ATTACHMENT 5

CRITICALITY SAFETY EVALAUTION REPORT
(NON-PROPRIETARY VERSION)

The following pages provide the non-proprietary version of the criticality safety analysis report provided by HOLTEC International supporting this license amendment request.

HI-2220020, "Criticality Safety Analysis of SFP for Callaway," Revision 1
[NON-PROPRIETARY]

139 pages follow this cover sheet

Revision Log

Revision	Description of Changes
0	Initial issue.
1	The client comments are incorporated, and changes are marked by a revision bar.

EXECUTIVE SUMMARY

This report documents the criticality safety analyses of the spent fuel pool performed for the Callaway Unit 1, which contains a single type of BORAL™ spent fuel racks designed for storage of the PWR 17x17 fuel assemblies. The criticality evaluations qualify the spent fuel racks loaded with two storage configurations, including uniform loading of spent fuel assemblies with various cooling times and a checkerboard configuration of fresh fuel assemblies and empty storage cells. The purpose of this report is to provide a complete up-to-date criticality safety evaluation based on the latest methodologies consistent with the current NRC expectations. The difference between this analysis and the analysis of record is an extended list of qualified fuel assemblies, simplified loading configurations (regions) that no longer include the MZTR (Mixed-Zone Three-Region) approach, and a new analysis methodology. There is no change of the spent fuel racks.

The analysis of fuel irradiation during core operation is performed with CASMO5 Version 2.08.00, a multigroup two-dimensional transport theory code based on the Method of Characteristics, using the ENDF/B-VII Library. The criticality calculations are performed with MCNP5 Version 1.51, a three-dimensional continuous energy Monte Carlo code, using continuous energy cross-section data predominantly based on ENDF/B-VII. To account for different temperatures, [redacted]

For a storage configuration with spent fuel assemblies, the minimum required burnups as a function of enrichment (a third-order polynomial fit) have been determined, considering various cooling times. All credible normal and accident conditions have been analyzed, and the results of the calculations show that the effective neutron multiplication factor (k_{eff}) of the spent fuel pool loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with unborated water and does not exceed 0.95 for the pool flooded with borated water, all for 95% probability at a 95% confidence level, in accordance with 10 CFR 50.68(b)(4).

All credible interface conditions in the spent fuel pool have been considered and the storage configurations interface criteria are established. Fuel assembly reconstitution activities and storage of fuel rod storage racks are also considered and qualified.

An evaluation of the potential reactivity effect of a degradation of the BORAL™ performance, and available margin in the criticality analysis to possibly offset such degradation is also performed.

Table of Contents

List of Tables

List of Figures

List of Abbreviations

ACPL	Holtec approved computer program list
BPR	burnable poison rod
CCW	component cooling water system
CFR	U.S. Code of Federal Regulations
FRSR	fuel rod storage rack
IBA	integral burnable absorber
IFBA	integral fuel burnable absorber
ID	inner diameter
MAFP	minor actinides and fission products
OD	outer diameter
PWR	pressurized water reactor
RCCA	rod cluster control assembly
RMWST	reactor makeup water storage tank
RWST	refueling water storage tank
SFP	spent fuel pool
SFR	spent fuel rack
SS	stainless steel
TCF	total correction factor
WABA	wet annular burnable absorber

1.1 PURPOSE

This report documents the criticality safety analyses of the spent fuel pool performed for the Callaway Unit 1. The SFP contains a single type of BORAL™ spent fuel rack designed for storage of the PWR 17x17 fuel assemblies. The criticality safety analysis of record for the Callaway SFP is documented in [1]. The purpose of the analyses presented in this report is to provide a complete up-to-date criticality safety evaluation for the Callaway SFP based on the latest methodologies consistent with current NRC expectations in [2] and [3], which will result in a replacement of the analysis of record. The difference between this analysis and the analysis of record is an extended list of qualified fuel assemblies, simplified loading configurations (regions) that no longer include the MZTR (Mixed-Zone Three-Region) approach, and a new analysis methodology. There is no change of the spent fuel racks.

The criticality control in the SFRs relies on various combinations of the following:

- Fixed neutron absorbers: BORAL™ poison panels;
- Burnup of spent fuel assemblies;
- Spent fuel cooling time;
- Empty SFR storage cells;
- Soluble boron in the SFP.

Specifically, the criticality evaluations qualify the SFRs loaded with the following configurations, hereinafter referred to as loading regions (see Figure 1-1):

- Region 1 – a 2x2 checkerboard pattern with two fresh fuel assemblies and two empty storage cells. No credit of the IBA and soluble boron (under normal conditions) in the SFP is applied;
- Region 2 – uniform loading of spent fuel assemblies with a credit of various cooling times and soluble boron in the SFP.

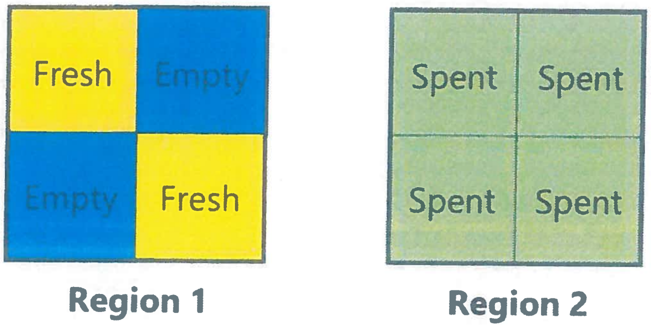

Figure 1-1 – Callaway SFR Permissible Loading Configurations

Additionally, the criticality evaluations are performed for the following:

- Normal conditions:
  - credible interface conditions in the spent fuel pool;
  - fuel movement, insertion, and removal operations;
  - storage of fuel rod storage racks;
  - specific Callaway fuel inventory, such as the fuel assemblies with the missing rods;
  - fuel assembly reconstitution activities;
- Abnormal and accident conditions.

An evaluation of the potential reactivity effect of a degradation of the BORAL™ performance, and available margin in the criticality analysis to possibly offset such degradation is also performed.

A summary of physical changes, technical specification changes and analytical scope is provided in Appendix A, Appendix B and Appendix C.

2.1 ACCEPTANCE CRITERIA

Codes, standard, and regulations or pertinent sections thereof that are applicable to the analysis include the following:

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."
- Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements".
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.1, Criticality Safety of Fresh and Spent Fuel Storage and Handling, Rev. 3 – March 2007.
- US NRC Regulatory Guide RG 1.240, "Fresh and Spent Fuel Pool Criticality Analyses," March 2021.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- USNRC, NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology, January 2001.
- DSS-ISG-2010-01, Revision 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools.
- Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water-Reactor

Power Plants, NEI 12-16, Revision 4, Nuclear Energy Institute.

The objective of this analysis is to ensure that the effective neutron multiplication factor (k_{eff}) of the SFP loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with unborated water and does not exceed 0.95 for the pool flooded with borated water, all for 95% probability at a 95% confidence level, in accordance with 10 CFR 50.68(b)(4).

3.0 METHODOLOGY

3.1 General Approach

The analysis is performed in a manner such that the results are below the regulatory limit with a 95% probability at a 95% confidence level. The calculations are performed using either the worst-case bounding approach or the statistical analysis approach with respect to the various calculation parameters. The approach considered for each parameter is discussed below. These calculations are used to determine the final k_{eff} used to show compliance with the regulatory limits for both normal and accident conditions. The accident calculations are essentially modifications of the design basis cases for the normal conditions but do not introduce a new fuel or rack design change. Therefore, the uncertainty and bias calculations for the normal conditions are applicable and do not need to be repeated for the accident calculations.

3.2 Computer Codes and Cross-Section Libraries

3.2.1 CASMO5 Overview

The analysis of fuel irradiation during core operation is performed with CASMO5 Version 2.08.00, which is commonly used in the industry for reactor analysis (depletion) when providing reactivity data for specific 3D simulator codes, and it has been reviewed by the NRC [4]. CASMO5 is a multigroup two-dimensional transport theory code for burnup calculations of BWR and PWR fuel assemblies based on the Method of Characteristics [5] and it is developed by Studsvik of Sweden [6]. CASMO5 allows modeling of a planar cross section of an individual fuel assembly, including all relevant details such as individual pellet and cladding diameters, locations of guide tubes and instrument tube, and material compositions of all materials. The calculations assume a planar and axially infinite array of fuel assemblies. The fuel and absorber dimensions, the operating parameters, an initial enrichment and a maximum burnup are provided in the input model, and the atom densities for actinides and fission products in the isotopic composition of spent fuel are determined by CASMO5. For all CASMO5 depletion calculations, the ENDF/B-VII Library [7] is used. Although CASMO5 has been extensively benchmarked, a code validation (i.e., bias and bias uncertainty) is not considered necessary because it is not used for reactivity calculations. Nonetheless, to ensure that CASMO5 produces reliable and predictable results, the uncertainty on the irradiated fuel isotopic composition (i.e., the number densities) is considered as discussed in Subparagraph 3.3.6.6.1, which covers all uncertainties associated with depletion, such as uncertainty in computation of the isotopic inventory by the depletion code, uncertainty in cross-sections (both actinides and fission products), etc.

3.2.2 MCNP5 Overview

MCNP5 Version 1.51 [8], a three-dimensional continuous energy Monte Carlo code developed at Los Alamos National Laboratory, is used for the criticality analyses. This code offers the capability of performing full three-dimensional calculations of the spent fuel storage racks. It has a long history of successful use in fuel storage criticality evaluations and has all of the necessary features for evaluation of the Callaway SFP. MCNP5 calculations use continuous energy cross-section data predominantly based on ENDF/B-VII [9]. To account for different temperatures, [redacted]

[redacted]

3.2.2.1 MCNP5 Validation

The benchmarking of MCNP5-1.51 is based on the guidance in [2] and [12], and includes calculations for a total of 562 critical experiments with fresh UO_2 fuel, fresh MOX fuel, and fuel with simulated actinide composition of spent fuel (Haut Taux de Combustion (HTC) experiments), chosen, in so far as possible, to bound the range of variables in the SFP designs. Validation of MCNP5-1.51 and continuous energy ENDF/B-VII data library to perform criticality safety calculations is documented in [13]. The validation confirms the accuracy of the calculational methodology to predict subcriticality. Validation includes identification of the difference between calculated and experimental neutron effective multiplication factor (k_{eff}), called the bias. The range of the benchmark parameters used to validate the calculational methodology primarily defines the area of applicability (AOA), which establishes the limits of the systems that can be analyzed using the validated criticality safety methodology. The applicable range of key parameters for the design application and benchmarks are summarized in Table 3-1.

The results of the benchmarking calculations for the full set of all 562 experiments and various subsets are presented in Table 3-2 along with an estimate of significance of the observed trends. Following the guidance in [2], the statistical treatment used to determine those values considered the variance of the population about the mean and used appropriate confidence factors and trend analysis. This is also consistent with the requirement in [3]. Trend analyses are performed for various subsets and parameters in [13], and the determined significant trends are presented in Table 3-2. In order to determine the maximum bias and bias uncertainty that are applicable to the criticality calculations in this report, the trend equations from [13] are evaluated in Table 3-3 for the specific parameters used in the current analysis.

Based on the results presented in Table 3-2 and Table 3-3, the maximum MCNP5 code validation bias and bias uncertainty associated with the appropriate benchmark subsets are determined for each loading region separately for unborated and borated water, and summarized in Table 3-4. The appropriate maximum bias and bias uncertainty are applied to all calculations to determine the maximum k_{eff}.

3.3 Analysis Methods

As discussed in Section 3.1, the calculations are performed using either the worst-case bounding approach or the statistical analysis approach with respect to the various calculation parameters. These bounding inputs and assumptions for the fuel and storage rack models are summarized below:

Bounding Fuel Designs and Fuel Assembly Parameters:

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Bounding Storage Rack Parameters:

[redacted]

[redacted]

- [redacted]

Bounding SFP Moderator Temperature:

[redacted]

In addition to the conservative inputs and assumptions discussed above, the base MCNP5 model used for the analysis is made as follows (with variations evaluated in the following subsections):

[redacted]

[redacted]

[redacted]

[redacted]

In order to determine the reactivity effect for the parameter of interest with a 95% probability at a 95% confidence level, two calculations with a reference and modified parameter are performed, and the following Equation 3.3-1 is applied, where $\pm 2 \times \sqrt{\sigma_1^2 + \sigma_2^2}$ is called the 95/95 uncertainty.

$$\Delta k_{calc} = \left(k_{calc_2} - k_{calc_1}\right) \pm 2 \times \sqrt{(\sigma_1^2 + \sigma_2^2)} \qquad \text{Equation 3.3-1}$$

The established maximum Δk_{calc} is then either statistically combined with the other uncertainties to determine the maximum k_{eff} value or the bounding parameter's value is incorporated into the design basis model.

Such bounding approach provides analysis simplicity and additional safety margin. The MCNP5 design basis model is shown in Figure 3-1. Additional details and analysis methodology discussions are provided in each subsection below.

3.3.1 Design Basis Fuel Assembly Design

The Callaway SFP contains various fuel assembly designs. The reactivity calculations are performed for the representative fuel types in Table 5-1, which bound all variations of the fuel designs in the Callaway SFP, using the 2x2 rack model discussed in Section 3.3. The following cases are evaluated:

- Case 3.3.1.1: Westinghouse 17x17 Standard (STD);
- Case 3.3.1.2: Westinghouse 17x17 Optimized (OFA);
- Case 3.3.1.3: Westinghouse 17x17 Vantage+ (V+);
- Case 3.3.1.4: Framatome GAIA 17x17 (GAI).

The fuel assembly designs that show the highest reactivity are used in all design basis criticality calculations.

3.3.2 Fuel Assembly Parameters

The reactivity effects of the fuel assembly parameters due to manufacturing tolerances are evaluated using the 2x2 rack model discussed in Section 3.3. The variation of these parameters (see Table 5-2) is applied to all fresh and spent fuel assemblies in the SFR model (if applicable). [redacted]. The following variations of the parameters are therefore considered for the bounding fuel assembly design:

- Case 3.3.2.0: Reference case. All fuel parameters are nominal;
- Case 3.3.2.1: Minimum cladding OD;
- Case 3.3.2.2: Maximum cladding OD;
- Case 3.3.2.3: Minimum fuel rod pitch;
- Case 3.3.2.4: Maximum fuel rod pitch;
- Case 3.3.2.5: Minimum fuel pellet OD;
- Case 3.3.2.6: Maximum fuel pellet OD;
- Case 3.3.2.7: Maximum fuel enrichment;

- Case 3.3.2.8: Maximum fuel density;
- Case 3.3.2.9: Maximum ^{10}B loading in the IFBA rods (spent fuel).

Separate depletion calculations are performed for Cases 3.3.2.1 through 3.3.2.9 so that the effect of the tolerance during depletion is accounted for. The reactivity effect of each parameter is determined using Equation 3.3-1. The maximum positive (if any) reactivity effect for each parameter is then selected and this maximum value is statistically combined with the other maximum values from every tolerance calculation to determine the combined reactivity effect of the fuel manufacturing tolerances.

3.3.3 Spent Fuel Rack Parameters

The minimum neutron absorber (BORAL™) panel length is slightly larger than the active fuel length, while the distance from the bottom of a fuel assembly to the beginning of active fuel region vary, thus slightly exposing active fuel above the top end of the absorber panel. [redacted]

The reactivity effects of the SFR parameters due to manufacturing tolerances are evaluated using the 2x2 rack model discussed in Section 3.3. Since a laterally infinite array of storage cells is used in the design basis calculations, a thicker stainless steel sheathing on the outside of the SFR is not included in the model, hence its manufacturing tolerances are not considered. In accordance with [3], the following variations of the SFR parameters are considered:

- Case 3.3.3.0: Reference case. All rack parameters are nominal except the BORAL™ panel length and ^{10}B loading;
- Case 3.3.3.1: Minimum storage cell ID;
- Case 3.3.3.2: Maximum storage cell ID;
- Case 3.3.3.3: Minimum storage cell pitch;
- Case 3.3.3.4: Maximum storage cell pitch;
- Case 3.3.3.5: Minimum storage cell wall thickness;
- Case 3.3.3.6: Maximum storage cell wall thickness;
- Case 3.3.3.7: Minimum sheathing thickness;
- Case 3.3.3.8: Maximum sheathing thickness;

- Case 3.3.3.9: Minimum poison thickness. [redacted]
- Case 3.3.3.10: Minimum poison width.

The reactivity effect of each parameter is determined using Equation 3.3-1. The maximum positive (if any) reactivity effect for each parameter is then selected and this maximum value is statistically combined with the other maximum values from every tolerance calculation to determine the combined reactivity effect of the rack manufacturing tolerances.

3.3.3.1 BORAL™ Panel B_4C Particle Size

BORAL™ is a composite of finely ground boron carbide particles dispersed in the metal matrix of pure aluminum to act as a neutron absorber. The BORAL™ documentation package [14] includes the results of various inspections and tests performed for each shipment of the BORAL™ material that was eventually used for fabrication of the SFRs for Callaway. Based on the results of sieve analyses, a typical distribution of the B_4C particle size in BORAL™ is the following:

- 0 – 45 μm: [redacted]
- 45 - 180 μm: [redacted]
- 180 - 300 μm: [redacted]
- over 300 μm: [redacted]

In order to investigate the reactivity effect of the B_4C particle size, calculations for the heterogeneous poison panels (B_4C particles in aluminum matrix) are performed. Inside the heterogeneous model of the poison panel, [redacted]

The following cases are evaluated using the 2x2 rack model discussed in Section 3.3:

- Case 3.3.3.1.0: Reference case. Homogeneous BORAL™;
- Case 3.3.3.1.1: Heterogeneous BORAL™ with B_4C particle size of 45 μm;
- Case 3.3.3.1.2: Heterogeneous BORAL™ with B_4C particle size of 180 μm.

[redacted] The results are compared to estimate a difference between the homogeneous and heterogeneous model with the variable B_4C particle size.

3.3.4 Spent Fuel Pool Water Temperature

The criticality analysis should be performed at the most reactive SFP temperature and density [3], and the temperature-dependent cross-section effects in MCNP need to be considered. In general, both density and cross-section effects are not necessarily the same for all storage rack scenarios, since configurations with strong neutron absorbers typically show a higher reactivity at lower water temperature, while configurations without such neutron absorbers typically show a higher reactivity at a higher water temperature. Hence for the Callaway SFRs with BORAL™ poison, the minimum SFP water temperature and maximum density are expected to produce the maximum reactivity condition.

[redacted]

Studies are performed to demonstrate the reactivity effect of the moderator temperature and density over the temperature range specified in Table 5-8 using the temperature adjusted cross-sections and S(α,β) cards. The bounding temperature is determined using the same 2x2 rack model as discussed in Section 3.3. The following studies are performed:

- Case 3.3.4.0: [redacted]
- Case 3.3.4.1: [redacted]
- Case 3.3.4.2: [redacted]

- Case 3.3.4.3: [redacted]

The bounding moderator temperature and density using temperature adjusted cross-sections and S(α,β) cards are considered in all design basis calculations.

3.3.5 Fuel Assembly Radial Positioning and Orientation

3.3.5.1 Fuel Assembly Radial Positioning

A fuel assembly located in the center of a SFR cell is surrounded by equal amounts of water on all sides, and hence the thermalization of the neutrons that occur between the assembly and the poison panel, and also the effectiveness of the poison is equal on all sides. For an eccentric positioning, the effectiveness of the poison is now reduced on those sides where the assembly is located close to the cell walls and increased on the opposite sides. This creates a compensatory situation for a single cell, where the net effect is not immediately clear. However, for a group of storage cells or entire SFR, and for the condition where all assemblies are located closest to the center of this group, the fuel assemblies at the center are now located close to each other, separated by poison plates with a reduced effectiveness since they are not surrounded by water on the outer sides. This may become the dominating condition in terms of reactivity increase.

The fuel assembly radial positioning is evaluated using the 2x2 rack model discussed in Section 3.3, which evaluate more local effects, as well as larger arrays that represent an entire rack, and therefore captures global positioning effects. The following fuel radial positioning configurations are considered:

- Case 3.3.5.0: Reference case. All assemblies are centered in their fuel storage cell of the 2x2 rack model;
- Case 3.3.5.1: All assemblies in the 2x2 rack model are moved as closely to the center of the model as permitted by the rack geometry. This creates a laterally infinite arrangement of 2x2 arrays where the assemblies are close together in each 2x2 array. Note that a configuration with assemblies moved away from the center in each 2x2 array would be equivalent due to the boundary condition and is therefore not separately considered;
- Case 3.3.5.2: All assemblies in the 2x2 rack model are moved towards the same corner of the cell. This creates a laterally infinite arrangement of 2x2 arrays where the assemblies are moved towards the same corner;
- Case 3.3.5.3: All assemblies in the 8x8 rack model are moved as closely to the center of the model as permitted by the rack geometry. This essentially represents the entire SFR and therefore captures global positioning effects. A periodic boundary condition is also used on the periphery of the model.

This neglects the gap between adjacent racks and is therefore conservative and simplifies model generation.

The bounding configuration is conservatively considered in all design basis calculations.

3.3.5.2 Fuel Assembly Radial Orientation

The rotation of the fuel assembly in the core and/or in the storage rack is possible. However, since the analyzed fuel lattices have uniform radial fuel enrichment and symmetric radial distribution of the IFBA and Gd rods, the fuel assembly orientation is expected to have a negligible effect on reactivity. Therefore, the fuel assembly orientation is not evaluated further in the report.

3.3.6 Spent Fuel Reactivity Calculation

To perform the criticality evaluation for spent fuel in MCNP5, the isotopic composition of the fuel material is calculated with the depletion code CASMO5 and then specified as input data into MCNP5. Table 3-5 provides a list of spent fuel isotopes credited in MCNP5. [redacted]

[redacted]

Assembly average isotopic compositions are extracted from the CASMO5 output files and used in the MCNP models, i.e., applied equally to all fuel rods or fuel rod segments with the corresponding burnup. [redacted]

3.3.6.1 *Core Operating Parameters*

Principal operating parameters of the fuel discussed here are moderator temperature, fuel temperature, soluble boron concentration in the core, and the power density. Other parameters such as axial burnup distribution and the effect of burnable absorbers are discussed in following paragraphs. Generic studies in [16] and [17] indicate that the operating parameters that result in higher reactivities are the upper bound moderator temperature, fuel temperature, and soluble boron concentration. The power density has a comparatively small effect with no clear trend. Also, a lower bound power density would be inconsistent with the higher fuel temperature. Consistent with the guidance in [2] and [3], the upper bound values are therefore used for all four parameters.

To show the effect of the individual parameters and confirm that the selected values are in fact conservative, the following sensitivity studies are performed using the 2x2 rack model discussed in Section 3.3:

- Case 3.3.6.1.0: Reference case. Upper bound values are used for all parameters;
- Case 3.3.6.1.1: Fuel temperature is increased by 300 K;
- Case 3.3.6.1.2: Fuel temperature is decreased by 300 K;
- Case 3.3.6.1.3: Moderator temperature is increased by 100 K;
- Case 3.3.6.1.4: Moderator temperature is decreased by 100 K;
- Case 3.3.6.1.5: Soluble boron is increased by 300 ppm;
- Case 3.3.6.1.6: Soluble boron is decreased by 300 ppm;
- Case 3.3.6.1.7: Specific power is increased by 5 MW/mtU;
- Case 3.3.6.1.8: Specific power is decreased by 5 MW/mtU.

Separate depletion calculations are performed for the parameters above.

3.3.6.2 *Reactivity Effect of Cooling Time*

Evaluations are performed to estimate the reactivity effect of cooling time using the 2x2 rack model discussed in Section 3.3. The following cooling times are considered:

- Case 3.3.6.2.0: Reference case. Cooling time of 0 hours;
- Case 3.3.6.2.1: Cooling time of 48 hours;
- Case 3.3.6.2.2: Cooling time of 72 hours;
- Case 3.3.6.2.3: Cooling time of 500 hours;
- Case 3.3.6.2.4: Cooling time of 1 year;
- Case 3.3.6.2.5: Cooling time of 5 years

- Case 3.3.6.2.6: Cooling time of 10 years;
- Case 3.3.6.2.7: Cooling time of 20 years.

Separate depletion calculations are performed for the cooling times above.

3.3.6.3 *Integral and Removable Burnable Absorbers*

Fuel assemblies operated at Callaway can contain various forms of control components during in-core depletion, such as Pyrex, WABA, and RCCA, hereinafter referred to as the fuel inserts. All these components are inserted into the guide tubes of the assembly during depletion. Additionally, assemblies can contain IBAs, consisting of neutron absorbing material mixed into the fuel pellet (Gadolinia) or added as a coating on the fuel pellet (ZrB_2). Below, each of these devices is briefly described, and its reactivity effect is characterized. At the end of this subsection, the approach taken in the burnup credit evaluation is outlined.

3.3.6.3.1 Burnable Poison Rods

The rods contain a certain amount of ^{10}B, in the form of Al_2O_3-B_4C or SiO_2-B_2O_3 in annular pellets inside a Zircaloy or SS cladding. Axially, the poisoned area is smaller than the active fuel length (see Table 5-5), and it is axially centered. There are different versions of these components with a different amount of ^{10}B, which is achieved by varying the number of rods. At the end of the first fuel cycle the ^{10}B is practically depleted, and the component is usually removed from the assembly for the subsequent cycles.

The detailed studies [18] have been performed on the reactivity effect of BPRs. The results of these studies show that the presence of the burnable poison rods results in an increase of the reactivity of the assembly. This is a result of the reduction of water in the assembly (the poison rods replace the water usually present in the guide tubes) and the presence of the neutron absorber, which both cause a hardening of the neutron spectrum, thereby increasing the plutonium production which in turn increases reactivity. The longer the poison rods remain in the assembly, the larger is the resulting increase in reactivity. However, if the poison rods are removed after the first cycle, which is usually the case, the increase in reactivity is limited, with a maximum of 0.012 Δk reported for the studies in [18], compared to an assembly with guide tubes filled with water.

3.3.6.3.2 Rod Cluster Control Assemblies

Control rods consist of highly neutron absorbing material inside the SS cladding and they are used for short term reactivity control in the core. Two different RCCA types were utilized in the Callaway reactor core. Specifically, the reactor operation up through Cycle 3 was controlled using the RCCAs made of Hafnium-Zirconium (Hf-Zr), while all subsequent and future cycles utilize the RCCAs made of Silver-Indium-Cadmium (Ag-In-Cd). They are connected to a control rod drive which allows axial movement of the RCCA during the reactor operation.

Typically, at full power, most RCCAs are completely withdrawn from the active region of the fuel, while some RCCAs may be inserted only slightly into the active region. [redacted], and the insertion depths considered in the analysis for the Hf-Zr and Ag-In-Cd RCCAs are presented in Table 5-5. [redacted].

3.3.6.3.3 Integral Burnable Absorber

IBAs are integral to the fuel rods, and therefore do not replace water in the guide tubes. Consequently, the spectrum hardening effect of the IBAs during irradiation, and therefore the reactivity effect, is significantly lower compared to the fuel inserts. The impact of burnable absorbers on reactivity has been studied extensively in [23] and the results show that in many cases the reactivity effect is negative, i.e., reducing the reactivity of the assembly. Specifically, it is concluded that for UO_2-Gd_2O_3 rods, the criticality evaluations may conservatively neglect the presence of the IBAs by assuming non-poisoned equivalent enrichment fuel [3]. For IFBA rods, a small positive reactivity effect is identified in [23], with a maximum Δk of 0.01.

3.3.6.3.4 Approach Used in the Burnup Credit Evaluation

To confirm the potential reactivity effects of integral and removable burnable absorbers applicable to Callaway fuel, several studies are performed using the 2x2 rack model discussed in Section 3.3 for selected cases where the fuel inserts and/or IBAs are explicitly modeled in the depletion analyses, so that the spent fuel composition transferred to the MCNP criticality calculation (without any residual absorber) contains the effect of the burnable absorbers.

The specific usage of the fuel inserts and IBA at Callaway is considered in accordance with Paragraph 4.2.1 of [3], as follows:

[redacted]

- [redacted];

[redacted]

[redacted]

[redacted]

- [redacted].

The conditions above bound all potential and hypothetical reactivity effects resulting from the integral and removable burnable absorbers. The following cases are evaluated:

- Case 3.3.6.3.0: Reference case. No IBA and inserts (empty guide tubes filled with water);
- Case 3.3.6.3.1: Pyrex rods;
- Case 3.3.6.3.2: WABA rods;
- Case 3.3.6.3.3: RCCA Ag-In-Cd rods (partial insertion);
- Case 3.3.6.3.4: RCCA Hf-Zr rods (partial insertion);
- Case 3.3.6.3.5: 104 IFBA rods;
- Case 3.3.6.3.6: 200 IFBA rods;
- Case 3.3.6.3.7: WABA rods and 104 IFBA rods;
- Case 3.3.6.3.8: WABA rods and 200 IFBA rods.

Separate depletion calculations are performed for the cases above. The bounding configuration is conservatively considered in all design basis calculations.

3.3.6.4 Axial Burnup Distribution

Irradiated Fuel Assemblies are not burned evenly over the height of the assembly. Rather, they exhibit an axial burnup distribution, i.e., the burnup of the fuel is a function of the axial location of the fuel within the assembly. In general, the fuel at the top and bottom end of the assembly shows a lower burnup than the fuel in the axial center of the assembly. This is caused by the increased neutron loss and therefore decreased neutron flux towards the top and bottom end of the assembly during irradiation in the reactor core. The reactivity of spent fuel is a strong function of the fuel burnup, with the reactivity decreasing when the burnup increases. For irradiated fuel assemblies, the reactivity at the top and bottom ends is therefore higher than the reactivity in the center of the assembly. Obviously, no axial burnup distribution is applicable in the analysis of fresh fuel. However, when credit is taken for the reduction in reactivity due to the burnup of the fuel, it is important that the axial burnup distribution is accounted for in a conservative way. Therefore, bounding axial burnup profiles need to be established, i.e., axial profiles which maximize the reactivity under the given conditions.

There is no direct or theoretical method to establish a bounding axial burnup profile for a given assembly at a given average burnup. Since the plant-specific axial burnup profiles are not available from the Callaway unit operation, bounding profiles are established based on Subsection 6.E.4.1 of [15] by analyzing databases that contain profiles for a large number of assemblies from different plants. The source of profiles is the axial burnup database documented in [24] developed by Yankee Atomic Engineering Corporation (YAEC) and the available axial burnup distributions documented in the CRC [25]. From this database, assembly profiles for the Westinghouse 17x17 fuel type, which is identical or similar to those used at Callaway, are selected for determining the bounding axial burnup profile. The combined total number of profiles is 1034 for the Westinghouse 17x17 assemblies with an enrichment range of 1.60 - 4.62 wt% ^{235}U and a burnup range of 2.1 – 53.5 GWd/mtU. [redacted]. The example bounding axial burnup profiles for WE 17x17 fuel generated for several burnup points are provided in Table 3-6. Additionally, the bounding axial burnup profiles established in NUREG/CR-6801 [26] and presented in Table 3-7 have been considered.

The calculations are performed to demonstrate the reactivity effect of the axial burnup distribution, using the 2x2 rack model discussed in Section 3.3. The following cases are evaluated:

- Case 3.3.6.4.1: Uniform burnup profile;
- Case 3.3.6.4.2: Bounding WE 17x17 profile (Table 3-6);
- Case 3.3.6.4.3: Bounding NUREG profile (Table 3-7).

The axial burnup profile that shows the highest reactivity is used in all design basis criticality calculations, unless noted otherwise.

Although the bounding profiles are based on a large number of assembly profiles, it is possible that there are some existing or future assemblies that might be outside the bounds of these databases. However, due to the large number and variety of profiles in these databases, any assembly exceeding the bounding profile would be considered a rare exception. Furthermore, for all assemblies in the SFR it is assumed that they have the bounding axial burnup distribution. To have an adverse effect on reactivity it would be necessary that the SFR contains a few of these exceptional assemblies, close to each other, and that all other assemblies in the SFR have the bounding burnup distribution. This is extremely unlikely. Therefore, these databases are considered to be sufficient for the determination of the bounding axial profiles and the profiles established in [15] and [26] are applicable for the burnup credit in the Callaway SFP.

3.3.6.5 *Reactivity Effect of Depletion Related Fuel Assembly Geometry Changes*

During irradiation in light water reactors the fuel assembly undergo small physical changes associated with irradiation and residence time in an operating reactor. Some of those changes are clad thinning due to fuel rod growth and creep, fuel densification, grid growth, and crud buildup on the outside surface of the fuel rod. These fuel assembly geometry changes can affect the neutron spectrum during depletion by changing the fuel to moderator ratio. In the SFP, the effect during depletion may lead to a different isotopic composition and the fuel geometry change can impact reactivity also by the change in the fuel to moderator ratio. These fuel geometry changes are accounted for as follows:

[redacted]

[redacted]

[redacted]

[redacted]

The calculations are performed to determine the reactivity effect of the fuel geometry changes using the 2x2 rack model discussed in Section 3.3. The following studies are performed:

- Case 3.3.6.5.0: Reference case. All fuel parameters are nominal;
- Case 3.3.6.5.1: Fuel rod growth and creep;
- Case 3.3.6.5.2: Grid growth.

Separate depletion calculations are performed for the cases above so that the effect of these changes during depletion is accounted for. The results of the calculations are considered as the bias and bias uncertainty, rather than the uncertainty only, to determine the maximum k_{eff} value. Therefore, the maximum positive (if any) reactivity effect is treated as bias and the 95/95 uncertainty of the bias is statistically combined with the other uncertainties.

3.3.6.6 Spent Fuel Isotopic Content Uncertainty

3.3.6.6.1 Depletion Uncertainty

To ensure that CASMO5 produces reliable and predictable results, an uncertainty on the irradiated fuel isotopic composition (i.e., the number densities) equal to 5% of the reactivity decrement is considered in accordance with [2] and [3]. Specifically, the depletion uncertainty is determined using the following Equation 3.3-2, i.e., by multiplying 5% with the reactivity difference (at 95%/95%) between the MCNP calculation with spent fuel at the minimum burnup requirement (i.e., for each point along the burnup versus enrichment curve) and a corresponding MCNP calculation with fresh fuel at the same fuel enrichment.

$$depl = \overline{\quad {}_1 + \quad {}^2)} \; x\, 0.05 \quad \text{Equation 3.3-2}$$

$$(\quad calc \; - \quad calc\,) + 2\overset{2}{x} \quad (\quad {}^{1}{}_{2} \qquad\qquad {}_{2}$$

where

$calc_1$ - calculated k_{eff} value for spent fuel;
$calc_2$ - calculated k_{eff} value for fresh fuel;
σ_1 - standard deviation of calculated k_{eff} value for spent fuel;
σ_2 - standard deviation of calculated k_{eff} value for fresh fuel.

The following calculations are performed using the 2x2 rack model discussed in Section 3.3:

- Case 3.3.6.6.1.1: Reference case. Spent fuel with an upper bound burnup that covers the expected burnup requirement for a given enrichment is conservatively considered;
- Case 3.3.6.6.1.2: Fresh fuel of the same enrichment without IBA is used instead of the spent fuel composition.

The established depletion uncertainty covers all uncertainties associated with depletion, such as uncertainty in computation of the isotopic inventory by the depletion code, uncertainty in cross-sections (both actinides and fission products), etc.

3.3.6.6.2 Burnup Uncertainty

The uncertainty of the recorded burnup value is typically less than 5%. In accordance with [3], an uncertainty of 5% is conservatively used. The following calculations are performed using the 2x2 rack model discussed in Section 3.3:

- Case 3.3.6.6.2.1: Reference case. Spent fuel with an upper bound burnup that covers the expected burnup requirement for a given enrichment is conservatively considered;
- Case 3.3.6.6.2.2: Spent fuel of the same enrichment, but the fuel burnup is 5% lower than the fuel burnup in the reference case.

The burnup uncertainty is determined using Equation 3.3-1 and included in the analysis. An additional margin for burnup uncertainty is therefore not necessary for the verification that a fuel assembly can be placed in a designated storage location.

3.3.6.6.3 MAFP Validation

Table 3-5 provides a spent fuel isotopic composition credited in the criticality calculations, where all the major actinides are bolded, while the remaining nuclides (except oxygen) are considered as the minor actinides and fission products. Since the adequate critical experiment data is not available for the MAFP nuclides, a bounding bias value which is 1.5% of the worth of the minor actinides and fission products is conservatively applied in accordance with [27] and [28]. This bias is determined using the following Equation 3.3-3, i.e., by multiplying 1.5% with the reactivity difference (at 95%/95%) between an MCNP calculation with the major actinides only and an MCNP calculation with all credited actinides and fission products.

$$MAFP\ {}_{2}= (\quad calc \quad - \quad \overline{\quad {}_{1} + \quad {}^{2}}\ )\ x\,0.015 \qquad \text{Equation 3.3-3}$$

$$calc\ ) + 2\,x\ (\quad {}^{2} \qquad\qquad {}_{2}$$

where

$calc_1$ - calculated k_{eff} value for spent fuel with all actinides and fission products;

$calc_2$ - calculated k_{eff} value for spent fuel with the major actinides only;

σ_1 - standard deviation of $calc_1$;

σ_2 - standard deviation of $calc_2$.

The following calculations are performed using the 2x2 rack model discussed in Section 3.3:

- Case 3.3.6.6.3.1: Reference case. Spent fuel with an upper bound burnup that covers the expected burnup requirement for a given enrichment is conservatively considered;
- Case 3.3.6.6.3.2: Spent fuel of the same enrichment and burnup, but all minor actinides and fission products are removed from the isotopic composition.

Since this term is conservatively applied as a bias, no additional uncertainty needs to be applied. According to [27], an upper value of 1.5% of the worth is applicable for the spent fuel isotopic compositions consisting of all nuclides in the SFP configuration. Particularly, the MAFP bias estimate is applicable to the current analysis because the computer code, cross-section library as well as the fuel assembly and SFR cell designs are similar to those considered in [27] and [28].

3.3.7 Design Basis Calculations

3.3.7.1 Calculation of a Maximum k_{eff} Value

Applying all the considerations from the previous sections, the calculated k_{eff} value (k_{calc}) is determined using the design basis model, as summarized in Figure 3-2. The maximum k_{eff} value is then determined using Equation 3.3-4, i.e., by adding the total correction factor, which includes all the biases and uncertainties, as summarized in Figure 3-3.

$$ {}^{max}_{eff} = \qquad {}_{calc} + \qquad\qquad \text{Equation 3.3-4} $$

where

${}_{calc}$ - calculated k_{eff} value, as described in Figure 3-2;
TCF - total correction factor, as described in Figure 3-3.

The TCF is the addition of all biases and the statistical combination of the uncertainties. Combining the uncertainties statistically is acceptable since they are all independent. For calculations with borated water, the appropriate MCNP5 code bias and bias uncertainty associated with borated water (see Paragraph 3.2.2.1) are applied. The determined maximum k_{eff} values are used to show compliance with the regulatory limit.

3.3.7.2 Determination of the Spent Fuel Loading Curve

As discussed in Chapters 1.0 and 2.0, the approach used in this report takes credit for soluble boron under normal conditions. Under this approach, the limiting condition is the non-borated condition and the multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 1.0 for the pool flooded with unborated water. Conservatively, a target value of 0.995 is used for the maximum k_{eff} when determining the burnup vs. enrichment curves for the loading configurations with spent fuel.

For establishing the loading curves, a two-step process is used. The first step is the generation of the curves as the polynomial functions for each loading region with spent fuel, and the second step is the validation of these curves for the applicable region and loaded content. This approach minimizes the overall number of required calculations, since it only requires a limited number of calculations to generate the curves.

For each loading region that includes spent fuel assemblies (see Chapter 1.0), for a given spent fuel cooling time, and selected enrichment values, two calculations with different burnups (upper and lower bound with a span of 5 GWd/mtU) are performed using the 2x2 rack model discussed in Section 3.3. Note that the case with the upper bound burnup has been already used as the reference case in the calculations for depletion/burnup uncertainty and MAFP validation.

Interpolations of the results are performed to determine the burnup that ensures that the target k_{eff} is not exceeded. The minimum required burnups are then matched by a third-order polynomial fit as a function of enrichment.

3.3.7.3 *Maximum k_{eff} Calculation with Borated Water*

As discussed above, the approach used in this report takes credit for soluble boron under normal conditions. To ensure that the effective neutron multiplication factor (k_{eff}) is less than the regulatory limit with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity, the calculation with the soluble boron content of 500 ppm is performed for Region 2 using the 2x2 rack model discussed in Section 3.3. Calculations for Region 1 are not made since it does not require the soluble boron credit under normal conditions. The determined maximum k_{eff} values are used to show compliance with the regulatory limit. A discussion of the boron dilution accident is provided in Subsection 3.6.8.

3.4 Spent Fuel Rack Interfaces

With a single type of the SFR, two loading regions (see Chapter 1.0) and different loading curves, there are four interface conditions that need to be considered in the Callaway SFP:

- Interfaces between different SFRs;
- Interfaces between storage racks and the SFP wall;
- Interfaces of different regions within a rack;
- Interfaces of the same region within a rack.

3.4.1 Interfaces between different SFRs

[redacted] The number of neutron absorber panels between assemblies across any interface between racks is therefore larger than the one analyzed in the design basis model (a laterally infinite arrangement of storage cells). This increased number of absorber panels, together with the water gap between racks, would reduce the reactivity at the periphery of the racks, and possibly for entire rack. However, for simplification and conservatism, these gaps and the corresponding potential reduction in reactivity are neglected, and rack cells are always assumed to be in a laterally infinite configuration. Hence no additional calculation of any rack to rack interface is required.

3.4.2 Interfaces between Storage Racks and the SFP Wall

[redacted] As for the rack to rack interface discussed in the previous subsection, the neutron leakage, in this case without another face-to-face assembly, would reduce the reactivity. Again, for simplicity and conservatism, this effect is neglected, hence no additional calculation of any rack to wall interface is required.

3.4.3 Region 1 to Region 2 Interface

Three configurations of an interface between Region 1 and Region 2 are evaluated, as follows:

The simplest interface between Region 1 and Region 2 is the straight-line interface between the two regions, where each fresh assembly on the interface line faces a spent fuel assembly across the interface, and each spent fuel assembly on the interface line faces either a fresh assembly or an empty cell across the interface. This configuration is principally shown in Figure 3-5 (b).

This interface is implemented as follows, as shown in the radial cross section of the MCNP model in Figure 3-4:

[redacted]

[redacted]

Since the reactivity of an infinite Region 1 area is very low, this interface is dominated by Region 2, and the results will approach the results for the infinite Region 2 configuration if interfacing of fresh and spent assemblies do not have an overall detrimental effect on reactivity. Calculations will verify that such a detrimental effect does not occur.

The following calculation is performed to qualify this transitional pattern:

- Case 3.4.3.1: Straight interface of Region 1 and Region 2.

However, only qualifying this pattern would be very restrictive since it would only qualify such straight interfaces. This is specifically problematic since the rack-to-rack gaps are conservatively neglected in their effect on reactivity (see previous subsection), hence the straight line would have to continue through the entire pool. That is not a practical approach, options are needed for corners of Region 1 areas facing Region 2 areas, and vice versa. In such situations, a single fresh assembly could now face two spent assemblies, or a single spent fuel assembly could face two fresh assemblies, and that is not covered by the straight-line model.

[redacted]

[redacted]

- Case 3.4.3.2: [redacted]
- Case 3.4.3.3: [redacted]

[redacted]

Additionally, since the reactivity of the Region 1 configuration is fairly low, additional Region 1 to Region 2 interfaces may be acceptable, such as placing individual spent fuel assemblies between the fresh assemblies in the Region 1 pattern. However, in the interest of clarity and simplicity, no such configuration is evaluated or qualified, and this is also to be excluded. Again, see Subsection 3.4.6 for the approach taken for implementing this exclusion.

3.4.4 Region 1 to Region 1 Interface

Since the reactivity of the Region 1 configuration is fairly low, it may be possible to have additional patterns, in addition to the checkerboard, where two fresh assemblies may be placed face-adjacent to each other. An example of such a pattern is shown in Figure 3-5 (a). However, in the interest of clarity and simplicity, no such configuration is evaluated or qualified.

3.4.5 Region 2 to Region 2 Interface

This configuration contains only spent fuel that is adjacent to each other (see Figure 3-5 (e)). These fuel assemblies are stored under the different Region 2 loading curves based on the cooling times. However, these different cooling times are compensated by the appropriate fuel burnup requirements, which ensure that the calculated k_{eff} values remain similar. Further evaluation of any Region 2 to Region 2 interface is therefore not necessary.

3.4.6 Combined Qualifications

Defining the permissible configurations consistent with the discussions and limitations in the subsections above just in terms of Region 1, Region 2, corners, and exclusions would potentially result in ambiguous descriptions, or descriptions that could easily be misinterpreted. Therefore, a different approach is taken, where the permissible content of any rack cell is defined purely on the basis of the content of the face-adjacent cells. The resulting set of definitions is presented in Appendix B, Section B-4.0. Not only does it define and qualify the interfaces consistent with all discussions above, it also implicitly defines the infinite arrangement of Region 2 (uniform spent fuel) and Region 1 (checkerboard of fresh assemblies and empty cells), without the need to use the terms "uniform" or "checkerboard". This avoids the ambiguity of those terms, which are known to have caused problems in the past. In other words, the set of rules in Appendix B, Section B-4.0, is a complete and unambiguous set of rules for the placement of fuel in the spent fuel pool, and no other rules are needed beyond those.

3.5 Normal Conditions

The normal conditions considered in the analyses are all those conditions that can normally occur with fuel assembly in the SFP. The normal locations of fuel in the SFRs according to the analyzed cases and patterns are evaluated in Subsection 3.3.7 and Section 3.4. Other normal conditions are discussed in the following subsections.

3.5.1 Fuel Movement Operations

Fuel movement procedures govern the movement and inspection of the fuel at all times that the fuel is onsite. The fuel assembly is always moved above the SFRs and never moved along the side of the SFR. The fuel assembly placed on a fuel elevator in the Cask Loading Pit or on the fuel transfer system cart for transporting fuel into containment is located at a reasonable distance away from the SFRs to preclude a criticality concern [19]. All fuel movement operations involve a single fuel assembly that is never in close enough proximity (i.e., directly adjacent) to any other fuel assembly. Therefore, a single fresh fuel assembly in water bounds all situations during normal fuel movement in the pool. Based on previous experience, the reactivity of a single fresh PWR fuel assembly in unborated water is below 0.95 and bounded by reactivity of the array of assemblies in the SFR, hence this condition is considered to be bounded by the design basis calculations, and no further evaluations are required.

3.5.2 Fuel Insertion and Removal Operations

Within each loading pattern, the reactivity is maximized by the fact that the axial sections that dominate in reactivity are aligned between neighboring assemblies. For example, for spent fuel assemblies, the dominating area is a low burned upper section of the active region. Since the calculations for the loading regions with face adjacent spent fuel utilize identical assemblies, those dominating regions are perfectly aligned. The same is true for the loading regions with fresh fuel, though the dominating area is in the middle part of the active region with the lowest neutron leakage. When one assembly is being removed from the rack, this alignment is locally disturbed, i.e., the dominating regions are no longer aligned between the assembly that is being removed and the surrounding assemblies, hence the reactivity is reduced.

Also, in case of a partially loaded fuel assembly (e.g., fuel assembly reconstitution), some fissile material in the SFR cell is replaced with water. This increases the neutron moderation and, eventually, increases the effectiveness of the surrounding thermal neutron absorber panels. This way, due to the reduced amount of fissile material and the increased neutron absorption, the reactivity of the SFR during the insertion/removal operation is reduced, and it is always bounded by the fully loaded condition. The exposed end of the partially loaded fuel assembly is surrounded by a large volume of water, hence effectively neutronically decoupled from the SFR contents. Therefore, no further evaluations are required.

3.5.3 Storage of Fuel Rod Storage Rack

The purpose of the FRSR is to collect and store individual fuel rods extracted from other fuel assemblies. Typically, the FRSR contains a limited number of storage cells at a regular square or hexagonal array with the lattice pitch, that is larger than the fuel rod pitch in the PWR assembly. Since the maximum number of fuel rods in the FRSR is well bounded by the number of rods in the fuel assembly, the amount of fissile material in the FRSR and its reactivity is expected to be significantly lower.

To verify this, the reactivity calculations are performed for the FRSR in Table 5-10, using the 2x2 rack model discussed in Section 3.3. [redacted]

[redacted] The MCNP5 model of the SFR with the FRSR is shown in Figure 3-6.

It should be noted that the purpose of this analysis is to demonstrate that the reactivity of the SFR loaded with the FRSRs in a bounding configuration is substantially lower than the bounding reactivity determined for any of the qualified loading regions. Hence the available margin is well sufficient to offset any potential uncertainties related to the FRSR tolerances, and no attempt is made to quantify them. The discussion in Subsection 3.5.2 is also applicable to the FRSR.

3.5.4 Storage of Fuel Assemblies with Missing Rods

Several fuel assemblies have undergone partial reconstitution (i.e., removing part or whole fuel rod without replacing it with a stainless-steel dummy rod). The fuel lattices of such assemblies are provided in Section 5.8. Since a loss of the fuel pin increases the amount of moderator in the undermoderated PWR assembly, a small increase in reactivity is expected. To ensure that the available safety margin is sufficient to offset a potential reactivity effect of the missing rods, the calculation is performed for each fuel lattice with the missing rods using the 2x2 rack model discussed in Section 3.3. Conservatively, [redacted]

During fuel assembly reconstitution activities, a larger number of fuel rods can be potentially removed as well as the different layouts of the missing rods can appear, in comparison with the lattices in Section 5.8. Near the center of the assembly lattice, the removal of a rod increases the moderation level for adjacent rods and results in an increase in reactivity, while closer to the edge of the assembly, the removal of a rod increases moderation near the thermal neutron poison panels and, in many cases, decreases the reactivity. With the increase in the number of removed fuel rods, at some point, a negative effect of the loss of the fissile materials becomes dominant. [redacted] When the fuel reconstitution is performed in Region 1, there is plenty of margin to offset a reactivity increase due to reconstitution with no limit on the number of missing rods. Therefore, fuel assembly reconstitution activities are restricted to the Region 1 configuration, and further evaluations are not required.

3.5.5 Storage of Low-Burned Fuel Assemblies

The Callaway SFP inventory includes fuel assemblies that for various reasons have been discharged from the reactor core before they achieved sufficient burnup. Such assemblies may need to be loaded into Region 1 storage configurations. However, the fuel assemblies initially loaded in Cycle 1 (i.e., Westinghouse STD) are susceptible to Top Nozzle Separation failure, thus their relocation within the pool is currently prohibited. For all those assemblies, the minimum burnup requirement has been determined based on the assembly-specific enrichment and cooling time using the loading curves generated in Paragraph 3.3.7.2, and compared with the actual burnups of those assemblies. It was identified that only one fuel assembly that does not meet the burnup requirement is of Westinghouse STD design, but this fuel assembly is already stored in the checkerboard configuration (Region 1). In summary, it is therefore concluded that the specification of Region 1 and Region 2 allows the storage of all assemblies without the need for any additional assembly-specific evaluations for low burnup.

3.6 Abnormal and Accident Conditions

The following accident conditions are considered in the following subsections for the Callaway SFP:

- Loss of SFP cooling;
- Dropped assembly resting horizontally on the SFR;
- Assembly dropped vertically into a storage cell;
- Mislocated fuel assembly (a fuel assembly in the wrong location outside the SFR);
- Misloaded fuel assembly (a fuel assembly in the wrong location within the SFR);
- Incorrect loading curve (multiple misload);
- Rack movement;
- Boron dilution.

Note that by virtue of the double contingency principle [3], two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

The maximum k_{eff} value calculations performed for the accident conditions include the same total correction factors as those used for the design basis models for the normal conditions, as discussed in Paragraphs 3.3.7.1 and 3.3.7.3. Previous analyses showed that the tolerance and uncertainty calculations performed separately for accident conditions produce similar results, hence the normal condition results are considered applicable. The calculations are performed using borated water with several soluble boron concentrations, and the soluble boron requirement, that results in a reactivity equal or lower than the regulatory limit of 0.95, is interpolated between these calculations.

3.6.1 Loss of SFP Cooling

Under the accident conditions (loss of cooling), the SFP water temperature could be elevated beyond the normal operating range. The reactivity effect of the SFP temperature over a range of credible values is evaluated in Subsection 3.3.4. All design basis calculations consider the bounding SFP water temperature and density. Therefore, no further calculations are necessary.

3.6.2 Dropped Assembly – Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of the SFR, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more than 12 inches, which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in an increase in reactivity.

3.6.3 Dropped Assembly – Vertical into a Storage Cell

It is also possible to vertically drop an assembly into a location that might be occupied by another assembly. Such a vertical impact would at most cause a small compression of the stored assembly, if present, or result in a small deformation of the baseplate for an empty cell. These deformations could potentially increase reactivity. However, the reactivity increase would be small compared to the reactivity increase created by the misloading of a fresh assembly discussed in Subsection 3.6.5. The vertical drop is therefore bounded by the misloading accident and no separate analysis is performed.

3.6.4 Mislocated Fuel Assembly

The fuel assembly mislocation accident could possibly occur if a fresh fuel assembly of the highest permissible enrichment (5.0 wt% ^{235}U) is accidentally mislocated or dropped outside of a storage rack adjacent to the other fuel assemblies. Considering that [redacted] and taking into account a high neutron leakage at the edges of the storage rack, the accidental mislocation of a fresh fuel assembly outside the rack is bounded by the fresh assembly misload (see Subsection 3.6.5). Therefore, no further calculations are necessary.

3.6.5 Misloaded Fuel Assembly

As discussed in Chapter 1.0, the SFP racks are qualified for various configurations of fresh fuel assemblies, spent fuel assemblies, and empty storage cells. It is possible that a fuel assembly that is not qualified for a given loading region could accidently be placed in that region. For instance, the fresh fuel assembly may be inadvertently placed into the storage cell intended for spent fuel in the Region 2 configuration. As a bounding approach, the misload of a single fresh fuel assembly of the highest permissible enrichment (5.0 wt% ^{235}U) is considered in a storage cell that provides the largest positive reactivity increase. The following cases are evaluated:

- Case 3.6.5.1: Misloading into an empty storage cell in Region 1;
- Case 3.6.5.2: Misloading into one of the storage cells intended to store a spent fuel assembly in Region 2.

The misload accident evaluations are performed using the model discussed in Section 3.3, but an 8x8 array of storage cells with the periodic boundary conditions is considered, which effectively represents a multiple misload (also see Subsection 3.6.6). The minimum soluble boron concentration is determined for each case that ensures that a maximum k_{eff} value does not exceed the regulatory limit.

Additionally, the misload accident evaluations are performed for Case 3.6.5.1 using the interface models discussed in Section 3.4.

3.6.6 Incorrect Loading Curve

While several independent fuel assemblies misloads are precluded by the double contingency principle, a multiple misload could occur as a result of an incorrect application of the loading curves. As a bounding approach, spent fuel assemblies with the lowest burnup requirement are assumed to be accidentally loaded into all storage cells qualified for spent fuel in the loading regions with the highest burnup requirement. The following cases are evaluated:

- Case 3.6.6.1: Multiple misload of spent fuel assemblies, which were intended to be loaded into Region 2 with 20 years of cooling time, is considered for Region 2 with 0 hours of cooling time.

The incorrect loading curve accident evaluations are performed using the model discussed in Section 3.3. The minimum soluble boron concentration is determined for each case that ensures that a maximum k_{eff} value does not exceed the regulatory limit.

3.6.7 Rack Movement

In the event of seismic activity, there is a hypothetical possibility that the storage rack arrays may slide and come closer to each other. In the worst-case scenario, two racks may touch each other at the baseplate, reducing the physical separation of the fuel assemblies along the rack interface, but still maintaining a minimum water gap width. The worst-case racks movement scenario for the entire SFP is when the water gap width between all SFRs is as low as allowed by the baseplate. This accident condition is bounded by the evaluations of the design basis cases discussed in Section 3.3 since the design basis models consider all the racks at their closest approach, i.e., a laterally infinite arrangement of 2x2 arrays so that the gap between the racks is neglected. Therefore, no further evaluations are necessary.

3.6.8 Boron Dilution

None of the previously discussed accidents would cause a simultaneous boron dilution event. The only hypothetical scenario is a seismic-related pipe break and rack movement. However, as discussed in Subsection 3.6.7, rack movement is bounded by the design basis model, hence there is no safety concern. Therefore, in accordance with [3], as long as the accident does not also result in a dilution of soluble boron, the analysis of a simultaneous boron dilution event is not required, per the double-contingency principle.

Significant loss or dilution of the soluble boron concentration in the SFP is extremely unlikely, however, the guidance presented in [3] requires that the boron dilution analysis should determine that sufficient time is available to detect and suppress the worst dilution event that can occur to reduce the boron concentration to the level needed to maintain k_{eff} less than or equal to 0.95. [redacted]

[redacted] Equation 3.6-1

[redacted]

[redacted] Equation 3.6-2

[redacted]

3.7 Margin Evaluation

3.7.1 Neutron Absorber Aging Effects

The SFR design in the Callaway SFP contains the BORAL™ poison panels constricted in-between the steel box and steel sheathing. Industrywide, there have been no indications of a loss of BORAL™ material of a nature that diminished neutron-absorbing capability [31]. However, Callaway is subject to a License Renewal commitment to perform *in situ* areal density measurements – a test method that has historically underestimated panel performance, occasionally to the point of test failure.

For the purpose of operational support of the potential of lower BORAL™ poison areal density, an evaluation of the potential reactivity effect of such lower areal density, and an evaluation of available margin in the criticality analysis to offset such effect is performed and documented in the following subsections. If an unanticipated BORAL™ poison areal density is identified, this information can be utilized to demonstrate operability, and to determine what technical, operational and licensing actions may need to be taken.

3.7.2 BORAL™ Panel ^{10}B Areal Density

The evaluation of the reactivity effect in terms of Δk per three changes in BORAL™ panel areal density: 5%, 10% and 20% reductions from the minimum allowed, is performed. For all cases, the reduction is assumed to be uniform throughout the pool. The areal densities are then matched by a second-order polynomial fit as a function of Δk such that the increase can be compared to any compensatory actions or assumptions. Note that the design basis analysis is already performed with the minimum BORAL™ panel areal density (treated as a bias), while the nominal areal density is expected to prevail in the SFR BORAL™ panels, which provides additional margin.

3.7.3 Criticality Analysis Safety Margin

The following calculations are performed to estimate available margins in the criticality analysis:

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

- Case 3.7.9: [redacted]

Note that no attempt is made here to quantify a negative reactivity associated with presence of the residual IBA in the spent fuel isotopic composition. Also, as discussed in Paragraph 3.3.7.3, only 500 ppm of soluble boron is credited for the normal conditions, which is significantly lower than a typical soluble boron content in the SFP (Table 5-8).

3.8 Permitted Future Fuel Assemblies

The criticality analysis documented in this report qualifies all currently stored and/or known future fuel assembly designs. As discussed in Subsection 3.3.1, the bounding fuel design is determined and used in all design basis calculations. In the future, new assembly designs may need to be qualified that have not been explicitly addressed by the criticality analysis in this report. In general, the qualification of these new assembly designs is governed by the criteria in the Callaway technical specification (summarized in Appendix B). While a change of the fuel assembly array configuration and/or instrument/guide tube patterns is not expected, since it's not typical for PWR, any such change as well as a change of the geometric dimensions and material compositions, which are important to criticality but not bounded by the design basis model, would require an additional evaluation.

Table 3-1 – Summary of Area of Applicability of the MCNP5 Benchmark

Parameter	Design Application	Benchmarks [13]
Fissionable Material	[redacted]	[redacted]
Isotopic Composition		
- ^{235}U/U, wt%	[redacted]	[redacted]
- Pu/(U+Pu), wt%	[redacted]	[redacted]
- Physical Form	[redacted]	[redacted]
- Fuel Density, g/cm^3	[redacted]	[redacted]
Moderator Material (Coolant)	[redacted]	[redacted]
- Physical Form	[redacted]	[redacted]
- Density, g/cm^3	[redacted]	[redacted]
Reflector Material	[redacted]	[redacted]
- Physical Form	[redacted]	[redacted]
- Density, g/cm^3	[redacted]	[redacted]
Reflector	[redacted]	[redacted]
Absorber and Separating Material		
- Soluble	[redacted]	[redacted]
- Rod	[redacted]	[redacted]
- Plate	[redacted]	[redacted]
Geometry		
- Lattice type	[redacted]	[redacted]
- Lattice Pitch, cm	[redacted]	[redacted]
Temperature, K	[redacted]	[redacted]
Neutron Energy	[redacted]	[redacted]
EALF, eV	[redacted]	[redacted]

---

[1] [redacted]

Table 3-2 – MCNP5 Benchmark Analysis for Various Fuel and Water Subsets of Experiments

Experiment Description	No. of exp.	Bias[1]	Bias Uncertainty[2]	Normality χ^2 ($P_d(\chi^2;d)$)	Linear Correlation	Residuals Normality, ($P_d(\chi^2;d)$)
All experiments	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
All except those with Gadolinium, Cadmium and Lead[3]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fresh UO2 Fuel with Fresh Water	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fresh UO2 Fuel with Borated Water	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
HTC + MOX Fuel with Fresh Water	[redacted]	[redacted]	[redacted]	[redacted]	[redacted] [redacted]	[redacted] [redacted]
HTC + MOX Fuel with Borated Water	[redacted]	[redacted]	[redacted]	[redacted]	[redacted] [redacted]	[redacted] [redacted]

---

[1] [redacted]

Table 3-3 – Significant Trending Analysis for Callaway Parameters

Experiment Description	Linear Correlation	Analysis Parameter Value[1]	Analysis Parameter Trend Bias
Fresh UO2 Fuel with Fresh Water	[redacted]	[redacted]	[redacted]
HTC + MOX Fuel with Fresh Water	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]
HTC + MOX Fuel with Borated Water	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]

---

[1] The maximum or minimum parameter value that provides the most negative bias is used. The positive bias values are conservatively neglected.

Table 3-4 – Summary of MCNP5 Code Validation Bias and Bias Uncertainty

Description	Applicable Loading Regions	Pure Water		Borated Water	
		Bias[1]	Bias Uncertainty	Bias[1]	Bias Uncertainty
Fresh Fuel	1	[redacted]	[redacted]	[redacted]	[redacted]
Spent Fuel	2	[redacted]	[redacted]	[redacted]	[redacted]

---

[1] The values in parentheses are based on trending analyses in Table 3-3.

Table 3-5 – List of Spent Fuel Isotopes[1]

MCNP ZAID [8]					
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	

---

[1] Deleted

[2] [redacted]

[3] The isotopes considered as the major actinides are bolded [28].

Table 3-6 – Bounding Axial Burnup Profiles for Westinghouse 17x17 Fuel Type [15]

Axial Segment (18 = Top)	Assembly Average Burnup (GWd/mtU)				
	7.5	17.5	27.5	37.5	≥ 45
	Relative Burnup per Segment[1]				
1	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
3	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
4	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
5	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
6	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
7	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
8	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
9	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
10	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
11	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
12	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
13	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
14	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
15	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
16	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
17	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
18	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

---

[1] Segment burnup divided by assembly average burnup.

Table 3-7 – Bounding Axial Burnup Profiles from NUREG/CR-6801 [26]

#[1]	Burnup Ranges (GWd/mtU)											
	> 46	42-46	38-42	34-38	30-34	26-30	22-26	18-22	14-18	10-14	6-10	< 6
1	0.582	0.666	0.660	0.648	0.652	0.619	0.630	0.668	0.649	0.633	0.658	0.631
2	0.920	0.944	0.936	0.955	0.967	0.924	0.936	1.034	1.044	0.989	1.007	1.007
3	1.065	1.048	1.045	1.070	1.074	1.056	1.066	1.150	1.208	1.019	1.091	1.135
4	1.105	1.081	1.080	1.104	1.103	1.097	1.103	1.094	1.215	0.857	1.070	1.133
5	1.113	1.089	1.091	1.112	1.108	1.103	1.108	1.053	1.214	0.776	1.022	1.098
6	1.110	1.090	1.093	1.112	1.106	1.101	1.109	1.048	1.208	0.754	0.989	1.069
7	1.105	1.086	1.092	1.108	1.102	1.103	1.112	1.064	1.197	0.785	0.978	1.053
8	1.100	1.085	1.090	1.105	1.097	1.112	1.119	1.095	1.189	1.013	0.989	1.047
9	1.095	1.084	1.089	1.102	1.094	1.125	1.126	1.121	1.188	1.185	1.031	1.050
10	1.091	1.084	1.088	1.099	1.094	1.136	1.132	1.135	1.192	1.253	1.082	1.060
11	1.088	1.085	1.088	1.097	1.095	1.143	1.135	1.140	1.195	1.278	1.110	1.070
12	1.084	1.086	1.086	1.095	1.096	1.143	1.135	1.138	1.190	1.283	1.121	1.077
13	1.080	1.086	1.084	1.091	1.095	1.136	1.129	1.130	1.156	1.276	1.124	1.079
14	1.072	1.083	1.077	1.081	1.086	1.115	1.109	1.106	1.022	1.251	1.120	1.073
15	1.050	1.069	1.057	1.056	1.059	1.047	1.041	1.049	0.756	1.193	1.101	1.052
16	0.992	1.010	0.996	0.974	0.971	0.882	0.871	0.933	0.614	1.075	1.045	0.996
17	0.833	0.811	0.823	0.743	0.738	0.701	0.689	0.669	0.481	0.863	0.894	0.845
18	0.515	0.512	0.525	0.447	0.462	0.456	0.448	0.373	0.284	0.515	0.569	0.525

---

[1] Axial Segment (18 = Top)

Withheld from public disclosure under 10 CFR 2.390

Figure 3-1 – Radial Cross-Section View of the MCNP5 Design Basis Model of the SFR

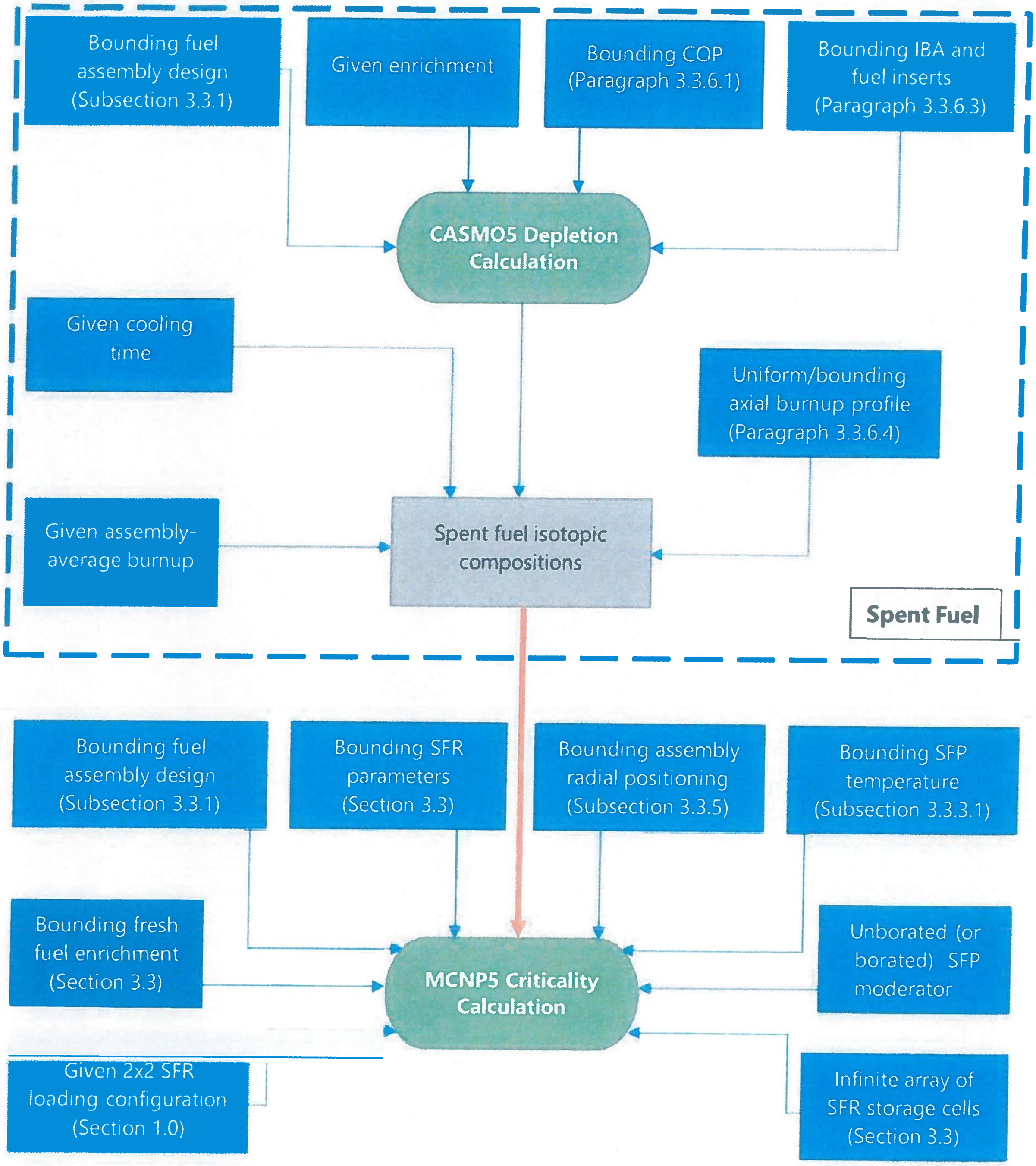

Figure 3-2 – Design Basis Calculation of a k_{eff} Value

Withheld from public disclosure under 10 CFR 2.390

Figure 3-3 – Determination of the Total Correction Factor

Withheld from public disclosure under 10 CFR 2.390

Figure 3-4 – Radial Cross-Section View of the MCNP5 Model for the SFR Interfaces

Withheld from public disclosure under 10 CFR 2.390

Figure 3-5 – Potential Interfaces between the Loading Regions

Withheld from public disclosure under 10 CFR 2.390

Figure 3-6 – Radial Cross-Section View of the MCNP5 Model for the FRSR

Withheld from public disclosure under 10 CFR 2.390

Figure 3-7 – MCNP5 Model of the Heterogeneous BORAL™ Panel

4.0 ASSUMPTIONS

A number of assumptions, either for conservatism or to simplify the calculation approach are applied in the analyses. Each assumption is appropriately discussed and justified in the text. Bounding or sufficiently conservative inputs and assumptions are used essentially throughout the entire analyses, and as necessary, studies are presented to show that the selected inputs and parameters are in fact conservative or bounding. For additional details, a reader is referred to Section 3.3.

While the fuel assembly and SFR models used in the analyses are very detailed (see Section 3.3), to assure that the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and simplifications are made:

#	Assumption, Approach	C, or R (Note 1)	Justification
1	[redacted]	[redacted]	[redacted]
2	[redacted]	[redacted]	[redacted]
3	[redacted]	[redacted]	[redacted]
4	[redacted]	[redacted]	[redacted]
5	[redacted]	[redacted]	[redacted]

#	Assumption, Approach	C, or R (Note 1)	Justification
6	[redacted]	[redacted]	[redacted]
7	[redacted]	[redacted]	[redacted]
8	[redacted]	[redacted]	[redacted]
9	[redacted]	[redacted]	[redacted]
10	[redacted]	[redacted]	[redacted]
11	[redacted]	[redacted]	[redacted]

Note 1: C stands for Conservative, and R stands for Reasonable.

5.0 INPUT DATA

5.1 Fuel Assembly Designs

The Callaway SFP is designed to accommodate various PWR fuel assembly designs, such as Westinghouse 17x17 Standard (STD), Westinghouse 17x17 Optimized (OFA), Westinghouse 17x17 Vantage 5 (V5), Westinghouse 17x17 Vantage+ (V+), and Framatome GAIA 17x17 (GAI).

[redacted]

[redacted] The PWR fuel assembly data used in the analysis is presented in Table 5-1 through Table 5-3.

5.2 Core Operating Parameters

As discussed in Paragraph 3.3.6.1, the upper bound core operating parameters (conservative) for all Callaway Unit 1 cycles are used for fuel depletion calculations performed with CASMO5. Core operating parameters are presented in Table 5-4.

5.3 Integral Burnable Absorber and Fuel Inserts

During the Callaway reactor operation, some fuel assemblies have made use of fuel inserts, namely Pyrex, WABA and RCCA. Additionally, assemblies can contain integral burnable absorbers, consisting of neutron absorbing material mixed into the fuel pellet (Gadolinia) or added as a coating on the fuel pellet (ZrB_2).

The burnable absorber rods contain a certain amount of ^{10}B, in the form of Al_2O_3-B_4C or SiO_2-B_2O_3 in annular pellets inside a Zircaloy or SS cladding. The control rods consist of highly neutron absorbing material inside the SS cladding. Specifically, the reactor operation [redacted] was controlled using the RCCAs made of Hafnium-Zirconium (Hf-Zr), while [redacted] utilize the RCCAs made of Silver-Indium-Cadmium (Ag-In-Cd). The design specifications for the Pyrex, WABA, and RCCA devices are provided in Table 5-5.

IBAs are integral to the fuel rods, and therefore do not replace water in the guide tubes. Consequently, the spectrum hardening effect of the IBAs during irradiation, and therefore the reactivity effect, is significantly lower compared to the fuel inserts. The design specifications for the Gd and IFBA rods are shown in Table 5-6.

[redacted]

5.4 Spent Fuel Rack Design

The SFP contains a single type of BORAL™ SFR designed for storage of PWR fuel assemblies. The SFR storage cells are composed of SS boxes with a single fixed neutron absorber panel centered on each side, attached by SS sheathing. The SS boxes are arranged in an alternating pattern and connected in a rigid structure such that the connection of the box corners form storage cells between them. Neutron absorber panels are also installed on all exterior walls of the SFR.

Figure 5-2 and Figure 5-3 provide a sketch of the storage rack in the radial and axial direction respectively. The SFR design data used in the analysis is presented in Table 5-7.

5.5 SFP Operating Parameters

The SFP operating parameters are presented in Table 5-8.

5.6 Material Compositions

The material compositions for the principal design components of the fuel assemblies and SFRs are listed in Table 5-9. The MCNP nuclide identification number (ZAID), presented for each nuclide in Table 5-9, includes the atomic number and mass number, which are consistent with the ZAIDs used in the benchmarking calculations documented in [13]. The appropriate temperature-specific cross-section library identifier (i.e., *ZAID.identifier*) is used with all ZAIDs in the MCNP model (see Subsection 3.3.4).

5.7 Fuel Rod Storage Rack

The FRSR parameters are presented in Table 5-10.

5.8 Fuel Assemblies with Missing Rods

The fuel lattices for the fuel assemblies with the missing rods are taken from [19] and [1], and presented in Figure 5-4.

Table 5-1 – Specification of the Fuel Assembly Parameters [20]

Fuel Assembly Design	**Westinghouse STD**	**Westinghouse OFA**	**Westinghouse V+**[1]	**Framatome GAIA** [21]
Fuel Assembly Data				
Fuel Rod Array	17x17			
Number of fuel rods	264			
Distance from Bottom of Fuel Assembly to Beginning of Active Length, inches	[redacted]	[redacted]	[redacted]	[redacted]
Active Fuel Length, inches	[redacted]			
Fuel Rod Pitch, inches	[redacted]			
Axial Blanket Length, inches	[redacted]	[redacted]	[redacted]	[redacted]
Fuel Rod Data				
Clad OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
Clad ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Clad Material	[redacted]	[redacted]	[redacted]	[redacted]
Pellet ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Pellet OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
As-Built UO_2 Density (Max %TD)	[redacted]	[redacted]	[redacted]	[redacted]
Guide/Instrument Tube Data				
Number of Guide Tube	[redacted]			
Guide Tube OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
Guide Tube ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Guide Tube Material	[redacted]	[redacted]	[redacted]	[redacted]
Number of Instrument Tube	[redacted]			
Instrument Tube OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
Instrument Tube ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Instrument Tube Material	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

Table 5-2 – PWR 17x17 Fuel Assembly Manufacturing Tolerances

Parameter	Value[1]	Reference
Fuel Rod Pitch, inches	[redacted]	[20]
Pellet OD, inches	[redacted]	[20], [21]
Clad OD, inches	[redacted]	[21]
Fuel Enrichment, wt% ^{235}U	[redacted]	[20], [21]
Fuel Density, %TD	[redacted]	[20], [21]
ZrB_2 Coating Loading, %	[redacted]	[20]

Table 5-3 – PWR 17x17 Fuel Assembly Depletion Related Geometry Changes

Parameter	Value	Reference
Maximum Fuel Rod Growth, inches	[redacted]	[21]
Maximum Fuel Grid Growth, inches	[redacted]	[20]

---

[1] Bounding values for all fuel assembly types are summarized.

Table 5-4 – Core Operating Parameters

Parameter	Value	Reference
Maximum Core Moderator Temperature, K	[redacted]	[21]
Maximum Fuel Temperature, K	[redacted]	[20]
Reactor Specific Power, W/gU	[redacted]	[20]
Soluble Boron Concentration (cycle average)[1], ppm	[redacted]	[19]
In-Core Assembly Pitch, inches	[redacted]	[19]

[redacted]

Table 5-5 – Specification of the Fuel Inserts [20]

Parameter	Pyrex	WABA	Hf-Zr (RCCA)	Ag-In-Cd (RCCA)
Maximum Number of Rods per Assembly[1]	[redacted]	[redacted]	[redacted]	[redacted]
Maximum Insertion Depth[2], inches	[redacted]	[redacted]	[redacted]	[redacted]
Burnable Absorber Material	[redacted]	[redacted]	[redacted]	[redacted]
Absorber Content, wt%	[redacted]	[redacted]	[redacted]	[redacted]
Burnable Absorber Density, g/cc	[redacted]	[redacted]	[redacted]	[redacted]
Burnable Absorber Composition, wt%				
Si	[redacted]	[redacted]		
O	[redacted]	[redacted]		
^{10}B	[redacted]	[redacted]	[redacted]	[redacted]
^{11}B	[redacted]	[redacted]		
Al	[redacted]	[redacted]		
C	[redacted]	[redacted]		
Inner Clad Material	[redacted]	[redacted]	[redacted]	[redacted]
Inner Clad ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Inner Clad OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
Burnable Absorber ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Burnable Absorber OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
Outer Clad Material	[redacted]	[redacted]	[redacted]	[redacted]
Outer Clad ID, inches	[redacted]	[redacted]	[redacted]	[redacted]
Outer Clad OD, inches	[redacted]	[redacted]	[redacted]	[redacted]
Burnable Absorber Length, inches	[redacted]	[redacted]	[redacted]	[redacted]

---

[1] See Figure 5-1 for the fuel inserts layouts.

[2] See Subparagraph 3.3.6.3.2.

[redacted]

Table 5-6 – Specification of the Integral Burnable Absorbers

Parameter	Value
IFBA Rods [20]	
ZrB_2 Coating Loading, mg ^{10}B/inch	[redacted]
^{10}B Enrichment of ZrB_2, wt%	[redacted]
Number of the IFBA rods	[redacted]
IFBA Stack Length, inches	[redacted]
IFBA Rods Layout	See Figure 5-1
Burnable Absorber Composition	See Table 5-9
Gd Rods [21]	
Gadolinia Loading, wt% Gd_2O_3	[redacted]
Number of the Gd rods	[redacted]
Gd Stack Length, inches	[redacted]
Gd Rods Layout	See Figure 5-1
Burnable Absorber Composition	See Table 5-9

Table 5-7 – Specification of the Callaway SFR Parameters

Parameter	Value [1], [29], [32], [33]
Rack Type	[redacted]
Number of Racks	[redacted]
Storage Rack Material	[redacted]
Rack Height (Top of Baseplate to Top of Rack), inches	[redacted]
Distance from Rack Baseplate to Bottom of Neutron Absorber, inches	[redacted]
Storage Cell ID, inches	[redacted]
Storage Cell Pitch, inches	[redacted]
Storage Cell Box Wall Thickness, inches	[redacted]
Inner Sheathing Thickness, inches	[redacted]
Peripheral Sheathing Thickness, inches	[redacted]
Neutron Absorber Panel	[redacted]
Type	[redacted]
Thickness, inches	[redacted]
Width, inches	[redacted]
Length, inches	[redacted]
^{10}B Areal Density (g/cm^2)	[redacted]

---

[1] [redacted]

Table 5-8 – SFP Operating Parameters [19], [29], [34], [35]

Parameter	Value
Maximum Moderator Temperature, °F	[redacted]
Soluble Boron Concentration, ppm	[redacted]
Minimum Water Level (Volume) in the SFP, inches	[redacted]
Water Volume above the SFRs[2], gal	[redacted]
Water Volume in the RMWST (0 ppm), gal	[redacted]
Soluble Boron Concentration in the RWST, ppm	[redacted]
Total CCW System Volume, gal	[redacted]
Maximum Blow-Down Rate, gpm	[redacted]
Probable Maximum Precipitation[3], gpm	[redacted]
Maximum Operator Response Time for Internal Flooding Events, min	[redacted]
SFP Boron Concentration Surveillance Interval, days	[redacted]
Annunciator Setpoint for Low SFP Level (Volume), inches	[redacted]
Annunciator Setpoint for High SFP Level (Volume), inches	[redacted]
SFP Water Overflow Level, inches	[redacted]

[redacted]

Table 5-9 – Material Compositions of the Major Design Components

Element	MCNP ZAID [8]	Weight Fraction
Stainless Steel (Density – 7.84 g/cm^3)		
Cr	24050	0.0079050
	24052	0.1585266
	24053	0.0183218
	24054	0.0046467
Mn	25055	0.0200100
Fe	26054	0.0389826
	26056	0.6345800
	26057	0.0149174
	26058	0.0020200
Ni	28058	0.0671977
	28060	0.0267760
	28061	0.0011834
	28062	0.0038348
	28064	0.0010082
Zirconium (Density – 6.55 g/cm^3)		
Zr	40090	0.5070612
	40091	0.1118009
	40092	0.1727810
	40094	0.1789110
	40096	0.0294379
BORAL™ ([redacted]		
B	5010	[redacted]
	5011	[redacted]
C	6000	[redacted]
Al	13027	[redacted]

---

[1] BORAL™ material calculated density based on the nominal panel thickness and minimum ^{10}B loading.

Table 5-9 – Material Compositions of the Major Design Components

Element	MCNP ZAID [8]	Weight Fraction
Pure Water (Density – 1.0 g/cm³)		
H	1001	0.1118854
	1002	0.0000257
O	8016	0.8857957
	8017	0.0022932
Borated Water (500 ppm, Density – 1.0 g/cm³)		
H	1001	0.1118300
	1002	0.0000257
O	8016	0.8853540
	8017	0.0022921
B	5010	0.0000922
	5011	0.0004078
Borated Water (1000 ppm, Density – 1.0 g/cm³)		
H	1001	0.1117740
	1002	0.0000257
O	8016	0.8849110
	8017	0.0022909
B	5010	0.0001843
	5011	0.0008157
[redacted]		
B	5010	[redacted]
	5011	[redacted]
Zr	40090	[redacted]
	40091	[redacted]
	40092	[redacted]
	40094	[redacted]
	40096	[redacted]

Table 5-9 – Material Compositions of the Major Design Components

Element	MCNP ZAID [8]	Weight Fraction
Fresh UO_2[1] (5.0 wt% ^{235}U, Density – 10.6312 g/cm^3)		
U	92235	0.0440800
	92238	0.8374200
O	8016	0.1185000
Fresh UO_2-Gd_2O_3[1] (5.0 wt% ^{235}U, [redacted]		
U	92235	[redacted]
	92238	[redacted]
O	8016	[redacted]
Gd	64152	[redacted]
	64154	[redacted]
	64155	[redacted]
	64156	[redacted]
	64157	[redacted]
	64158	[redacted]
	64160	[redacted]

---

[1] The design basis case is provided as an example; other fresh fuel compositions may be used.

Table 5-10 – Fuel Rod Storage Rack Parameters

Parameter	**Value** [19]
Fuel Rod Array	[redacted]
Number of Storage Tubes	[redacted]
Normal Storage Tube OD, inches	[redacted]
Normal Storage Tube Thickness, inches	[redacted]
Enlarged Storage Tube OD, inches	[redacted]
Enlarged Storage Tube Thickness, inches	[redacted]
Storage Tube Pitch, inches	[redacted]
Storage Tube Material	[redacted]

Withheld from public disclosure under 10 CFR 2.390

Figure 5-1 – Considered PWR 17x17 Fuel Assembly Layouts

Withheld from public disclosure under 10 CFR 2.390

Figure 5-2 – Planar Cross-Section of the Callaway SFR

Withheld from public disclosure under 10 CFR 2.390

Figure 5-3 – Axial Cross-Section of the Callaway SFR

Withheld from public disclosure under 10 CFR 2.390

Figure 5-4 – Fuel Assembly Layouts with Missing Rods

6.0 COMPUTER PROGRAMS

Holtec International maintains an active list of QA validated computer codes on the Company's network, hereinafter referred to as the ACPL, that are approved for use in safety significant projects. The table below identifies the Code and its version (listed in the ACPL) that has been used in this work effort. For additional details, a reader is referred to Section 3.2.

Generic Report & ACPL Information			
Generic Report #	HI-2104750, HI-2115064		
Code name (listed in the ACPL)	CASMO5	MCNP5	Python SX
Code version # (approved in the ACPL)	2.08.00	1.51	1.0
Code name and versions used in previous revisions of the report (if different than listed above)	N/A		

All calculations were performed on computers under Windows at Holtec's offices.

7.0 CALCULATIONS AND RESULTS

As discussed in Section 3.3 of the main report, the analysis is performed using a combination of bounding analysis parameters and statistical uncertainties. The analysis results and discussions are provided in each section below.

7.1 Design Basis Fuel Assembly Design

As discussed in Subsection 3.3.1, all representative PWR fuel designs are evaluated in the Callaway SFR. In accordance with [3], the evaluations for all storage configurations listed in Chapter 1.0 are performed. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the calculations are presented in Table 7-1. Westinghouse V+ fuel assembly shows the highest reactivity at zero burnup; therefore, it is used in all design basis criticality calculations for Region 1. Framatome GAI fuel assembly shows the highest reactivity at spent fuel configurations; therefore, it is used in all design basis criticality calculations for Region 2.

7.2 Reactivity Effect of Fuel Assembly Parameters

As discussed in Subsection 3.3.2, evaluations are performed to determine the reactivity effect of the fuel assembly manufacturing tolerances. The bounding fuel designs established in Section 7.1 are considered in all storage configurations listed in Chapter 1.0. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the evaluations are presented in Table 7-2. The maximum reactivity effect of the fuel assembly parameters is treated as an analysis uncertainty.

7.3 Reactivity Effect of SFR Parameters

As discussed in Subsection 3.3.3, the SFR parameters are evaluated to determine the reactivity effect of the storage rack manufacturing tolerances. Calculations are performed for all storage configurations listed in Chapter 1.0. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the evaluations are presented in Table 7-3. The maximum reactivity effect of the storage rack parameters is treated as an analysis uncertainty.

7.3.1 Reactivity Effect of the B_4C Particle Size

The calculations are performed for all storage configurations listed in Chapter 1.0 to estimate a reactivity effect of the heterogeneous BORAL™ panel model with the variable B_4C particle size, as discussed in Paragraph 3.3.3.1. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results presented in Table 7-26 demonstrate that [redacted]

7.4 Reactivity Effect of SFP Water Temperature

As discussed in Subsection 3.3.4, the reactivity effect of SFP water temperature and density is evaluated. Calculations are performed for all storage configurations listed in Chapter 1.0. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the evaluations are presented in Table 7-4 and show that the bounding temperature (and corresponding density) are the minimum temperature and maximum density. Therefore, these values are used in all design basis calculations.

7.5 Reactivity Effect of Fuel Assembly Radial Positioning

As discussed in Subsection 3.3.5, the reactivity effect of the fuel radial location is evaluated. Calculations are performed for all storage configurations listed in Chapter 1.0. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the evaluations are presented in Table 7-5, and show that the reference case, i.e., all assemblies centered in their fuel storage cell of a 2x2 array, is bounding. Therefore, the bounding radial position of the fuel assemblies is included in the design basis calculations, and incorporation of the bias and bias uncertainty for the reactivity effect of fuel radial positioning into the TCF is not necessary.

7.6 Spent Fuel Reactivity Calculation

7.6.1 Reactivity Effect of Core Operating Parameters

As discussed in Paragraph 3.3.6.1, a sensitivity study is performed on the effect of the core operating parameters on the fuel composition in the uniform loading of spent fuel assemblies (Region 2). Fuel enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the calculations are listed in Table 7-6 and confirm that higher moderator and fuel temperature and higher soluble boron concentration result in higher reactivity, while the power density has a small effect. Therefore, conservative high values are used for all parameters in all design basis calculations.

7.6.2 Reactivity Effect of Cooling Time

As discussed in Paragraph 3.3.6.2, a sensitivity study is performed on the effect of cooling time on the fuel composition in the uniform loading of spent fuel assemblies (Region 2). Fuel enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve (0 hours) are considered. The results of these calculations are presented in Table 7-7 and confirm that the reactivity decreases with the cooling time.

7.6.3 Reactivity Effect of IBA and Fuel Inserts

As discussed in Paragraph 3.3.6.3, a sensitivity study is performed on the effect of the IBA and fuel inserts on the fuel composition in the uniform loading of spent fuel assemblies (Region 2). Fuel enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the calculations are presented in Table 7-8.

Based on the results [redacted] Hence more abundant and representative WABA inserts are justified for the design basis depletion calculations.

Due to a common RCCA operation with a low-depth insertion of the control rods during the full power operation, the reactivity effect of the Ag-in-Cd rods though positive at higher burnups is well bounded by the effect of WABAs. Also, considering a shorter height of the IFBA stack in comparison with the active fuel height (due to cutback regions) and low-depth insertion of the Ag-In-Cd rods, an impact of these absorbers during irradiation mostly occurs in different axial regions, hence bounded by [redacted]

The reactivity effect of the Hf-Zr RCCA (strong absorber) is also small at low burnup but substantially increases at higher burnups. [redacted]

The comparison of the IBA configurations with 104 and 200 IFBA rods shows that the latter is bounding.

In accordance with [3], [redacted]

[redacted]

7.6.4 Reactivity Effect of Axial Burnup Profiles

As discussed in Paragraph 3.3.6.4, the reactivity calculations are performed for a comparison of the axially constant burnup and the axial burnup profiles in Table 3-6 and Table 3-7 in the uniform loading of spent fuel assemblies (Region 2). Fuel enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours are considered. The results of the calculations are listed in Table 7-9. As expected, the WE 17x17 profile is bounding or equivalent in all cases. Therefore, the bounding axial burnup profile (WE 17x17) is used to determine the loading curves, as described in Paragraph 3.3.7.2. Nevertheless, both the bounding axial profile and flat profile are considered in the confirmatory calculations described in Subsection 7.7.2 and in the interface analysis in Section 7.8.

7.6.5 Reactivity Effect of Depletion Related Fuel Assembly Geometry Changes

As discussed in Paragraph 3.3.6.5, the reactivity effect of the depletion related fuel geometry changes is evaluated for spent fuel in Region 2 with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups along the expected region-specific loading curve at the cooling time of 0 hours. The results of the evaluations presented in Table 7-10 are considered as bias and bias uncertainty for determination of the maximum k_{eff} value. The maximum positive (if any) reactivity effect is treated as bias and the 95/95 uncertainty of the bias is statistically combined with the other uncertainties.

7.6.6 Spent Fuel Isotopic Content Uncertainty

The depletion uncertainty calculations, burnup uncertainty calculations and MAFP validation calculations are performed for spent fuel in Region 2, as discussed in Subparagraph 3.3.6.6.1 through Subparagraph 3.3.6.6.3, respectively. Specifically, the fuel enrichments from 2.0 to 5.0 wt% ^{235}U in increments of 0.5 wt% and fuel burnups along the expected region-specific loading curve at various cooling times are considered. The results of these calculations are presented in Table 7-11.

7.7 Design Basis Calculations

As discussed in Paragraph 3.3.7.1, various evaluations have been performed for all storage configurations listed in Chapter 1.0 to determine the bounding set of parameters, biases and bias uncertainties for the design basis model. Based on the results of these evaluations, discussed in the previous sections, the design basis calculations for normal conditions have been performed and the TCF values are determined.

7.7.1 Determination of the Spent Fuel Loading Curves

The approach to determine the individual points of the loading curves follow the process outlined in Paragraph 3.3.7.2. For various spent fuel cooling time, calculations with different burnups are performed at the spent fuel enrichments from 2.0 to 5.0 wt% ^{235}U in increments of 0.5 wt%. The results of the design basis calculations and TCF values used to generate the loading curves are presented in Table 7-12. Interpolations of the results are performed to determine the burnup that ensures that the target k_{eff} is not exceeded. The minimum required burnups are then matched by a third-order polynomial fit as a function of enrichment. The resulting equations are summarized in Table 7-13, and graphically shown in Figure 7-1 in comparison with the actual Callaway fuel inventory.

7.7.2 Confirmatory Calculations

To validate the loading curves, calculations are performed, considering various spent fuel cooling time, for selected fuel enrichments and burnups that are calculated from the polynomial functions. Both the bounding axial burnup profile and flat profile are considered in the loading curve confirmatory calculations, and the most reactive case is reported for each of the incremental enrichment steps. The results of the calculations are summarized in Table 7-14. The highest maximum k_{eff} value in Table 7-14 is consistent with the target value of 0.9950, and below the regulatory limit of 1.0 for the pool flooded with unborated water.

The results of the calculations for the checkerboard configuration of fresh fuel (Region 1) are summarized in Table 7-15, and confirm compliance with the regulatory limits for the pool flooded with unborated water. It should be noted that the maximum k_{eff} value is well below the regulatory limit of 0.95, hence no credit of the soluble boron in the SFP is applied to Region 1.

7.7.3 Maximum k_{eff} Calculation with Borated Water

The calculations with the soluble boron credit are performed for Region 2, considering various spent fuel cooling time, for selected fuel enrichments and burnups that are calculated from the polynomial functions. Both the bounding axial burnup profile and flat profile are considered, and the most reactive case is reported for each of the incremental enrichment steps. The results of the calculations are summarized in Table 7-16. The highest maximum k_{eff} value in Table 7-16 is below the regulatory limit of 0.95 for the pool flooded with borated water.

7.8 SFR Interfaces

The calculations are performed for the spent fuel rack interfaces as discussed in Section 3.4. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups calculated using the polynomial functions are considered.

[redacted]

[redacted]

As far as the fuel assembly positioning in the storage cells, both the cell centered and eccentric fuel positioning (i.e., where all assemblies are moved towards the interface as permitted by the rack geometry) are considered. [redacted]

The results of the calculations are summarized in Table 7-17, where the most reactive cases are reported. The TCF values for both loading regions across the interface are considered, and the maximum one is conservatively used to determine the maximum k_{eff} value. Additionally, the spatial distribution plots for the neutron flux and the total reaction rate are generated in order to identify the reactivity-dominating regions.

[redacted]

7.9 Normal Conditions

7.9.1 Storage of Fuel Rod Storage Rack

As discussed in Subsection 3.5.3, the calculations are performed to estimate the reactivity effect of the uniform storage of the FRSRs with fresh fuel. The results of the evaluations presented in Table 7-18 confirm that the reactivity of the SFR loaded with the FRSRs is very low, hence the FRSR is qualified for storage in any SFR cells allocated for storage of the fuel assemblies, and no further calculation is necessary.

7.9.2 Storage of Fuel Assemblies with Missing Rods

As discussed in Subsection 3.5.4, the calculations are performed to estimate the reactivity effect of the missing fuel pins for all lattices in Figure 5-4 with the assembly specific enrichment (conservatively rounded up to the next available enrichment in the depletion calculations) and burnup. Conservatively, the isotopic composition at the cooling time of 0 hours is used. Since the fuel burnup of [redacted] it is assumed to be stored in Region 1, while all other assemblies are considered in Region 2. The results of the evaluations presented in Table 7-19 confirm that the reactivity of the SFR loaded with the fuel assemblies with the missing rods is below the regulatory limits, hence these assemblies are qualified for storage in the appropriate SFR cells.

7.10 Accident Conditions

7.10.1 Misloaded Fuel Assembly

As discussed in Subsection 3.6.5, the calculations are performed for all storage configurations in Chapter 1.0 to estimate the reactivity effect of a single fresh fuel assembly misload and determine the minimum required soluble boron concentration in the SFP. Fresh fuel with the maximum fuel enrichment as well as spent fuel with enrichments of 2.0, 3.5 and 5.0 wt% ^{235}U and fuel burnups calculated using the polynomial functions at the cooling time of 0 hours and 20 years are evaluated. Both fuel radial positioning configurations, i.e., where all assemblies are cell centered and all assemblies are moved towards the misloaded assembly as permitted by the rack geometry, are considered. Additional calculations are performed to estimate the reactivity effect of a single fresh fuel assembly misload [redacted]

Several calculations at 500, 1000 and 1500 ppm of soluble boron are performed, and the bounding results used for interpolation of the minimum soluble boron concentrations are summarized in Table 7-21.

7.10.2 Incorrect Loading Curve

As discussed in Subsection 3.6.6, the calculations are performed to estimate the reactivity effect of an incorrect application of the loading curves and determine the minimum required soluble boron concentration in the SFP. The uniform spent fuel loading (Region 2) is evaluated at the maximum enrichment of 5.0 wt% ^{235}U and 0 hours cooling time, but the spent fuel composition is based on the lowest burnup from the loading curve for 20 years of cooling time (see Subsection 3.6.6).

Several calculations at 500, 1000 and 1500 ppm of soluble boron are performed, and the bounding results used for interpolation of the minimum soluble boron concentrations are summarized in Table 7-22.

7.10.3 Boron Dilution

The low and high flow rate boron dilution accident scenarios are analyzed using information in Table 5-8, following the methodology described in Subsection 3.6.8. The results of the boron dilution analysis are presented in Table 7-23.

[redacted]

[redacted]

[redacted]

[redacted]

Therefore, a boron dilution event resulting in an SFP boron concentration reduction from the technical specification limit to the minimum required concentration established in Paragraph 3.3.7.3 is not considered credible.

7.11 Margin Evaluation

The BORAL™ degradation and margin evaluation that is used in this report is described in detail in Section 3.7. The calculations are performed for all storage configurations at the maximum fuel enrichment of 5.0 wt% ^{235}U and spent fuel burnup (if applicable) that is calculated from the polynomial functions for 0 years cooling time. This case is bounding and applicable to other combinations of the spent fuel enrichment and minimum required burnup that have a larger margin to the regulatory limit.

The results of the calculations for various BORAL™ panel ^{10}B areal densities as well as the final polynomial fits for the areal density as a function of Δk are summarized in Table 7-24, and graphically shown in Figure 7-4.

The results for various margins inherent in the criticality analyses are presented in Table 7-25. The total margin for each considered configuration is calculated as a sum of the lowest individual margins. Specifically, [redacted]

The results indicate that potentially significant amounts of margin may be available to address conditions with a reduced BORAL™ panel ^{10}B areal densities (up to 20% reduction from the minimum areal density). This may help with operational considerations should such reduction be identified.

Table 7-1 – Bounding Fuel Assembly Design

Withheld from public disclosure under 10 CFR 2.390

Table 7-2 – Reactivity Effect of Fuel Assembly Parameters

Withheld from public disclosure under 10 CFR 2.390

Table 7-3 – Reactivity Effect of SFR Parameters

Withheld from public disclosure under 10 CFR 2.390

Table 7-3 – Reactivity Effect of SFR Parameters

Withheld from public disclosure under 10 CFR 2.390

Table 7-4 – Reactivity Effect of SFP Water Temperature

Withheld from public disclosure under 10 CFR 2.390

Table 7-5 – Reactivity Effect of Fuel Assembly Radial Positioning

Withheld from public disclosure under 10 CFR 2.390

Table 7-6 – Reactivity Effect of Core Operating Parameters

Withheld from public disclosure under 10 CFR 2.390

Table 7-7 – Reactivity Effect of Cooling Time

Withheld from public disclosure under 10 CFR 2.390

Table 7-8 – Reactivity Effect of Irradiation with the IBA and Fuel Inserts

Withheld from public disclosure under 10 CFR 2.390

Table 7-9 – Reactivity Effect of Axial Burnup Profile

Withheld from public disclosure under 10 CFR 2.390

Table 7-10 – Reactivity Effect of Depletion Related Fuel Assembly Geometry Changes

Withheld from public disclosure under 10 CFR 2.390

Table 7-11 – Determination of Depletion Uncertainty, Burnup Uncertainty and MAFP Bias

Withheld from public disclosure under 10 CFR 2.390

Table 7-11 – Determination of Depletion Uncertainty, Burnup Uncertainty and MAFP Bias

Withheld from public disclosure under 10 CFR 2.390

Table 7-12 – Summary of the Analysis for Region 2 (Spent Fuel)

Withheld from public disclosure under 10 CFR 2.390

Table 7-12 – Summary of the Analysis for Region 2 (Spent Fuel)

Withheld from public disclosure under 10 CFR 2.390

Table 7-12 – Summary of the Analysis for Region 2 (Spent Fuel)

Withheld from public disclosure under 10 CFR 2.390

Table 7-12 – Summary of the Analysis for Region 2 (Spent Fuel)

Withheld from public disclosure under 10 CFR 2.390

Table 7-12 – Summary of the Analysis for Region 2 (Spent Fuel)

Withheld from public disclosure under 10 CFR 2.390

Table 7-13 – Summary of the Loading Curves for Callaway SFP

Withheld from public disclosure under 10 CFR 2.390

Table 7-14 – Loading Curves Confirmatory Calculations

Withheld from public disclosure under 10 CFR 2.390

Table 7-15 – Summary of the Analysis for Region 1 (Fresh Fuel)

Withheld from public disclosure under 10 CFR 2.390

Table 7-16 – Summary of the Analysis for Normal Conditions with Soluble Boron Credit

Withheld from public disclosure under 10 CFR 2.390

Table 7-17 – Summary of the Analysis for the SFR Interfaces

Withheld from public disclosure under 10 CFR 2.390

Table 7-18 – Summary of the Analysis for the FRSR

Withheld from public disclosure under 10 CFR 2.390

Table 7-19 – Summary of the Analysis for Fuel Assemblies with Missing Rods

Withheld from public disclosure under 10 CFR 2.390

Table 7-20 – Deleted

Table 7-21 – Maximum k_{eff} Calculation for the Fuel Misload Accident

Withheld from public disclosure under 10 CFR 2.390

Table 7-21 – Maximum keff Calculation for the Fuel Misload Accident

Withheld from public disclosure under 10 CFR 2.390

Table 7-21 – Maximum keff Calculation for the Fuel Misload Accident

Withheld from public disclosure under 10 CFR 2.390

Table 7-22 – Maximum k_{eff} Calculation for the Incorrect Loading Curve Accident

Withheld from public disclosure under 10 CFR 2.390

Table 7-23 – SFP Boron Dilution Accident Analysis

Withheld from public disclosure under 10 CFR 2.390

Table 7-24 – Reactivity Effect of the BORAL™ Panel ^{10}B Areal Density

Withheld from public disclosure under 10 CFR 2.390

Table 7-25 – Margin Evaluation

Withheld from public disclosure under 10 CFR 2.390

Table 7-26 – Reactivity Effect of the B_4C Particle Size

Withheld from public disclosure under 10 CFR 2.390

Withheld from public disclosure under 10 CFR 2.390

Figure 7-1 – Loading Curves for Uniform Loading of Spent Fuel Assemblies (Region 2)

Withheld from public disclosure under 10 CFR 2.390

Figure 7-2 – Total Reaction Rate Distribution for Region 1 to Region 2 Interface

Withheld from public disclosure under 10 CFR 2.390

Figure 7-3 – Total Reaction Rate Distribution for Region 1 (2x2) to Region 2 Interface

Withheld from public disclosure under 10 CFR 2.390

Figure 7-4 – BORAL™ Panel ^{10}B Areal Density as a Function of Δk

8.0 CONCLUSION

The criticality safety analyses have been performed for the Callaway SFP that contains a single type of BORAL™ spent fuel racks designed for storage of the PWR 17x17 fuel assemblies. Two storage configurations listed in Chapter 1.0, including uniform loading of spent fuel assemblies with various cooling times and a checkerboard configuration of fresh fuel assemblies and empty storage cells, have been qualified using the bounding fuel assembly designs – Framatome GAI and Westinghouse V+, respectively, with fuel enrichment up to 5.0 wt% ^{235}U. All credible normal and accident conditions have been analyzed, and the key conclusions are provided below.

For the fresh fuel assemblies in Region 1 under normal conditions, the effective neutron multiplication factor (k_{eff}) of the SFP loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 0.95 for the pool flooded with unborated water with 95% probability at a 95% confidence level, in accordance with 10 CFR 50.68(b)(4).

For spent fuel assemblies in Region 2 under normal conditions, considering various cooling times, the minimum required burnups as a function of enrichment (a third-order polynomial fit) have been determined and summarized in Table 7-13 as well as in Figure 7-1. The results of the calculations show that the effective neutron multiplication factor (k_{eff}) of the spent fuel pool loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with unborated water and does not exceed 0.95 for the pool flooded with borated water (550 ppm[1]), all for 95% probability at a 95% confidence level, in accordance with 10 CFR 50.68(b)(4).

Under accident conditions, the minimum soluble boron concentration of 1081.2 ppm[1] is required to ensure that the effective neutron multiplication factor (k_{eff}) of the SFP does not exceed 0.95.

All credible interface conditions in the Callaway SFP have been considered and qualified.

Any SFR cell qualified for loading of a fuel assembly is also permitted for storage of the FRSR. Specific Callaway fuel inventory, such as fuel assemblies with certain missing rods and low-burned fuel assemblies, have been analyzed or evaluated and qualified for storage in the SFP. Fuel assembly reconstitution activities are restricted to a storage cell in the Region 1 configuration that is face adjacent to empty cells at all sides.

The criticality safety analysis documented in this report also provides information about the potential reactivity effect of lower BORAL™ panel ^{10}B areal densities (up to 20% reduction from the minimum areal density), and about margin in the analysis to possibly offset such reduction.

---

[1] The soluble boron requirements are increased by additional 50 ppm in accordance with Paragraph 5.1.1 of [3].

The key criticality analysis parameters for Callaway, which must be specifically tracked to ensure continual compliance with the criticality safety analysis, are summarized in Appendix B. This also includes a set of rules to be followed, in Section B-4.0, to assure placement of assemblies in the racks is in accordance with the analyses presented here.

9.1 REFERENCES

[1] "Criticality Safety Analysis of Spent Fuel Storage Racks for Union Electric," Holtec International, HI-971790, Revision 4 (proprietary).

[2] "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," US NRC, DSS-ISG-2010-01, Revision 0.

[3] "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water-Reactor Power Plants," Nuclear Energy Institute, NEI 12-16, Revision 4.

[4] "Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report SSP-14-P01/028-TR, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors," Office of Nuclear Reactor Regulation, US NRC, September, 2017.

[5] "CASMO5/CASMO5M A Fuel Assembly Burnup Program Methodology Manual," Studsvik Scandpower, Inc., SSP-08/405, Revision 5.

[6] "CASMO5 A Fuel Assembly Burnup Program, User's Manual," Studsvik Scandpower, Inc., SSP-07/431, Revision 10.

[7] "ENDF/B-VII.0 586 Group Neutron Data Library for CASMO-5 and CASMO-5M," Studsvik Scandpower, Inc., SSP-07/402, Revision 5.

[8] "MCNP - A General Monte Carlo N-Particle Transport Code, Version 5," Los Alamos National Laboratory, LA-UR-03-1987 (2003, Revised 2/1/2008).

[9] "ENDF/B-VII.0 Evaluated Nuclear Data Library," December 15, 2006.

[10] "MCNP5 Temperature Dependent Library Construction with NJOY," Holtec International, HI-2156450, Revision 2 (proprietary).

[11] F. B. Brown, "A Review of Monte Carlo Criticality Calculations – Convergence, Bias, Statistics," International Conference on Mathematics, Computational Methods and Reactor Physics, Saratoga Springs, NY, LA-UR-08-6558, May 3, 2009.

[12] "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," USNRC, NUREG/CR-6698, January 2001.

[13] "Nuclear Group Computer Code Benchmark Calculations," Holtec International, HI-2104790,

Revision 3 (proprietary).

[14] "BORAL Documentation Package," DOC-70384-3, Project 70384, Holtec International, June 1999 (proprietary).

[15] "Safety Analysis Report HI-STAR 100 Cask System," Holtec International, USNRC Docket 71-9261, HI-951251, Revision 20 (proprietary).

[16] "Assessment of Reactivity Margin and Loading Curves for PWR Burnup-Credit Cask Designs," Oak Ridge National Laboratory, NUREG/CR-6800, ORNL/TM-2002/6, March 2003.

[17] M. D. DeHart, "Sensitivity and Parametric Evaulations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," Oak Ridge National Laboratory, ORNL/TM-12973, May 1996.

[18] "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," ORNL-TM-2000/373, NUREG/CR-6761, USNRC Office of Nuclear Regulatory Research, March 2002.

[19] "Ameren Inputs for Callaway Spent Fuel Pool Criticality Analysis," NED20220010, Ameren Missouri, February 8, 2022.

[20] "Westinghouse Inputs for Callaway Spent Fuel Pool Criticality Analysis," NED20220012, Ameren Missouri, February 22, 2022.

[21] Inputs for Callaway Spent Fuel Pool Criticality Analysis, Framatome Fuel, FS1-0058769, Revision 2.

[22] "Parametric Study of the Effect of Control Rods for PWR Burnup Credit," ORNL-TM-2001/69, NUREG/CR-6759, USNRC Office of Nuclear Regulatory Research, February 2002.

[23] "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," ORNL-TM-2000/321, NUREG/CR-6760, USNRC Office of Nuclear Regulatory Research, March 2002.

[24] R. J. Cacciapouti and S. Van Volkinburg, "Axial Burnup Profile Database for Pressurized Water Reactors," Yankee Atomic Electric Company Report YAEC-1937, 1997.

[25] "Summary Report of Commercial Reactor Criticality Data for McGuire Unit 1," B00000000-01717-5705-00063, Revision 1, U.S. Department of Energy, April 1998.

[26] "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," ORNL/TM-2001/273, NUREG/CR-6801, USNRC Office of Nuclear Regulatory Research, March 2003.

[27] An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses - Criticality (Keff) Predictions, ORNL/TM-2011/514, NUREG/CR-7109, USNRC Office

of Nuclear Regulatory Research, April 2012.

[28] Bias Estimates Used in Lieu of Validation of Fission Products and Minor Actinides in MCNP Keff Calculations for PWR Burnup Credit Casks, ORNL/TM-2012/544, NUREG/CR-7205, USNRC Office of Nuclear Regulatory Research, September 2015.

[29] Pool Layout for Spent Fuel Racks, Holtec International, Drawing 1997, Revision 1. [30]

"Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks," ORNL/TM-2002/255, NUREG/CR-6835, USNRC Office of Nuclear Material Safety and Safeguards, September 2003.

[31] "Overview of BORAL Performance Based Upon Surveillance Coupon Measurements," EPRI Report 1021052, Palo Alto, CA, December 2010.

[32] Rack Construction - Region II, Holtec International, Drawing 1998, Revision 1.

[33] Rack Construction - Region II, Holtec International, Drawing 1999, Revision 1.

[34] "Ameren Spent Fuel Pool Levels for Dilution Event Analysis," NED20220013, Ameren Missouri, February 28, 2022.

[35] "Ameren Supplemental Inputs for Dilution Event Analysis," NED20220033, Ameren Missouri, June 20, 2022.

[36] "Fresh and Spent Fuel Pool Criticality Analyses," US NRC Regulatory Guide RG 1.240, March 2021.

APPENDIX A NEI 12-16 CRITICALITY ANALYSIS CHECKLIST

The criticality analysis checklist provides a summary of the evaluation that confirms that all the applicable subject areas are addressed in this report, and all the alternative approaches are identified and justified.

The checklist also assists the NRC reviewer in identifying areas of the analysis that conform or do not conform to the guidance in NEI 12-16 [3]. Subsequently, the NRC review can then be more efficiently focused on those areas that deviate from NEI 12-16 and the justification for those deviations.

Subject	Included	Notes / Explanation
1.0 Introduction and Overview		
Purpose of submittal	**YES**/NO	Chapter 1.0
Changes requested	**YES**/NO	Chapter 1.0
Summary of physical changes	**YES**/NO	Chapters 1.0 and 8.0, Appendix B
Summary of Tech Spec changes	**YES**/NO	Chapters 1.0 and 8.0, Appendix B
Summary of analytical scope	**YES**/NO	Chapter 1.0, Section 3.3
2.0 Acceptance Criteria and Regulatory Guidance		
Summary of requirements and guidance	**YES**/NO	Chapter 2.0
Requirements documents referenced	**YES**/NO	
Guidance documents referenced	**YES**/NO	
Acceptance criteria described	**YES**/NO	
3.0 Reactor and Fuel Design Description		
Describe reactor operating parameters	**YES**/NO	Section 5.2, Paragraph 3.3.6.1
Describe all fuel in pool	**YES**/NO	Section 5.1, Subsection 3.3.1
Geometric dimensions (nominal and tolerances)	**YES**/NO	Section 5.1, Subsection 3.3.2
Schematic of guide tube patterns	**YES**/NO	Section 5.1, Figure 3-1
Material compositions	**YES**/NO	Section 5.6
Describe future fuel to be covered	**YES**/NO	Section 3.8
Geometric dimensions (nominal and tolerances)	**YES**/NO	
Schematic of guide tube patterns	**YES**/NO	
Material compositions	**YES**/NO	
Describe all fuel inserts	**YES**/NO	Section 5.3, Paragraph 3.3.6.3
Geometric dimensions (nominal and tolerances)	**YES**/NO	
Schematic (axial/cross-section)	**YES**/NO	

Subject	Included	Notes / Explanation
Material compositions	**YES**/NO	
Describe non-standard fuel	**YES**/NO	Sections 5.7 and 5.8, Subsections 3.5.3 and 3.5.4
Geometric dimensions	**YES**/NO	
Describe non-fuel items in fuel cells	YES/**NO**	Not applicable
Nominal and tolerance dimensions	YES/**NO**	
4.0 Spent Fuel Pool/Storage Rack Description		
New fuel vault & storage rack description	YES/**NO**	Not applicable
Nominal and tolerance dimensions	-	
Schematic (axial/cross-section)	-	
Material compositions	-	
Spent fuel pool, storage rack description	**YES**/NO	Section 5.4, Subsection 3.3.3
Nominal and tolerance dimensions	**YES**/NO	Section 5.4, Subsection 3.3.3
Schematic (axial/cross-section)	**YES**/NO	Figure 5-2, Figure 5-3
Material compositions	**YES**/NO	Section 5.6
Other reactivity control devices (inserts)	YES/**NO**	Not applicable
Nominal and tolerance dimensions	-	
Schematic (axial/cross-section)	-	
Material compositions	-	
5.0 Overview of the Method of Analysis		
New fuel rack analysis description	YES/**NO**	Not applicable
Storage geometries	-	
Bounding assembly design(s)	-	
Integral absorber credit	-	
Accident analysis	-	
Spent fuel storage rack analysis description	**YES**/NO	
Storage geometries	**YES**/NO	Chapter 1.0, Section 3.3
Bounding assembly design(s)	**YES**/NO	Subsection 3.3.1
Soluble boron credit	**YES**/NO	Chapter 1.0, Paragraph 3.3.7.3
Boron dilution analysis	**YES**/NO	Subsection 3.6.8
Burnup credit	**YES**/NO	Chapter 1.0, Subsection 3.3.6
Decay/cooling time credit	**YES**/NO	Chapter 1.0, Subsection 3.3.6, Paragraphs 3.3.6.2 and 3.3.7.2
Integral absorber credit	YES/**NO**	Section 3.3
Other credit	**YES**/NO	Empty storage cells, Chapter 1.0
Fixed neutron absorbers	**YES**/NO	Chapter 1.0, Section 3.3, Subsection 3.3.3
Aging management program	**YES**/NO	Subsection 3.7.1
Accident analysis	**YES**/NO	Section 3.6
Temperature increase	**YES**/NO	Subsection 3.6.1

Subject	Included	Notes / Explanation
Assembly drop	**YES**/NO	Subsections 3.6.2 and 3.6.3
Single assembly misload	**YES**/NO	Subsection 3.6.5
Multiple misload	**YES**/NO	Subsection 3.6.6
Boron dilution	**YES**/NO	Subsection 3.6.8
Other	**YES**/NO	Fuel mislocation (Subsection 3.6.4), Rack movement (Subsection 3.6.7)
Fuel out of rack analysis	**YES**/NO	
Handling	**YES**/NO	Subsection 3.5.2
Movement	**YES**/NO	Subsection 3.5.1
Inspection	**YES**/NO	Subsection 3.5.2
6.0 Computer Codes, Cross Sections and Validation Overview		
Code/modules used for calculation of k_{eff}	**YES**/NO	Subsection 3.2.2
Cross section library	**YES**/NO	Subsection 3.2.2
Description of nuclides used	**YES**/NO	Section 5.6, Subsection 3.3.6
Convergence checks	**YES**/NO	Subsection 3.2.2
Code/module used for depletion calculation	**YES**/NO	Subsection 3.2.1
Cross section library	**YES**/NO	Subsection 3.2.1
Description of nuclides used	**YES**/NO	Subsection 3.3.6
Convergence checks	YES/**NO**	Not applicable
Validation of code and library	**YES**/NO	Paragraph 3.2.2.1, Subparagraph 3.3.6.6.1
Major actinides and structural materials	**YES**/NO	Paragraph 3.2.2.1
Minor actinides and fission products	**YES**/NO	Subparagraph 3.3.6.6.3
Absorbers credited	**YES**/NO	Paragraph 3.2.2.1
7.0 Criticality Safety Analysis of the New Fuel Rack	YES/**NO**	Not applicable
Rack model	-	
Boundary conditions	-	
Source distribution	-	
Geometry restrictions	-	
Limiting fuel design	-	
Fuel density	-	
Burnable poisons	-	
Fuel dimensions	-	
Axial blankets	-	
Limiting rack model	-	
Storage vault dimensions and materials	-	
Temperature	-	
Multiple regions/configurations	-	
Flooded	-	

Subject	Included	Notes / Explanation
Low density moderator	-	
Eccentric fuel placement	-	
Tolerances	-	
Fuel geometry	-	
Fuel pin pitch	-	
Fuel pellet OD	-	
Fuel clad OD	-	
Fuel content	-	
Enrichment	-	
Density	-	
Integral absorber	-	
Rack geometry	-	
Rack pitch	-	
Cell wall thickness	-	
Storage vault dimensions/materials	-	
Code uncertainty	-	
Biases	-	
Temperature	-	
Code bias	-	
Moderator conditions	-	
Fully flooded and optimum density moderator	-	
8.0 Depletion Analysis for Spent Fuel		
Depletion model considerations	**YES**/NO	Subsections 3.2.1 and 3.3.6
Time step verification	**YES**/NO	Subsection 3.3.6
Convergence verification	YES/**NO**	Not applicable
Simplifications	**YES**/NO	Subsection 3.3.6, Chapter 4.0
Non-uniform enrichments	YES/**NO**	Section 3.3
Post depletion nuclide adjustment	**YES**/NO	Subsection 3.3.6
Cooling time	**YES**/NO	Chapter 1.0, Subsection 3.3.6, Paragraphs 3.3.6.2 and 3.3.7.2
Depletion parameters	**YES**/NO	Sections 5.2 and 5.3, Subsection 3.3.6
Burnable absorbers	**YES**/NO	Section 5.3, Subparagraph 3.3.6.3.1
Integral absorbers	**YES**/NO	Section 5.3, Subparagraph 3.3.6.3.3
Soluble boron	**YES**/NO	Section 5.2, Paragraph 3.3.6.1
Fuel and moderator temperature	**YES**/NO	Section 5.2, Paragraph 3.3.6.1
Power	**YES**/NO	Section 5.2, Paragraph 3.3.6.1
Control rod insertion	**YES**/NO	Section 5.3, Subparagraph 3.3.6.3.2
Atypical cycle operating history	**YES**/NO	Subparagraph 3.3.6.3.2
9.0 Criticality Safety Analysis of Spent Fuel Pool Storage Racks		

Subject	Included	Notes / Explanation
Rack model	**YES**/NO	Section 3.3
Boundary conditions	**YES**/NO	Section 3.3
Source distribution	**YES**/NO	Subsection 3.2.2
Geometry restrictions	YES/**NO**	Not applicable
Design basis fuel description	**YES**/NO	Section 3.3, Section 7.1, Chapter 4.0
Fuel density	**YES**/NO	Section 3.3, Subsection 3.3.2, Section 7.2
Burnable poisons	**YES**/NO	Section 3.3, Subparagraph 3.3.6.3.3, Subsection 7.6.3
Fuel assembly inserts	**YES**/NO	Section 3.3, Subparagraphs 3.3.6.3.1 and 3.3.6.3.2, Subsection 7.6.3
Fuel dimensions	**YES**/NO	Section 3.3, Subsection 3.3.2, Section 7.2
Axial blankets	YES/**NO**	Section 3.3, Chapter 4.0
Configurations considered	**YES**/NO	Chapter 1.0, Subsection 7.7.2
Borated	**YES**/NO	Chapter 1.0, Paragraph 3.3.7.3, Subsection 7.7.3
Unborated	**YES**/NO	Chapter 1.0, Paragraph 3.3.7.2, Subsections 7.7.1 and 7.7.2
Multiple rack designs	YES/**NO**	Section 5.4
Alternate storage geometry	YES/**NO**	Not applicable
Reactivity control devices	YES/**NO**	Not applicable
Fuel assembly inserts	-	
Storage cell inserts	-	
Storage cell blocking devices	-	
Axial burnup shapes	**YES**/NO	Paragraph 3.3.6.4
Uniform/distributed	**YES**/NO	Paragraph 3.3.6.4, Subsections 7.6.4, 7.7.2 and 7.7.3
Nodalization	**YES**/NO	Paragraph 3.3.6.4
Blankets modeled	YES/**NO**	Section 3.3, Chapter 4.0
Tolerances/uncertainties	**YES**/NO	Section 3.3, Paragraph 3.3.7.1, Figure 3-3
Fuel geometry	**YES**/NO	Subsection 3.3.2, Figure 3-3
Fuel rod pin pitch	**YES**/NO	Subsection 3.3.2, Section 7.2
Fuel pellet OD	**YES**/NO	Subsection 3.3.2, Section 7.2
Cladding OD	**YES**/NO	Subsection 3.3.2, Section 7.2
Axial fuel position	**YES**/NO	Section 3.3, Subsection 3.3.3
Fuel content	**YES**/NO	Subsections 3.3.2 and 3.3.6, Sections 7.2 and 7.6
Enrichment	**YES**/NO	Subsection 3.3.2, Section 7.2
Density	**YES**/NO	Subsection 3.3.2, Section 7.2
Assembly insert dimensions and materials	YES/**NO**	Subparagraph 3.3.6.3.4

Subject	**Included**	**Notes / Explanation**
Rack geometry	**YES**/NO	Subsection 3.3.3, Figure 3-3, Section 7.3
Flux-trap size (width)	YES/**NO**	Not applicable
Rack cell pitch	**YES**/NO	Subsection 3.3.3, Section 7.3
Rack wall thickness	**YES**/NO	Subsection 3.3.3, Section 7.3
Neutron absorber dimensions	**YES**/NO	Subsection 3.3.3, Section 7.3
Rack insert dimensions and materials	YES/**NO**	Not applicable
Code validation uncertainty	**YES**/NO	Paragraph 3.2.2.1, Figure 3-3
Criticality case uncertainty	**YES**/NO	Figure 3-3
Depletion uncertainty	**YES**/NO	Subparagraph 3.3.6.6.1, Figure 3-3, Subsection 7.6.6
Burnup uncertainty	**YES**/NO	Subparagraph 3.3.6.6.2, Figure 3-3, Subsection 7.6.6
Biases	**YES**/NO	Section 3.3, Figure 3-3
Design basis fuel design	YES/**NO**	Bounding fuel is used in the design basis model. Subsection 3.3.1, Section 7.1
Code bias	**YES**/NO	Paragraph 3.2.2.1, Figure 3-3
Temperature	YES/**NO**	Bounding temperature is used in the design basis model. Section 3.3, Subsections 3.3.4 and 3.6.1, Section 7.4
Eccentric fuel placement	YES/**NO**	Bounding fuel placement is used in the design basis model. Subsection 3.3.5, Section 7.5
In-core thimble depletion effect	YES/**NO**	Bounding effect is used in the design basis model. Subparagraph 3.3.6.3.4, Subsection 7.6.3
NRC administrative margin	YES/**NO**	No additional administrative margin, over and above what is already prescribed by the regulations and guidance documents.
Modeling simplifications	**YES**/NO	Section 3.3, Chapter 4.0
Identified and described	**YES**/NO	Section 3.3, Chapter 4.0
10.0 Interface Analysis		
Interface configurations analyzed	**YES**/NO	Sections 3.4 and 7.8
Between dissimilar racks	**YES**/NO	Sections 3.4 and 7.8
Between storage configurations within a rack	**YES**/NO	Sections 3.4 and 7.8
Interface restrictions	YES/**NO**	Chapter 8.0
11.0 Normal Conditions		
Fuel handling equipment	**YES**/NO	Bounded by normal storage conditions. Subsection 3.5.1

Subject	Included	Notes / Explanation
Administrative controls	**YES**/NO	Chapters 1.0, Subsections 3.7.1 and 7.10.3
Fuel inspection equipment or processes	**YES**/NO	Bounded by normal storage conditions. Subsection 3.5.2
Fuel reconstitution	**YES**/NO	Subsections 3.5.2 and 3.5.4
12.0 Accident Analysis		
Boron dilution	**YES**/NO	Subsection 3.6.8
Normal conditions	**YES**/NO	Subsections 3.6.8 and 7.10.3
Accident conditions	**YES**/NO	Subsection 3.6.8
Single assembly misload	**YES**/NO	Subsections 3.6.5 and 7.10.1
Fuel assembly misplacement	**YES**/NO	Subsection 3.6.4
Neutron absorber insert misload	YES/**NO**	Not applicable
Multiple fuel misloads	**YES**/NO	Subsections 3.6.6 and 7.10.2
Dropped assembly	**YES**/NO	Subsections 3.6.2 and 3.6.3
Temperature	**YES**/NO	Subsection 3.6.1
Seismic event/other natural phenomena	**YES**/NO	Subsection 3.6.7
13.0 Analysis Results and Conclusions		
Summary of results	**YES**/NO	Chapter 8.0
Burnup curve(s)	**YES**/NO	Table 7-13, Figure 7-1
Intermediate decay time treatment	YES/**NO**	-
New administrative controls	**YES**/NO	Table 7-12, Chapter 8.0, Appendix B, Subsections 3.5.4 and 3.5.5
Technical Specification markups	YES/**NO**	-
14.0 References	**YES**/NO	Chapter 9.0
Appendix A: Computer Code Validation:		
Code validation methodology and biases	**YES**/NO	Paragraph 3.2.2.1
New fuel	**YES**/NO	Paragraph 3.2.2.1, Table 3-2
Depleted fuel	**YES**/NO	Paragraph 3.2.2.1, Table 3-2
MOX	**YES**/NO	Paragraph 3.2.2.1, Table 3-2
HTC	**YES**/NO	Paragraph 3.2.2.1, Table 3-2
Convergence	**YES**/NO	
Trends	**YES**/NO	Paragraph 3.2.2.1, Table 3-3
Bias and uncertainty	**YES**/NO	Paragraph 3.2.2.1, Table 3-4
Range of applicability	**YES**/NO	Paragraph 3.2.2.1, Table 3-1
Analysis of area of applicability coverage	**YES**/NO	Paragraph 3.2.2.1, Table 3-1

APPENDIX B SUMMARY OF KEY PARAMETERS

Table of Contents

B-1.0 INTRODUCTION

This appendix documents the key criticality analysis parameters for Callaway, which must be specifically tracked to ensure continual compliance with the criticality safety analysis. As discussed in Chapter 1.0, the criticality safety analysis, that takes credit for various combinations of the following, is provided in the main body of the report:

- Fixed neutron absorbers: BORAL™ poison panels;
- Burnup of spent fuel assemblies;
- Spent fuel cooling time;
- Empty SFR storage cells;
- Soluble boron in the SFP.

Changes to the storage rack design, neutron absorber or soluble boron content should be evaluated under another process. Therefore, the focus of the parameters discussed in this appendix is related to the fuel design, SFP fuel arrangement and core operating parameters only. It is also assumed that the fuel design will not vary from the currently used version of the PWR 17x17 fuel assembly. The key parameters for the criticality safety analysis in the main body of the report are presented in the following sections.

B-2.0 STANDARD KEY PARAMETERS

For the burnup credit analyses, the parameters provided in Table B-1 either have an impact on the analysis uncertainties or have an impact on the analysis directly (bias), and therefore should be treated as parameters that may have an impact to the analysis results.

Table B-1 – Summary of the Standard Key Parameters

Parameter	Limiting Value	Impact
Fuel cladding OD, inches	[redacted]	Bias and Uncertainty
Fuel rod pitch, inches	[redacted]	Bias and Uncertainty
Fuel pellet OD, inches	[redacted]	Bias and Uncertainty
Fuel enrichment, wt% ^{235}U	[redacted]	Bias and Uncertainty
Fuel density, g/cm^3	[redacted]	Bias
Distance from Bottom of Fuel Assembly to Beginning of Active Length, inches	[redacted]	Bias

---

[1] UO_2 theoretical density.

B-3.0 KEY PARAMETERS FOR THE BURNUP CREDIT

In additional to the parameters presented in Section B-2.0, for the analyses of the loading regions that involve spent fuel with burnup credit, the parameters provided in Table B-2 either have an impact on the analysis uncertainties or have an impact on the analysis directly (bias), and therefore should be treated as parameters that may have an impact to the analysis results.

Table B-2 – Summary of Key Parameters for the Burnup Credit

Parameter	Limiting Value	Impact
Core Operating Parameters		
Maximum fuel temperature, K	[redacted]	Bias
Maximum core moderator temperature, K	[redacted]	Bias
Soluble boron concentration (cycle average), ppm	[redacted]	Bias
Fuel Inserts and IBA		
Irradiation duration with fuel inserts, GWd/mtU	[redacted]	Bias
WABA[1] absorber content, wt% B_4C	[redacted]	Bias
WABA absorber ID, inches	[redacted]	
WABA absorber OD, inches	[redacted]	Bias
Number of the IFBA rods	[redacted]	Bias
IFBA[1] ZrB_2 coating loading, mg ^{10}B/inch	[redacted]	Bias
Burnup-weighted cycle-average RCCA insertion depth during full power operation[2], inches	[redacted]	Bias (potential)
Depletion Related Fuel Geometry Changes		
Fuel rod growth, inches	[redacted]	Bias and Uncertainty
Fuel grid growth, inches	[redacted]	Bias and Uncertainty
Other Parameters		
Axial burnup profile	[redacted]	Bias
Burnup uncertainty, %	[redacted]	Uncertainty
Cooling time for Region 2, years	[redacted]	Different loading curves

---

[1] WABA and IFBA have been considered in all depletion analyses for the design basis calculations.

[2] The full power operation here means any reactor operation other than the short-term transients (e.g., reactor startup, shutdown, etc.) that may provide a reasonable contribution to the fuel exposure.

B-4.0 PROPOSED RULES FOR PERMISSIBLE LOADING

The proposed rules for permissible loading of the spent fuel racks are as follows:

1. The permissible content for any rack cell depends on the content of the rack cells that are face adjacent to that cell.
2. Rack cells that face each other across a rack-to-rack gap are considered face-adjacent.
3. A cell can either
    a. Contain a Region 1 assembly; or
    b. Contain a Region 2 assembly; or
    c. Be empty.
4. The placement rules are as follows. All requirements applicable to a cell must be met.
    4.1 For cells containing a Region 1 assembly
        4.1.1 None of the face-adjacent cells may contain a Region 1 assembly;
        4.1.2 A minimum of two (2) of the face-adjacent cells must be empty;
        4.1.3 The remaining face-adjacent cells may contain a maximum of two (2) Region 2 assemblies. See also rule 4.2.3;
        4.1.4 If both remaining face-adjacent cells contain Region 2 assemblies, then rule 4.2.1 is restricted to one (1) Region 1 assembly for those cells.
    4.2. For cells containing a Region 2 assembly
        4.2.1 A maximum of two (2) of the face-adjacent cells may contain a Region 1 assembly. See also rule 4.1.4;
        4.2.2 The remaining face-adjacent cells may be empty or contain a Region 2 assembly;
        4.2.3 If two (2) face-adjacent cells contain Region 1 assemblies, then rule 4.1.3 is restricted to one (1) Region 2 assembly for those cells.

APPENDIX C RG 1.240 COMPLIANCE

The table below provides a summary that confirms compliance with clarifications and exceptions to the guidance in NEI 12-16 [3], which are explicitly mentioned in the Regulatory Guide RG 1.240 [36]. The table assists the NRC reviewer in verifying compliance with RG 1.240.

Item C.1.x	Subject	Notes / Explanation
a	Section 1.4 states that the double contingency principle, as applied to criticality accidents, means, in part, that licensees do not need to consider the simultaneous occurrence of two independent and unlikely conditions... However, if no controls or documents exist to preclude such a condition, then the licensee or applicant should treat it as part of the normal condition.	All accident events have been explicitly evaluated in Sections 3.6 and 7.10. None of these events is considered applicable to the normal condition.
b	... Licensees or applicants should establish how they will maintain any excess safety margins being used to justify assumptions or simplifications when they update the criticality analyses, using their approved methodology, to accommodate changes in the fuel storage characteristics.	All the major assumptions and simplifications in the design basis calculations provided in Section 3.3 and Chapter 4.0 are conservative. No extra margin is used to justify assumptions or simplifications in the design basis model.
c	Section 2 discusses acceptance criteria for fresh fuel vault storage...	Not applicable
d	Section 3.1.3 discusses the treatment of nuclides credited in the depletion and criticality analysis; however, it doesn't provide any guidance on the treatment of lumped fission products...	No lumped fission products are used.
e	Section 4.2.3 states that the depletion bias and uncertainty described in this section account for all uncertainties associated with depletion. If licensees are following the guidance in Section 4.2.3 about treatment of the depletion parameters, the staff would find this approach acceptable.	The guidance in Section 4.2.3 of NEI 12-16 is followed.
f	... Each unique axial plane in the bundle designs should be evaluated. For example, some bundle designs may use different fuel rod pitches at different axial planes. Licensees or applicants should justify their selection of lattice parameters for evaluation.	All the axial variations of the fuel lattice parameters have been reviewed and they are either considered conservatively (e.g., IBA with the cutback regions are neglected) or neglected (e.g., blankets).

Item C.1.x	Subject	Notes / Explanation
g	Section 5.1.6 discusses a conservative approach to modeling integral burnable absorbers using nominal dimensions combined with a minimum absorber loading...	Not applicable
h	Section 5.2.2 states that credit can be taken for radial leakage near the walls of the spent fuel pool for allowing lower burnup fuel requirements on the periphery of the spent fuel pool.	Not applicable
i	Section 5.2.2.4 provides recommendations on the treatment of eccentric positioning for fuel assemblies within spent fuel pool cells ... Licensees or applicants should consider any unique aspects of the configuration being analyzed that may lead to a more limiting eccentric positioning.	All eccentric positioning configurations have been explicitly analyzed.
j	... The NRC agrees that the limiting abnormal condition will be the one which requires the highest soluble boron to meet regulatory requirements. However, while misloading events are typically the limiting abnormal condition, that is not always the case. Therefore, licensees or applicants should consider all credible abnormal and accident conditions.	Both a single assembly misload and multiple misload event have been analyzed, and a soluble boron requirement for the limiting accident condition has been determined.
k	Section 9.4 lists some parameters that may need to be verified as part of post irradiation fuel characterization activities. One of the parameters is "soluble boron (burnup averaged)." The NRC endorses use of cycle burnup averaged soluble boron, consistent with Section 4.2.1, but the NRC does not endorse other interpretations of the phrase "burnup averaged," such as averaging across the whole burnup range for a given fuel assembly.	A cycle burnup averaged soluble boron concentration has been used in the depletion calculations and presumed to be used as a part of post irradiation fuel characterization activities.
l	... An important aspect of validation that is not covered in much detail is the importance of selecting appropriately representative benchmarks and critical experiments, especially when performing trend evaluation. Licensees or applicants may need to consider smaller sets of data to avoid confounding effects that obscure trends or that lead to conclusions based on data that are not highly representative of the spent fuel pool geometry and compositions of interest.	Various subsets of the critical experiments have been evaluated in the benchmark analysis, and the applicable subsets were used for different loading regions and conditions.

Item C.1.x	Subject	Notes / Explanation
m	Section A.2.2 states that startup critical data from boiling-water reactors (BWRs) can be used to benchmark depletion codes and compute a bias and bias uncertainty...	Not applicable
n	Section A.4 discusses use of a secondary code as an intermediate means to validate the primary code used for the nuclear criticality safety analyses...	Not applicable
o	NEI 12-16, Revision 4, provides many recommendations that are based on analyses performed using typical geometries and compositions associated with spent fuel pools and bundle designs that are currently in widespread use in the United States (e.g., cylindrical uranium dioxide fuel pellets enclosed in zirconium alloy tubes). Novel configurations and concepts, such as accident-tolerant fuel designs, may require justification for continued use of the assumptions. For example, dispositions of specific uncertainties as not significant may no longer be valid, simplifying assumptions may become nonconservative, and additional uncertainties may need to be considered. Licensees or applicants are responsible for justifying use of the guidance in NEI 12-16, Revision 4, in any such applications.	The fuel assembly designs with the typical geometries and compositions are operated and stored in the Callaway SFP.
p	NEI 12-16, Revision 4, includes some general conclusions based on sensitivity studies performed to support the guidance ... Licensees or applicants should ensure that a conclusion is applicable to their circumstances before implementing the guidance associated with that conclusion.	All such applicable general conclusions are confirmed to remain applicable to the design basis model used in the criticality calculations.
q	Appendix B to NEI 12-16, Revision 4, includes an example to supplement the guidance... Licensees or applicants should ensure that the example in Appendix B is applicable to their circumstances before implementing the guidance as described in the example.	The applicable depletion related fuel assembly geometry changes have been explicitly evaluated in Paragraph 3.3.6.5 and Subsection 7.6.5.