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1 INTRODUCTION

This report outlines the assumptions, design criteria, and methodology utilized in the criticality analysis for the South Texas Units 3 & 4 Advanced Boiling Water Reactor (ABWR) fuel storage racks. The criticality safety analysis is performed for the New Fuel Storage Vault and Spent Fuel Storage Pool preliminary rack design, which is based on a stainless steel rack using a B_4C/Al fixed neutron absorber. Furthermore, the analysis assumes that fresh and spent fuel will be stored in identical racks, with the possible exception of the number of storage locations in the rack modules installed in the two different storage locations/areas. The fuel bundle design considered in the analysis is a representative fuel design.

1.1 DESIGN CRITERIA

The design criteria are consistent with General Design Criterion (GDC) 62, Reference [1], and consider NRC guidance given in Reference [2]. Section 2 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief description of the design criteria follows.

- New Fuel Storage Vault – 10CFR50.68 [3]
  - Requires that k_{eff} at the 95/95 upper tolerance limit value is less than 0.95 (including applicable biases and uncertainties) when flooded with full density unborated water.
  - Requires that k_{eff} at the 95/95 upper tolerance limit value is less than 0.98 (including applicable biases and uncertainties) under optimum moderation conditions.
  - Note that an additional administrative margin of [ ]^{a,c} is included in this criticality safety analysis.
- Spent Fuel Storage Pool – 10CFR50.68 [3]
  - Requires that k_{eff} at the 95/95 upper tolerance limit value is less than 0.95 (including applicable biases and uncertainties) when flooded with full density unborated water.
  - Note that an additional administrative margin of [ ]^{a,c} is included in this criticality safety analysis.

1.2 DESIGN APPROACH

This criticality safety analysis does not take credit for burnup, but does take credit for the fixed poison in the racks and a modest number of fuel rods containing gadolinia as necessary. The most reactive spent fuel pool temperature (with full moderator density of 1.0 g/cm^3) is used throughout the analysis such that the results are valid over the nominal spent fuel pool temperature range (32 °F to 160 °F). Note that this is also applicable to the New Fuel Storage Vault because it was demonstrated that the highest reactivity is caused by full density water. [ ]^{a,c}.

The reactivity characteristics of the storage racks were evaluated using an [ ][a,c].
This environment was also used in the evaluation of physical tolerances and uncertainties.

2 METHODOLOGY

This section describes the methodology used to assure criticality safety when performing the analysis of the storage of fresh and depleted fuel at South Texas Units 3 & 4. The analysis methodology employs: (1) the SCALE Version 5.1 code system [4] for the criticality calculations utilizing the 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross section library, and (2) the PHOENIX4 code used for simulation of in-reactor fuel bundle depletion [8] to determine maximum reactivity with gadolinia. The latter code utilizes an ENDF/B-VI based 89 group cross section library.

The following two sections describe the application and validation of these codes in more detail.

2.1 THE SCALE VERSION 5.1 CODE AND VALIDATION

The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs.

The SCALE Version 5.1 system, used in both the benchmarking and the modeling of fuel bundle configurations, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II and KENO V.a. All references to KENO in this report refer to the KENO V.a module.

Validation of SCALE Version 5.1 for the purposes of fuel storage rack analysis is based on the analysis of selected critical experiments. The validation suite is based on fresh UO_2 experiments selected from Reference [6] and judged to be appropriate and applicable to this analysis. The suite of critical experiments is specifically selected to match the physical characteristics of the representative fuel bundles in the spent fuel pool. The fuel lattices are low enriched UO_2 clad in zirconium alloy tubes and surrounded by light water. The benchmarks used are also applicable since no burnup credit is taken in this BWR analysis. This means that the fresh UO_2 experiments are entirely applicable. Furthermore, the benchmark suite contains some experiments containing fixed poison panels containing boron as well as some experiments featuring gadolinia, Gd_2O_3, absorbers. Therefore, the utilized validation study is judged to be appropriate and applicable to this analysis. In addition, the area of applicability of the benchmark suite and trends relative to important parameters were also considered in the validation.

2.2 THE PHOENIX4 CODE AND VALIDATION

PHOENIX4 is a two-dimensional, multi-group neutron transport theory lattice code developed for nuclear design of BWRs and used to calculate the lattice physics constants for BWR fuel assemblies [8]. It solves the discrete ordinate's form of the two-dimensional neutron transport equation in Cartesian coordinates. Furthermore, PHOENIX4 uses a two-step method where a pin cell calculation is performed using an integral method to obtain a spectrum to collapse the fine group structure to a problem specific working group structure. The problem specific working group structure is then used to complete the two-dimensional S_N calculation. The multigroup cross sections are based on an ENDF/B-VI based 89-group cross section library, as stated before.

For the purpose of spent fuel criticality analysis calculations, PHOENIX4 is used for fuel bundle depletion calculations to determine its reactivity behavior as a function of burnup. PHOENIX4 has been evaluated against a set of critical experiments and fission rate data to provide verification and validation of the cross section library. The data set includes Strawbridge & Barry critical, the BAPL critical and KRITZ data (which also include assemblies containing Gd_2O_3 burnable absorbers). The critical pin cell experiments modeled geometries that cover the range of current light water reactor fuel assemblies. Comparing PHOENIX to the KRITZ fission rate data showed that the code can accurately predict local pin power distributions [8].

2.3 PEAK REACTIVITY DETERMINATION

When considering fuel rods containing gadolinia burnable absorber (BA) for reactivity hold-down, it is possible that the fuel bundle reactivity will increase as a function of depletion. This is because gadolinia, as a BA, depletes faster than the uranium fuel in the fuel bundle. Fuel bundle depletion calculations are performed in the PHOENIX4 lattice code to determine the reactivity behavior of the fuel bundle as a function of burnup. This allows for the determination of the most reactive time in the lifetime of the design basis bundle. Any reactivity increase from fresh conditions to peak reactivity must be accounted for in this analysis.

[

]a,c

After the rack model is deemed to be acceptable, depletion calculations can be performed. The bundle depletion is performed with appropriate in-core conditions and geometry. The depleted compositions are then restarted in the rack model at a range of burnups to establish the reactivity behavior of the fuel bundle in the cold rack conditions as a function of depletion.

Fuel bundles with a large number of BA pins and/or a large BA content per pin can achieve peak reactivity after Beginning-Of-Life (BOL). This peak tends to occur between [

]a,c

[

]a,c.

2.4 TOLERANCES, UNCERTAINTIES, AND BIASES

Biases and uncertainties are used to define a conservative Upper Subcritical Limit (USL). Three categories of these allowances are considered:

1. Calculational and methodology biases and uncertainties;
2. Manufacturing tolerances; and
3. Temperature bias (to cover operating range).

The approach used for the determination of the calculational and methodology biases is described below. For a given KENO Monte Carlo-calculated value of k_{eff} and associated one sigma uncertainty, a rigorous treatment is used to ensure each uncertainty is calculated as a 95% upper bound on the uncertainty. This method, or approach, determines the best estimate k_{eff} difference in the perturbed case and then adds a term for the calculation uncertainties. This term is determined by root-sum-squaring the two uncertainties and then multiplying by the one-sided 95% confidence interval multiplier. This is shown below in Equation 1:

$$\Delta k_{eff} = k_{pert} - k_{base} + M_{95/95}\left(\sqrt{\sigma_{pert}^{2} + \sigma_{base}^{2}}\right) \qquad \text{Eqn (1)}$$

where,

Δk_{eff}	=	the design basis reactivity change for a particular perturbation;
k_{pert}	=	the k_{eff} from the perturbed calculation;
k_{base}	=	the k_{eff} from the base calculation;
$M_{95/95}$	=	one-sided 95% confidence interval multiplier [11];
σ_{pert}	=	the Monte Carlo uncertainty from the perturbed calculation; and
σ_{base}	=	the Monte Carlo uncertainty from the base calculation.

Manufacturing tolerance variations can result in an increase in the system k_{eff} and are therefore considered in this analysis. Both fuel and rack manufacturing variations are evaluated. The fuel manufacturing variations include enrichment uncertainty, pellet diameter, clad thickness, pin pitch, and bundle channel thickness. The rack tolerance manufacturing variations include storage cell thickness, center-to-center bundle pitch, eccentric bundle position, neutron absorber thickness, and B-10 loading/content.

The temperature bias is considered by varying the temperature from 32 °F to 160 °F. The bounding pool temperature is [ ]a,c. Each integer temperature from [ ]a,c was run. While [ ]a,c is the limiting case, it is worth noting that there are no statistically significant differences among any of the temperatures between [ ][a,c]. This is in agreement with the expected results. Temperature increases in the SFP will result in decreased reactivity.

The uncertainties and biases described above are used to determine the maximum credible reactivity impact of these effects. This reactivity impact must be accounted for to guarantee subcritical storage of fuel in the New Fuel Storage Vault and the Spent Fuel Storage Pool. The USL, or target k_{eff}, can be determined by subtracting the sum of biases and uncertainties and additional administrative margin (see Section 1.1) from the regulatory limit. This is shown below in Equation 2:

$$USL = k_{RL} - \Delta k_{B\&U} - \Delta k_{Ad} \qquad \text{Eqn (2)}$$

where,

USL	=	upper subcritical limit;
k_{RL}	=	regulatory limit on k_{eff} from 10CFR50.68 (see Section 1.1);
$\Delta k_{B\&U}$	=	sum of biases and uncertainties; and
Δk_{Ad}	=	additional administrative margin (see Section 1.1).

Safe storage of fuel bundles can also be achieved by crediting fuel rods containing gadolinia BA. This is an acceptable approach because these BA rods are an essential feature of the bundle design used to shape power distributions and improve performance. One key feature of gadolinia as a BA is that it depletes, or burns out, faster than the uranium fuel in the bundle. It is possible, therefore, that the bundle reactivity will increase as a function of depletion. Consequently, the USL needs to account for the potential reactivity increase caused by gadolinia depletion, which is shown below in Equation 3:

$$USL = k_{RL} - \Delta k_{B\&U} - \Delta k_{Ad} - \Delta k_{Gd} \qquad \text{Eqn (3)}$$

where,

USL	=	upper subcritical limit;
k_{RL}	=	regulatory limit on k_{eff} from 10CFR50.68 (see Section 1.1);
$\Delta k_{B\&U}$	=	sum of biases and uncertainties;
Δk_{Ad}	=	additional administrative margin (see Section 1.1); and
Δk_{Gd}	=	reactivity increase to peak reactivity from gadolinia depletion.

3 ANALYSIS

This preliminary analysis, supporting the design of the fuel storage racks at South Texas Project Units 3 & 4, considers an [ ]^{a,c} arrangement in the X and Y dimensions with a square lattice pitch. No credit for burnup is taken. Furthermore, this analysis considers a fixed poison panel (BORAL) with an areal density of [ ]^{a,c}. Note that this is not the final selection of the fixed neutron material and/or the loading. To this point, all the poison materials considered are dispersions of B_4C in an aluminum matrix and are therefore equivalent for the purposes of criticality safety. The primary motivation for any change from BORAL would be an attempt to increase the longevity of the poison material. The long term performance of the poison material can be evaluated after a final selection is made in the final design of the racks. A coupon program will most likely be viewed as mandatory, both from a viewpoint of prudence and to assure compliance with recent NRC staff guidance on monitoring fixed poisons in spent fuel pools as part of the plant aging process [9].

Also, no soluble boron is present in this analysis. The only other absorber of note credited in this analysis is gadolinia, Gd_2O_3.

3.1 ASSUMPTIONS

The following assumptions are used throughout the criticality safety analysis:

- It is assumed that fresh and spent fuel will be stored in identical racks.
- A representative fuel design is used in the design.
- A preliminary rack design is employed with assumed tolerances based on recent experience with fuel storage of this type. The preliminary rack design features stainless steel racks using B_4C/Al fixed neutron absorbers.
- It is assumed that a potential fuel drop accident will not cause a crush zone at the top of the cell that is large enough to impact the neutron absorber. The neutron absorber will continue to completely cover the active fuel region of the fuel bundle. Furthermore, the fuel bundle is assumed to remain intact.

3.2 POSTULATED ACCIDENT SCENARIOS

A range of abnormal conditions, or accident scenarios, can occur in the spent fuel pool. The rearrangement of storage modules caused by a seismic event along with fuel handling or misplacement scenarios must be considered.

A hypothesized earthquake could cause a relocation of rack modules within the spent fuel pool and consequently reduce the spacing between modules due to a lateral shift. The potential accident scenario is considered (or covered) by not taking any credit for spacing, and the conservative analysis model being infinite in the lateral (X and Y) directions, which bounds any reactivity caused by a reduction in inter-module spacing.

A fuel bundle could be dropped while being handled or it could be accidentally misplaced. Through a potential handling drop accident scenario, the fuel bundle could come to rest horizontally on the top of the fuel storage racks. Since the top of the racks is taller than the fuel bundles stored in the racks, the horizontally dropped fuel bundle and the fuel bundles stored in the racks will be neutronically decoupled. In other words, the drop will not cause an increase in reactivity. This conclusion is based on the assumption that the crush zone caused by the fuel bundle drop onto the top of the racks remains above the active fuel region (see Section 3.1).

A fuel bundle drop or an accidental misplacement could cause a fuel bundle to either be placed in a storage cell not qualified for storage or outside of the storage racks (space permitting). The event of a fuel bundle being accidentally misplaced inside the storage rack is bounded by the conservative analysis model which utilizes an infinite lateral array of storage cells with a fuel bundle in every location. This means that any fuel bundle design that meets the fuel loading requirements will be allowed to be stored in any location in the pool at any time in the life cycle of the bundle. Therefore, this bounds the reactivity increase caused by any fuel bundle drop or mislocation event within the spent fuel storage racks.

The second scenario of mislocating a fuel bundle outside of the storage rack is explicitly modeled and evaluated. A series of models were constructed to investigate the reactivity impact of several scenarios involving a fuel bundle placed outside the storage racks. [

]^{a,c} These mislocating/misloading scenarios were considered both with and without fixed poison panels along the outside face of the rack modules. The results indicate that while some scenarios do increase reactivity slightly over the base case, none of them cause an increase over the USL.

4 SUMMARY

A methodology has been established, and is presented herein, for performing criticality safety analyses for the South Texas Units 3 & 4 ABWR fuel storage racks in the New Fuel Storage Vault and Spent Fuel Storage Pool. The methodology will be employed to establish storage requirements for the final fuel and rack designs. Furthermore, assumptions and design criteria used in the criticality safety analysis are also outlined herein. The criticality safety analysis based on representative fuel and preliminary rack design shows a comfortable margin to the regulatory limits. Also, an additional administrative margin of [ ]a,c is reserved.

5 REFERENCES

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2. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, August 19, 1998.

3. Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements."

4. "SCALE: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation," ORNL/TM-2005/39, Version 5.1, Vols. I–III, November 2006. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.

5. Not Used.

6. "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC(95)03, Volume IV, Nuclear Energy Agency, OECD, Paris, September, 2008.

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