.3.5-9 through 2.3.5-12 list the densities for zircalloy cladding, polyethylene, SS304, and oak flooring. All number densities are given in units of at/b-cm.

**Table 2.3.5-1 Heterogeueous Fuel Densities**
(at/b-cm)

[b., c., e.]

**Table 2.3.5-2 Heterogeneous Number Densities for CF-3 Stainless Steel**

[b., c., e.]

Table 2.3.5-3 Heterogeneous Water Densities
(at/b-cm)

[b., c., e.]

Table 2.3.5-4 Homogenized Upper End Fitting Densities w/ Water
(at/b-cm)

[b., c., e.]

Table 2.3.5-5 Homogenized Lower End Fitting Densities w/ Water
(at/b-cm)

[b., c., e.]

**Table 2.3.5-6 Iuterspersed Moderator Densities at 50°F**
(at/b-cm)

[b., c., e.]

Note that to homogenize the **upper end fitting** with the moderator, the hydrogen and oxygen densities are multiplied by 0.843856. For the **lower eud fitting** the densities are multiplied by 0.819533.

**Table 2.3.5-7 Cured Portland Concrete**

[b., c., e.]

*The values in ( ) were adjusted to reflect dried or cured concrete and were used in the analysis.

The borated phenolic resin has a volumetric loading of .$B_4C$. of approximately 50%. The .$B_4C$. constituents are the following:

[b., c., e.]

The phenolic resin is of the Novolac type and contains only methylene links and are phenol terminated. The cured phenolic resin has molecular formula $C_8H_8O$ with density of 0.97 g/cc for pure phenolic resin.

**Table 2.3.5-8 Borated Phenolic Resin With** [b., c., e.] **Areal Density For A 0.19" Plate**

[b., c., e.]

Table 2.3.5-9 Zircalloy Cladding

[b., c., e.]

Table 2.3.5-10 Polyetbylene

[b., c., e.]

Table 2.3.5-11 SS304

[b., c., e.]

Table 2.3.5-12 Oak

[b., c., e.]

## 3.0 Summary of Criticality Results

This section contains a description of analysis performed on the NFSR and SFSR to define limiting conditions. Additionally this section defines the limiting hypothetical accident cases for both rack types. A KENO V.a bias is applied. The development of the bias is discussed in Appendix 7.0. A summary of bias, penalties and uncertainties is contained in Section 3.4.

### 3.1 Tolerance Effects

The tolerances that affect reactivity for this analysis are the assembly tolerances, and the tolerances on the NFSR and SFSR. The tolerances on the NFSR and SFSR design were addressed in Reference 1 and have not changed since the storage racks have not been modified. The only difference is the use of a 14x14 Siemens assembly with a heavier KgU loading.

#### 3.1.1 Siemens Heavy Fuel Assembly Tolerances

The tolerances for the Siemens heavy fuel assembly are defined in Appendix 7.5.1. The limiting assembly dimensions that maximize reactivity are defined in Table 2.3.1-1. The most notable of these dimensions are the maximum pellet diameter, minimum fuel rod O.D., minimum cladding thickness, maximum active fuel length, and minimum cladding volumes for the guide tubes and instrument tube. No spacer grids are modeled. The number densities for fuel were based on the maximum KgU per assembly. The maximum as-built enrichment of 5.00 wt% $^{235}U$ was used.

#### 3.1.2 NFSR Tolerances

The NFSR modeled the part length assembly guides using minimum dimensions on the stainless-steel gridwork that defines the guides. The analytical model for the NFSR is shown in Figure 2.3.3-1 and 2.3.3-2. The flared top guides above the rack structure are conservatively not modeled. The rack bracing and intermediate fuel assembly grids are also not modeled. The center-to-center spacing of the assemblies in the steel guides has been reduced to account for off-center spacing of both the assemblies and cans. Therefore, off-center movement has been accommodated by artificially making the rack smaller.

#### 3.1.3 SFSR Tolerances

The SFSR model is discussed in Section 2.3.4. The SFSR was modeled with minimum flux traps, minimum borated plate thickness, and minimum $B^{10}$ areal density. The dimensions that maximize reactivity are shown in parentheses in Figure 2.3.4-3 and were established in the Reference 1 analysis.

To demonstrate that the tolerances shown in Figure 2.3.4-3 are conservative two infinite lattice KENO V.a cases were evaluated. The first case used can dimensions designed to minimize the flux trap between assemblies, maximize water in the assembly, maximize fuel assembly loading, and used a minimum $B^{10}$ loading in the plates. The second case used nominal can and assembly dimensions, nominal assembly loading, and nominal $B^{10}$ loading in the borated plates. Both cases were performed at [b., c., e.]. Comparison of cases 1 and 2 in Table 3.1.3-1 for the Siemens heavy fuel assembly demonstrates that the case with worst tolerances applied is [b., c., e.] more reactive than the case with nominal dimensions. Therefore, the toleranced model used in all calculations for the SFSR is very conservative.

**Table 3.1.3-1 SFSR Tolerance Study**

[b., c., e.]

## 3.2 NFSR Analysis

Section 3.2 describes the NFSR analysis for Kewaunee. This analysis evaluates both the flooded and interspersed moderator conditions. The flooded cases must satisfy the criticality criterion of K-maximum $<0.95$, while the low density mist cases must demonstrate K-maximum $< 0.98$.

### 3.2.1 Fuel Assembly Loading Study

The heterogeneous fuel number densities are calculated using the maximum pellet diameter and fuel rod length, since these features maximize K-effective. Since there are variations in pellet dish factors, the fuel assembly loading was used to determine number density input. The fuel assembly loading has a $\pm 1\%$ tolerance. Therefore, two KENO V.a cases were run, one for a nominal loading of [b., c., e.] KgU and one for a maximum loading of [b., c., e.] KgU, to determine the affect on K-effective. The results are shown in Table 3.2.1-1. The analytical model used for the NFSR is shown in Figures 2.3.3-1 and 2.3.3-2. These results indicate that the base K-effective is most reactive for the [b., c., e.] loading of [b., c., e.] KgU by approximately [b., c., e.]. Therefore, the maximum loading is maintained throughout the balance of the analysis.

**Table 3.2.1-1 K-effective Versus Fnel Assembly Loading**

[b., c., e.]

[b., c., e.]

### 3.2.2 NFSR Temperature Study

A study was performed to evaluate which temperature conditions are most reactive for the fully flooded NFSR. Temperatures selected correspond to 39, 50, 70, 120, and 210°F. The temperature study was performed only for a fully flooded NFSR because the interspersed moderator cases are at low water densities and temperature is not very significant relative to the misted water density variations examined. The results from Table 3.2.2-1 demonstrate that K-maximum was reached at 50°F. Although the water density is slightly greater at 39°F, temperature effects on scattering make 50 °F slightly more reactive by [b., c., e.]. Therefore, subsequent NFSR problems are run at [b., c., e.].

**Table 3.2.2-1 NFSR Temperature Dependence**
(5.00 wt% $^{235}U$; w/ Bias 1D [b., c., e.]/Assm; SS304 Guides)

[b., c., e.]

### 3.2.3 Interspersed Moderator Conditions

The most reactive NFSR occurs with low-density mist or interspersed moderator conditions. The most reactive conditions are typically for misted water densities between [b., c., e.]. Therefore a series of misted cases were run using the 1/2 NFSR rack model. The [b., c., e.] misted case results, shown in Table 3.2.3-1, demonstrate that K-maximum occurs at [b., c., e.] water density with a K-maximum value of [b., c., e.] and maintains [b., c., e.] margin to the 0.98 criticality criterion for low-density moderator cases [8].

[b., c., e.]

The NFSR flooded with 100% dense water has a K-maximum of [b., c., e.] (from Table 3.2.2-1) and satisfies the criticality criterion of 0.95. Accident cases for a **dry** NFSR with either a dropped or misplaced assembly adds little reactivity to the system and are significantly less than 0.95. Therefore, the NFSR does not require any assembly locations to be locked out for 5.00 wt% $^{235}$U fuel in the Siemens heavy loaded fuel assemblies.

**Table 3.2.3-1 NFSR Interspersed Moderator Cases w/SS304 Cage**

[b., c., e.]

### 3.2.4 NFSR Off-Center Assembly Placement

When fuel assemblies are placed in the NFSR it is possible for the assemblies to be located off-center within the stainless-steel guides. Additionally, the assembly guides may be located off-center by a non-cumulative radius of [b., c., e.] inches. This causes the distance between two fuel assemblies to be reduced while the distance between two other assemblies is increased. Since the nominal assembly spacing in the NFSR is not uniform due to the spacing of rows, it is possible that specific assemblies could be selected such that off-center movement could result in an increase in K-effective of the system. To avoid performing numerous cases to define the worst off-center placement, it was decided to artificially alter the rack dimensions such that all fuel assemblies are .squeezed. together by the amount the assemblies can move within the assembly guides, the amount the guides can move within the rack, and the amount of the tolerance on the assembly pitch [b., c., e.] . This modification results in all assemblies being moved [b., c., e.] inches closer together with the rack becoming approximately [b., c., e.] inches shorter in length and [b., c., e.] inches shorter in width. The geometrical approach to this problem is very conservative because it is not physically possible to move all assemblies uniformly closer to each other and the nominal off-center placement of fuel is random. All NFSR calculations in this document incorporated this conservative approximation of off-center fuel.

### 3.2.5 NFSR Hypothetical Accident Cases

Examination of the Reference 1 criticality analysis reveals that the NFSR had a penalty for the T-Bone drop accident of [b., c., e.] for flooded conditions and [b., c., e.] for misted conditions with fuel at 5.05 wt% $^{235}U$. In Reference 1, two accident conditions were modeled (flooded or misted condition with a dropped assembly) and K-maximum was less than 0.95 for flooded cases and 0.98 for misted cases. The correct interpretation of the double contingency principle is that only a single accident case must be modeled and K-maximum remain less than the applicable criterion. Relative to the flooded or misted conditions a T-Bone drop accident is a small reactivity effect and not limiting. The misplaced accident is precluded by the NFSR rack design. Therefore, specific T-Bone accident calculations are not required in the evaluation of the NFSR. The limiting NFSR accident condition occurs for [b., c., e.] misted conditions with a K-maximum of [b., c., e.]. This case provides [b., c., e.] margin to the 0.98 criticality limit for misted conditions. For the NFSR flooded with 100% dense moderator the K-maximum value is [b., c., e.] and provides [b., c., e.] margin to the 0.95 K-effective limit. **No locations are required to be locked out for the NFSR for the Siemens hcavy assembly.**

## 3.3 SFSR Analysis

Section 3.3 describes the Spent Fuel Storage Rack (SFSR) analysis for Kewaunee. This analysis evaluates temperature effects, tolerances, and hypothetical accident cases. All cases must satisfy the criticality criterion that K-maximum < 0.95. The SFSR analytical model is shown in Figures 2.3.4-1, 2.3.4-2, and 2.3.4-3.

### 3.3.1 SFSR Temperature Stndy

A study was performed to evaluate which temperature conditions are most reactive for the fully flooded SFSR. Temperatures selected correspond to 39, 50, 70, 120, and 210°F. The results of the temperature study are shown in Table 3.3.1-1. The results from Table 3.3.1-1 demonstrate that K-maximum was reached at [b., c., e.]. Although K-maximum is slightly greater at [b., c., e.], the difference in reactivity between [b., c., e.] is negligible. Subsequent problems were run at [b., c., e.] for the SFSR.

**Table 3.3.1-1 SFSR Temperature Dependence**

[b., c., e.]

[b., c., e.]

### 3.3.2 Off-Center Fuel Placement

When fuel assemblies are placed in the storage racks, it is possible for the assemblies to be located slightly to one side in the stainless steel can. This causes the distance between fuel assemblies to be reduced and may result in an increase in K-effective of the system. To determine a reactivity penalty for off-centered fuel placement in the SFSR, an infinite assembly lattice was modeled. For conservatism it was assumed that each assembly was shifted until it made contact against two sides of the steel can. Furthermore, assemblies in groups of four were moved such that the distance between all four assemblies in a group was minimized as shown in Figure 3.3.2-1. These cases were analyzed for 5.00 wt% $^{235}U$ fresh fuel. The results of cases 1 and 2 are tabulated in Table 3.3.2-1 and indicate that no reactivity penalty applies for off-center fuel placement of assemblies in cans in groups of four assemblies.

It should be pointed out that the water gap spacing between cans in the 9x10 array racks is non-uniform with the outer two peripheral rows of assemblies having the largest water spacings between cans. For this reason assemblies in the interior region of the rack were modeled since they have the smallest water gap between cans thus maximizing reactivity. A water gap space of [b., c., e.] was determined and used in cases 1 and 2 by distributing the cumulative rack tolerance of [b., c., e.] over nine assemblies in a 9x10 rack and by assuming the maximum outside can dimension of [b., c., e.] .

The pitch between any two assemblies has a tolerance of [b., c., e.] inches which could result in a minimum water gap of [b., c., e.] inches between cans. However, this can spacing would require the spacing between other cans to be greater by [b., c., e.] inches to avoid violation of the cumulative rack tolerance. Examination of material tolerances indicates a more limiting minimum water spacing of [b., c., e.] inches is possible between cans either due to local bowing of the can or off-center placement of the can center by a **non-cumulative** radial distance of [b., c., e.] inches. This problem was evaluated by modeling an infinite grouping of 4x4 fuel assembly arrays with all fuel assembly cans in a 4x4 array shifted toward one corner. A 4x4 array was chosen since it approximately models 1/4 of a 9x10 assembly rack. The water gap spacing in part of the interior region of a 4x4 array was maintained at the minimum of [b., c., e.] inches. Furthermore, all assemblies within the cans are shifted in such a manner as to minimize the distance between fuel assemblies along two edges of the 4x4 array (see Figure 3.3.2-2). Calculations in Table 3.3.2-1 were performed for the conditions of an off-centered can (case 3) as well as an off-centered can and assembly (case 4). **Case 4 defines a small off-center penalty of** [b., c., e.]. **The uncertainty associated with the calculation is much greater than the small reactivity increase. Therefore, no penalty is applied for this scenario.**

If all sixteen assemblies and cans in the 4x4 array are moved closer together to maintain a [b., c., e.] water gap between all cans in the array, then a reactivity penalty is possible. From case 6 an off-center penalty of [b., c., e.] is possible if both cans and assemblies are off-centered. If all 16 cans are off-centered with a minimum spacing of [b., c., e.] but the assemblies are centered (case 5) the reactivity penalty is maximized as [b., c., e.]. **However, the penalty defined by cases 5 and 6 canuot be incurred without violating the rack tolerances.** Therefore, the only penalty applicable is that defined by case 4 and this penalty ([b., c., e.]) is statistically insignificant.

Table 3.3.2-1 SFSR Off-Center Study

[b., c., e.]

Figure 3.3.2-1 SFSR Off-Center Assembly Representation

[b., c., e.]

**Figure 3.3.2-2 SFSR 4x4 Off-Center Assembly and Can Representation**

[b., c., e.]

### 3.3.3 SFSR Hypothetical Accident Cases

The analysis performed in the Reference 1 criticality analysis indicated that the T-Bone accident case (an assembly dropped on top of the rack) did not increase reactivity for an assembly at 5.05 wt% $^{235}U$. Therefore, it is not necessary to perform this analysis again for the Siemens heavy assembly since this accident is not limiting. The misloaded assembly accident cases in Reference 1 (an assembly loaded between the pool wall and the rack) resulted in a penalty of [b., c., e.]. This penalty was also computed for 5.05 wt% $^{235}U$ fuel versus 5.00 wt% $^{235}U$ in this analysis. The T-Bone drop and misplaced assembly penalties are **insignificaut** relative to the limiting accident case which is the deboration of the pool. All calculations performed thus far assume no soluble boron in the SFSR. To determine the reactivity penalty associated with the lack of soluble boron, the 1/4 rack model (see Figures 2.3.4-1 and 2.3.4-2) was evaluated assuming [b., c., e.] ppm boron. The results from Table 3.3.3-1 indicate the soluble boron is worth [b., c., e.] in reactivity ([b., c., e.]). Therefore, the deborated SFSR is clearly the limiting hypothetical accident case.

**Table 3.3.3-1 SFSR Deboration Study**
(5.00 wt% $^{235}U$)

[b., c., e.]

The limiting value of K-maximum for the SFSR is [b., c., e.] from the 1/4 rack model and provides [b., c., e.] margin to the 0.95 criticality limit. Even the infinite array model (see case 1 Table 3.3.2-1) provides [b., c., e.] margin. The infinite array model is overly conservative because it is based on an infinite array of "Type F" cells, which are the ones most closely spaced in the interior.

### 3.4 Summary of Reactivity Penalties and Uncertainties

A summary of the various penalties, uncertainties, and conditions for which they apply, is provided in this section and were used in determining the maximum K-effective. These values are tabulated in Table 3.4-1 in terms of delta K-effective. All uncertainties listed correspond to a 95/95 tolerance factor. The KENO V.a bias is explained in Appendix 7.0.

The combined uncertainty from different penalties and the KENO V.a statistical uncertainty is computed by the square root of the sum of the squares of the individual uncertainty components.

When this uncertainty component is computed it is multiplied by a factor k, such that the uncertainty reflects a one-sided 95/95 upper tolerance factor for a normal distribution. [b., c., e.]

Note that some penalties defined in the Reference 1 criticality analysis for Westinghouse and Siemens 14x14 assemblies are included in Table 3.4-1 for comparison purposes and demonstrate that the misted and flooded condition is limiting for the NFSR and the deboration accident is limiting for the normally flooded SFSR.

**Table 3.4-1 Summary of Reactivity Penalties and Uncertaimties**

[b., c., e.]

# 4.0 Criticality Conclusions

The results of the criticality analysis for the Siemens heavy fuel assembly indicate that no cell locations are required to be locked out for the NFSR and no burnup credit is required for the SFSR. The limiting NFSR accident condition occurs for [b., c., e.] misted conditions with a K-maximum of [b., c., e.]. This case provides [b., c., e.] margin to the 0.98 criticality limit for misted conditions. For the NFSR flooded with 100% dense moderator, the K-maximum value is [b., c., e.] and provides [b., c., e.] margin to the 0.95 K-effective limit. The limiting value of K-maximum for the SFSR is [b., c., e.] from the 1/4 rack model for the deborated pool condition and provides [b., c., e.] margin to the 0.95 criticality limit.

The maximum as-built enrichment is **5.00 wt% $^{235}$U.**

The maximum as-built g $^{235}$U /axial-cm is:

[b., c., e.]

## 5.0 Speut Fuel Pool Cooling System Analysis

This section presents the spent fuel pool cooling system analysis. All aspects of WPSC.s current analysis will be updated to include the effects of a full core off-load (121 FAs) of the new Siemens 14x14 heavy fuel assembly design and a full spent fuel pool (990 FAs). The higher enrichment fuel will cause higher decay heats. This analysis determines the thermal performance of the spent fuel pool with these new assemblies and higher decay heats. It is divided into the three parts, described below.

*Spent Fuel Pool Cooling System Capacity Evaluation*
This section evaluates whether or not the heat exchangers have sufficient capacity to remove heat from the spent fuel pool such that the water in the pool does not exceed the design temperature of 150°F. The analysis is performed under the assumption of a single pump failure in the cooling system, leaving only one pump in operation.

*Verification of Heating, Ventilation, and Air Conditioning System*
The impact of a full core off-load on the ambient air in the spent fuel building is assessed.

*Spent Fuel Assembly Local and Bulk Fluid Temperature Evaluation*
The purpose of these sections is to show that the bulk fluid water temperature in the pool as well as the local fuel rod peak temperature is sufficiently low that no boiling will occur in the pool.

### 5.1 Fuel and System Design Parameters

WPSC has verified by letter [9] that the spent fuel pool parameters have not changed since the reference [1] analysis. A summary of inputs requested for the thermal / hydraulic analysis was verified [17], and parameters updated where applicable [31]. These values, along with some of the pertinent dimensions of the new fuel assembly design are shown in the following tables.

**Table 5.1-1 Spent Fuel Pool Design Parameters**

[b., c., e.]

**Table 5.1-2 Spent Fuel Pool Heat Exchanger Parameters**

[b., c., e.]

Where: [b., c., e.]

Also: [b., c., e.]

Table 5.1-3 Fuel Assembly Design Parameters

[b., c., e.]

## 5.2 Spent Fuel Pool Cooling System Capacity Evaluation

This section examines the performance of the spent fuel pool heat exchangers and the rate at the which the pool would heat up if the heat exchange system failed.

### 5.2.1 Spent Fuel Pool Heat Exchangers

This evaluation examines the ability of the spent fuel pool cooling system to maintain spent fuel pool temperatures less than the 150 .F design temperature for a normal refueling off-load and for a full core off-load. The limiting condition is a full core off-load (121 FA.s) of the new Siemens heavy fuel assembly design. The total heat load in the pool (including background heat) resulting from this scenario has been calculated to be [b., c., e.] . This value occurs [b., c., e.] after shutdown, the minimum radiological limit, and bounds all burnups up to [b., c., e.] (assembly average burnup).

As previously stated, it is assumed that only one pump is in operation in the spent fuel pool and that it pumps water at [b., c., e.]. The spent fuel pool heat exchangers are of the parallel flow U-tube type with two tube passes and one parallel shell pass [16]. The performance of the heat exchangers was empirically determined as shown in Table 5.1-2. WPSC has stated that these original values remain applicable [17].

Knowledge of all four inlet and outlet temperatures allows the heat exchanger effectiveness, ε, to be easily calculated. This parameter is defined as the ratio of actual to maximum possible heat transfer [19] such that

$$q = \in C_{\min}(T_{h,i} - T_{c,i}) \tag{1}$$

and

$$\in = \frac{C_h}{C_{\min}}\left[\frac{T_{h,i} - T_{h,o}}{T_{h,i} - T_{c,i}}\right] = \frac{C_c}{C_{\min}}\left[\frac{T_{c,o} - T_{c,i}}{T_{h,i} - T_{c,i}}\right] \tag{2}$$

where $C_h$, $C_c$ = product of mass flow rate and specific heat of hot or cold fluid
$C_{min}$ = lesser of $C_h$ and $C_c$
$T_{h,i}$, $T_{h,o}$, $T_{c,i}$, $T_{c,o}$ are defined in section 5.1

The inlet and outlet temperatures at [b., c., e.] are (from Table 5.1-2)

[b., c., e.]

Substituting these temperatures into the above equation and multiplying both sides by

[b., c., e.]

[b., c., e.]

[b., c., e.]

At the design temperature of 150 .F and atmospheric pressure, ρ = 61.19 lbm/ft³. For single pump operation ([b., c., e.]) the mass flow rate is [b., c., e.]. At these same conditions the specific heat is about 1 BTU/lbm-F [20]. Recall that the maximum service water temperature was given as [b., c., e.]. Solving equation (1) for $T_{h,i}$ yields

[b., c., e.]

Thus even with only one pump in operation, there is adequate heat removal to allow for a full core off-load and still keep the spent fuel pool water temperature below 150°F.

Note that with a half core offload the total decay heat is [b., c., e.]. With one pump operation the maximum spent fuel pool water temperature would be

[b., c., e.]

For the other flow rates given in Table 5.1-2, the resulting values for $\varepsilon$ are as follows
At 150°F, 14.7 psi

[b., c., e.]

and from (4)

[b., c., e.]

Therefore

[b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

### 5.2.2 Heat Up Rate and Boiloff w/ Cooling System Failure

This section assumes that spent fuel pool cooling system has completely failed. A transient situation will occur in which the water in the pool will heat up to the boiling point and then boil off. From Table 5.1-1 the volumes of the pool, rack, and fuel assemblies are as follows:

| | | |
|---|---|---|
| Pool Volume (north & south) | = | [b., c., e.] |
| Rack Volume | = | [b., c., e.] |
| Fuel Assembly Volume | = | [b., c., e.] |

The total amount of water in the pool is then:

$V(\rho)$ = [b., c., e.]

The water in the pool will heat up at the rate of

$q / (m\ c_p)$ = [b., c., e.]

for a fully loaded spent fuel pool with a full core off-load. If the pool is initially at the design temperature of 150 .F, then the time to boiling is

$$t = \frac{T_{sat} - T_i}{heating\ rate}$$

[b., c., e.]

The rate of mass transfer from the pool by boiling is then

$$m = \frac{q}{h_{fg}}$$

At 14.7 psia, 212 .F, $h_{fg}$ = 970.3 BTU/lbm

The mass boil off rate is then [b., c., e.]

## 5.3 Verification of Hcating, Ventilation, and Air Conditioning System

The purpose of this section is to determine the impact of the spent fuel pool on the temperature of the air in the surrounding building. The spent fuel pool is cooled by air which enters the spent fuel building from the auxiliary building. The air passes through the spent fuel building and is vented to the atmosphere through filters. The flow rate of air into the spent fuel building is [b., c., e.] and is at a temperature of [b., c., e.] as stated in Table 5.1-1.

While most of the heat generated in the pool will be removed by the spent fuel pool heat exchangers, some will be transferred from the pool to the air in the building. Assume that a failure in the spent fuel cooling system has occurred and that only one pump is in operation, as in the first part of section 5.2.1. The maximum temperature in the spent fuel pool was calculated to be [b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

### 5.4 Spent Fuel Assembly Bulk Fluid Temperature Calculation

This section evaluates the fuel assembly peak clad temperatures and bulk fluid temperature of the spent fuel pool water in the rack.

[b., c., e.] A model of the spent fuel pool cross-section will be generated for use with [b., c., e.].

[b., c., e.]

#### 5.4.1 Flow Resistance Network and FSPLIT Model

The cooling of assemblies in the spent fuel pool is accomplished by free convection. The flow is driven by a density gradient in the assembly region of the rack. For water to circulate through the racks it must first enter through the downcomer region. A flow path is created which begins at the downcomer region and proceeds along the spent fuel pool floor. The flow then splits and enters the various rack canisters and assemblies; fnially exiting into the open area of the pool. [b., c., e.]

The flow of water through the assembly racks is modeled in

[b., c., e.]

. See Figure 5.4.1-1 for the physical configuration. Flow elements will be created for each region in the pool having the same cross-section. Flow losses are determined [b., c., e.].

[b., c., e.]

Fignre 5.4.1-1 Spent Fuel Pool Cross Section

[b., c., e.]

[b., c., e.]

Figure 5.4.1-2 Spent Fuel Pool Downcomer Region (Upper & Lower)

[b., c., e.]

[b., c., e.]

[b., c., e.]

**Table 5.4.1-1** [b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

Table 5.4.1-2 [b., c., e.]

[b., c., e.]

[b., c., e.]

Table 5.4.1-3 Average Flow Split Form Losses

[b., c., e.]

[b., c., e.]

[b., c., e.]

Table 5.4.1-4 Summary of FSPLIT Average Flow Paths [b., c., e.]

[b., c., e.]

[b., c., e.]

### 5.4.2 FSPLIT Model Validation

[b., c., e.]

[b., c., e.]

## 5.5 Spent Fuel Assembly Local Temperature Evaluation

[b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

The results show that the maximum temperature of the fuel rod surface is [b., c., e.], leaving a margin to local boiling of [b., c., e.].

### 5.6 Spent Fuel Pool Evaluation Conclusion

The analysis of the spent fuel pool cooling system showed that even with a single pump failure the pool temperature is maintained below 150°F for a full core off-load of the new Siemens heavy fuel assembly. The maximum pool temperature is [b., c., e.] for a full core offload and [b., c., e.] for a half core offload. If there is a complete failure of the cooling system then the pool will heat up at a rate no greater than [b., c., e.]. The air temperature of the spent fuel building was determined to be [b., c., e.]

. The margin to bulk boiling was determined to be [b., c., e.] while the margin to local boiling was determined to be [b., c., e.].

## 6.0 Radiological Analysis

A radiological safety analysis [32] was previously performed for design enrichments up to 5.00 wt% $U^{235}$ and for assembly average burnups up to 60 GWD/MTU. That analysis was performed using an initial heavy metal loading (IHM) of [b., c., e.] kgU and evaluated

(I) adequacy of shielding,
(2) fuel handling accident doses,
(3) adequacy of spent fuel pool ventilation system, and
(4) adequacy of spent fuel pool cleanup system.

These items were evaluated using conservative source terms calculated by the ORIGEN2 code. The conclusions of the above items, as reported in the FTI report, BAW-2095 [I], remain valid and applicable for the new Siemens 14 x 14 heavy fuel assembly with an IHM of [b., c., e.] kgU $\pm$ I%. These conclusions are re-stated and summarized below.

### (1) Adequacy of Shielding

The use of fuel with an initial enrichment of 5.00 wt% $U^{235}$, an initial IHM of [b., c., e.] kgU $\pm$ 1%, and burnups up to 60 GWD/MTU will not significantly increase any dose rates (where shielding is required). Table 6-1 presents the effects of increasing the IHM up to [b., c., e.] kgU $\pm$ 1% at I00 hours after shutdown. One hundred hours is the technical specification limit for the minimum time after subcriticality prior to movement of irradiated fuel. Table 6-1 presents the relative differences between an assembly with a [b., c., e.] kgU loading and the new Siemens 14 x 14 heavy fuel assembly with a [b., c., e.] kgU loading. The comparison shows that increasing the IHM has a negligible effect on the total energy emission.

Table 6-1 **Comparison of photon and euergy emission rates for a single assembly with different IHM loadings and an enrichment of 5.00 wt% U-235 as a function of burnup - at 100 hours after shutdown**

| | | (Based on a radial power peaking factor of 1.46) | | | | | |
|---|---|---|---|---|---|---|---|
| ORIGEN2 Group | | Ratio = (IHM [b., c., e.] kgU / IHM [b., c., e.] kgU) for several burnups (GWD/MTU) | | | | | |
| | Emean | 14 | 26 | 36.5 | 46.5 | 48 | 60 |
| 1 | 1.50E-02 | 0.99195 | 0.97934 | 0.96989 | 0.96356 | 0.96150 | 0.95784 |
| 2 | 2.50E-02 | 0.99910 | 0.99728 | 0.99631 | 0.99547 | 0.99549 | 0.99545 |
| 3 | 3.75E-02 | 1.00060 | 1.00018 | 0.99945 | 0.99747 | 0.99717 | 0.99501 |
| 4 | 5.75E-02 | 0.98615 | 0.96569 | 0.95900 | 0.95556 | 0.95496 | 0.96077 |
| 5 | 8.50E-02 | 0.99562 | 0.98793 | 0.98080 | 0.97499 | 0.97352 | 0.97025 |
| 6 | 1.25E-01 | 0.99559 | 0.98792 | 0.98135 | 0.97552 | 0.97450 | 0.96979 |
| 7 | 2.25E-01 | 0.99450 | 0.98499 | 0.97830 | 0.97193 | 0.97050 | 0.96692 |
| 8 | 3.75E-01 | 0.99826 | 0.99583 | 0.99397 | 0.99177 | 0.99151 | 0.98942 |
| 9 | 5.75E-01 | 0.99672 | 0.99285 | 0.99023 | 0.98739 | 0.98693 | 0.98545 |
| 10 | 8.50E-01 | 1.00139 | 1.00219 | 1.00207 | 1.00079 | 1.00078 | 0.99542 |
| 11 | 1.25E+00 | 0.98711 | 0.97400 | 0.95920 | 0.94421 | 0.94153 | 0.93055 |
| 12 | 1.75E+00 | 1.00166 | 1.00273 | 1.00279 | 1.00248 | 1.00277 | 1.00147 |
| 13 | 2.25E+00 | 0.99310 | 0.97441 | 0.94874 | 0.92481 | 0.92105 | 0.90536 |
| 14 | 2.75E+00 | 1.00263 | 1.00272 | 1.00465 | 1.00473 | 1.00558 | 1.00597 |
| 15 | 3.50E+00 | 1.00170 | 1.00273 | 1.00324 | 1.00362 | 1.00365 | 1.00414 |
| 16 | 5.00E+00 | 0.82877 | 0.82011 | 0.81308 | 0.80612 | 0.80590 | 0.80183 |
| 17 | 7.00E+00 | 0.82661 | 0.81999 | 0.81298 | 0.80650 | 0.80578 | 0.80155 |
| 18 | 1.10E+01 | 0.82703 | 0.81956 | 0.81310 | 0.80659 | 0.80617 | 0.80179 |
| | **Total Photon Emission** | **0.99675** | **0.99065** | **0.98544** | **0.98008** | **0.97945** | **0.97489** |
| | **Total Energy Emission** | **0.99921** | **0.99687** | **0.99407** | **0.99047** | **0.99001** | **0.98483** |

## (2) Fuel Handling Accident Doses

The radial power peaking factor (RPF) for the new Siemens 14 x 14 heavy fuel assembly is 1.70. Source terms corresponding to this peaking factor were generated for burnups up to 60 GWD/MTU using the ORIGEN2 code and an adjustment to account for the increased RPF. The off-site doses were calculated at the exclusion area boundary (EAB) and at the low population zone boundary (LPZ) using conservative source terms decayed for 100 hours and 168 hours. The parameters used to calculate the FHA doses are shown below in Table 6-2. In addition, ICRP 30 iodine dose conversion factors were used to calculate the thyroid doses. It was assumed that the FHA occurred inside containment and that the released activity from

containment to the atmosphere was unfiltered. The gap activities are shown in Table 6-3. The off-site dose results are presented in Table 6-4.
All doses are significantly less than the NUREG-0800 criteria (75 rem thyroid and 6 rem whole-body) [30].

**Table 6-2 Parameters Used in Fuel Handling Accident Analysis**

| Parameter or Assumption | Value |
|---|---|
| FHA occurs in containment | n/a |
| Core Power Level (MWt) | 1721 |
| No. of assemblies damaged | 1 |
| Number of fuel rods damaged | 179 |
| Shutdown (decay) time (hrs) | 100 and 168 |
| Power peaking factor to be applied | 1.70 |
| Release fractions into gap (per R.G. 1.25 and NUREG 5009) (%): | |
| iodine | 12 |
| noble gases | 10 |
| Krypton-85 | 30 |
| NO FILTRATION | n/a |
| Pool decontamination factor (per R.G. 1.25) | 100 |
| Atmospheric Dispersion Factors (sec/m$^3$): | |
| 2 hr EAB | 2.232E-4 |
| 2 hr LPZ | 3.977E-5 |
| Breathing rate (m$^3$/sec) | 3.47 E-4 |

## Table 6-3 Single Assembly Gap Activity

| | Gap Activity (Ci) (RPF=1.70) | |
|---|---|---|
| Nuclide | 100 hours | 168 Hours |
| Kr83m | 9.22E-09 | 2.51E-17 |
| Kr85m | 3.21E-03 | 8.66E-08 |
| Kr85 | 1.88E+03 | 1.88E+03 |
| Kr87 | 6.82E-20 | 5.63E-36 |
| Kr88 | 1.12E-06 | 6.88E-14 |
| | | |
| Xe131m | 6.98E+02 | 6.62E+02 |
| Xe133m | 1.62E+03 | 6.90E+02 |
| Xe133 | 8.83E+04 | 6.16E+04 |
| Xe135m | 5.60E-01 | 4.49E-04 |
| Xe135 | 1.66E+02 | 9.76E-01 |
| Xe138 | 0.00E+00 | 0.00E+00 |
| | | |
| I131 | 5.60E+04 | 4.40E+04 |
| I132 | 4.73E+04 | 2.59E+04 |
| I133 | 5.83E+03 | 6.05E+02 |
| I134 | 3.17E-29 | 0.00E+00 |
| I135 | 4.20E+00 | 3.36E-03 |

## Table 6-4 Off-site Fuel Handling Accident Doses.

| Dose Type | Dose (Rem) |
|---|---|
| With 100 hrs of Decay | |
| 2 hr thyroid at EAB | 48.7 |
| 2 hr whole-body at EAB | 0.147 |
| 2 hr thyroid at LPZ | 8.69 |
| 2 hr whole-body at LPZ | 0.026 |
| | |
| With 168 hrs of Decay | |
| 2 hr thyroid at EAB | 37.7 |
| 2 hr whole-body at EAB | 0.095 |
| 2 hr thyroid at LPZ | 6.72 |
| 2 hr whole-body at LPZ | 0.017 |

### (3) Adequacy of Spent Fuel Pool Ventilation System

The adequacy of the spent fuel pool ventilation system was previously re-evaluated in 1989 [1]. The evaluation was based on the current capacity of 990 spent fuel assemblies, a maximum enrichment of 5.0 wt% $U^{235}$, a maximum heavy metal loading of [b., c., e.] kgU/assembly, and a maximum burnup of 60 GWD/MTU. For the same reasons as discussed in [1], the increase in the initial heavy metal loading (from [b., c., e.] kgU to [b., c., e.] kgU maximum) will not prevent the spent fuel pool ventilation system from performing its intended function adequately.

Also, in addition to Table 6-1 of Item (1), Table 6-5 presents the relative nuclide activity differences between an assembly with a [b., c., e.] kgU loading and the new Siemens 14 x 14 heavy fuel assembly with a [b., c., e.] kgU loading (maximum of [b., c., e.] kgU). The single assembly activities were calculated using a conservative power history via the ORIGEN2 code. Tables 6-1 and 6-5 show that increasing the IHM has a small effect on the energy release and activities of the volatile nuclides for a single assembly.

Based on the conclusions of [1], and Table 6-1 and Table 6-5 results, the use of fuel with an initial enrichment of 5.00 wt% $U^{235}$, an initial IHM of [b., c., e.] kgU $\pm$ 1%, and burnups up to 60 GWD/MTU will not have a significant effect upon the spent fuel pool ventilation system, i.e. the spent fuel pool ventilation system will remain adequate for use with the new Siemens' assemblies.

**Table 6-5 Comparison of nuclide activities for a single assembly with different IHM loadings and an enrichment of 5.00 wt% U-235 as a function of burnup - at 100 hours after shutdown**

| | Ratio = (IHM [b., c., e.] kgU / IHM [b., c., e.] kgU) for several burnups (GWD/MTU) | | | | | |
|---|---|---|---|---|---|---|
| Isotope | 14 | 26 | 36.5 | 46.5 | 48 | 60 |
| | | | | | | |
| **Kr83m** | 1.00957 | 1.01761 | 1.02382 | 1.02919 | 1.02926 | 1.03332 |
| **Kr85m** | 1.01043 | 1.02120 | 1.03024 | 1.03859 | 1.03884 | 1.04640 |
| **Kr85** | 1.00418 | 1.00818 | 1.01177 | 1.01509 | 1.01550 | 1.01916 |
| **Kr87** | 1.01451 | 1.02654 | 1.03682 | 1.04659 | 1.04736 | 1.05707 |
| **Kr88** | 1.00304 | 1.02682 | 1.03395 | 1.05187 | 1.05091 | 1.05742 |
| | | | | | | |
| **Xe131m** | 0.99789 | 0.99615 | 0.99510 | 0.99393 | 0.99420 | 0.99282 |
| **Xe133m** | 1.00000 | 1.00069 | 1.00069 | 1.00000 | 0.99928 | 1.00000 |
| **Xe133** | 1.00129 | 1.00156 | 1.00196 | 1.00209 | 1.00185 | 1.00237 |
| **Xe135m** | 1.00250 | 1.00188 | 1.00189 | 1.00210 | 1.00167 | 1.00253 |
| **Xe135** | 1.00900 | 1.00910 | 1.00988 | 1.00996 | 1.00918 | 1.01006 |
| **Xe138** | 1.00000 | 1.00000 | 1.00000 | 1.00000 | 1.00000 | 1.00000 |
| | | | | | | |
| **I131** | 0.99841 | 0.99664 | 0.99593 | 0.99524 | 0.99479 | 0.99454 |
| **I132** | 0.99969 | 0.99819 | 0.99791 | 0.99705 | 0.99735 | 0.99650 |
| **I133** | 1.00214 | 1.00263 | 1.00313 | 1.00314 | 1.00240 | 1.00365 |
| **I134** | 1.01102 | 1.01322 | 1.01493 | 1.01666 | 1.01614 | 1.01863 |
| **I135** | 1.00234 | 1.00201 | 1.00202 | 1.00202 | 1.00167 | 1.00270 |

(4) **Adequacy of Spent Fuel Pool Cleanup System.**

The adequacy of the spent fuel pool cleanup system was previously re-evaluated in 1989 [1]. The evaluation was based on the current capacity of 990 spent fuel assemblies, a maximum enrichment of 5.0 wt% $U^{235}$, a maximum heavy metal loading of [b., c., e.] kgU/assembly, and a maximum burnup of 60 GWD/MTU. For the same reasons as discussed in [1] and above for the ventilation system, the increase in the initial heavy metal loading (from [b., c., e.] kgU to [b., c., e.] kgU maximum) will not prevent the spent fuel pool cleanup system from performing its intended function adequately.

# 7.0 Appendix

This appendix is divided into seven sections. Appendix 7.1 through 7.4 contains a discussion relevant to the KENO V.a bias plus uncertainty. Section 7.5 describes the determination of limiting dimensional data for the Siemens heavy fuel assembly , the NFSR, and the SFSR.

## 7.1 27 Group LRC Critical Benchmark KENO IV Resnlts

[b., c., e.]

[b., c., e.]

Table 7.1-1 KENO IV LRC Critical Results Using [b., c., e.] Library

[b., c., e.]

**Table 7.1-2 KENO IV LRC Critical Results**
**Using** [b., c., e.] **Library**

[b., c., e.]

**Table 7.1-3 KENO IV LRC Core VI Results Using Variable Generations and Densities With** [b., c., e.] **Library**

[b., c., e.]

**Table 7.1-4 KENO IV LRC Core V and XXI Results With Corrected $B^{10}$ and $B^{11}$ Densities** [b., c., e.] **Library**

[b., c., e.]

**Table 7.1-5 KENO IV LRC Critical Results Using [b., c., e.] Library**

[b., c., e.]

[b., c., e.]

## 7.2 27 Group Statistically Calculated Maximum Bias

[b., c., e.]

[b., c., e.]

### 7.3 Bias Calculations with KENO V.a.

[b., c., e.]

[b., c., e.]

[b., c., e.]

**Table 7.3-1 KENO V.a LRC Critical Results Using [b., c., e.] Library and [b., c., e.] Library**

[b., c., e.]

**Table 7.3-2 KENO V.a Core XVI results Using the** [b., c., e.] **Library and** [b., c., e.] **Library**

[b., c., e.]

**Table 7.3-3 LRC [b., c., e.] Bias Plus Uncertainty Statistical Analysis**

[b., c., e.]

**Table 7.3-4 LRC [b., c., e.] Bias Plus Uucertainty Statistical Analysis**

[b., c., e.]

### 7.4 Bias As A Function of Spacing [b., c., e.]

[b., c., e.]

## 7.5 Siemens Fuel Assembly and Storage Rack Dimensional Amplification

Section 7.5 contains an amplification and derivation of important dimensional parameters used in the criticality analysis. Section 7.5.1 addresses the Siemens heavy fuel assembly while Section 7.5.2 defines dimensions relevant to the NFSR. Section 7.5.3 addresses dimensional considerations for the SFSR.

### 7.5.1 Siemeus Heavy Fuel Assembly Dimensions

[b., c., e.]

[b., c., e.]

[b., c., e.]

### 7.5.2 NFSR Dimensions

[b., c., e.]

[b., c., e.]

[b., c., e.]

### 7.5.3 SFSR Dimeusions

[b., c., e.]

[b., c., e.]

[b., c., e.]

[b., c., e.]

## 8.0 References


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23) [b., c., e.]

24) [b., c., e.]

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ATTACHMENT 2

Letter from C. R. Steinhardt (WPSC)

to

Document Control Desk (NRC)

Dated

November 12, 1998

Framatome-Cogema Fuels Report

"Kewaunee New and Spent Fuel Storage Rack Licensing Report"

Non-Proprietary

51-1267175-01

October 1998