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Comanche Peak Nuclear Power Plant Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis

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REVISION HISTORY

Revision	Description and Impact of the Change	Date
0	This is the original version.	03/2013
1	This version is identical to the original version except where changes in identification of proprietary information have been made.	10/2013

TRADEMARK NOTICE

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LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS

2-D	Two-Dimensional
3-D	Three-Dimensional
AEG	Average Energy Group of Neutrons Causing Fission
ANP	Designation for fuel made by Siemens/Framatom ANP
AOA	Area of Applicability
at%	Atom Percent
BA	Burnable Absorber
BORAL	BORAL®
Comanche Peak	Comanche Peak Nuclear Power Plant
Decay time	Post-irradiation cooling time
EALF	Energy of Average Lethargy causing Fission
En	Enrichment
ENDF/B	Evaluated Nuclear Data File
GT	Guide Tube
GWd	Gigawatt-days
HTC	Haut Taux de Combustion
ID	Inner Dimension
IFBA	Integral Fuel Burnable Absorber
ISG	Interim Staff Guidance
IT	Instrumentation Tube
k_{eff}	Effective neutron multiplication factor
KENO	SCALE Module KENO V.a
LWR	Light Water Reactor
MTU	Metric Ton Uranium
MWt	Megawatts-thermal
NITAWL	NITAWL-III
OD	Outer Dimension
OECD	Organization for Economic Co-operation and Development
OFA	Optimized Fuel Assembly
ORNL	Oak Ridge National Lab
Plexiglas	Plexiglas™
PNL	Pacific Northwest National Laboratory
ppm	parts per million
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assemblies
RCS	Reactor Coolant System
SFP	Spent Fuel Pool
STD	Standard Fuel Assembly
TD	Theoretical Density
V+	Vantage +
V5H	Vantage 5-Hybrid
WABA	Wet Annular Burnable Absorber
Westinghouse	Westinghouse Electric Company LLC
wt%	Weight Percent
Zirlo	ZIRLO®

1 INTRODUCTION

The purpose of this report is to document the criticality safety analysis performed to support the operation of Comanche Peak Nuclear Power Plant (Comanche Peak) at 3612 Megawatts-Thermal (MWt) and to update the methodology used to develop the burnup requirements for storage of fuel in the Spent Fuel Pool (SFP).

2 OVERVIEW

The existing SFP storage racks are evaluated for the placement of fuel within the storage arrays described in Section 5.2. Credit is taken for the negative reactivity associated with burnup and post-irradiation cooling time (decay time). Additionally, credit is taken for the presence of soluble boron in the SFP and for the presence of neutron poison integral to the Region I spent fuel racks.

2.1 ACCEPTANCE CRITERIA

The objective of this SFP criticality safety analysis is to ensure that the SFP operates within the bounds of Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements" (Reference 1) discussed here.

1. The effective neutron multiplication factor (k_{eff}) of all permissible storage arrangements at a soluble boron concentration of 0 parts per million (ppm) shall be less than 1.0 including margin for all applicable biases and uncertainties with 95% probability at a 95% confidence level.

2. The k_{eff} of all permissible storage arrangements when crediting soluble boron shall yield results less than or equal to 0.95 including margin for all applicable biases and uncertainties with 95% probability at a 95% confidence level.

3. The k_{eff} when crediting soluble boron shall be less than 0.95 under all postulated accident conditions, including margin for all applicable biases and uncertainties with 95% probability at a 95% confidence level.

2.2 DESIGN APPROACH

This document assures compliance with the acceptance criteria in Section 2.1 by establishing limits on the minimum allowable burnup as a function of enrichment and decay time for each fuel storage array. A conservative combination of best estimate and bounding values have been selected to model the fuel in this analysis to ensure that fuel represented by the proposed Comanche Peak Technical Specifications is less reactive than the fuel modeled in this analysis. Therefore, burnup limits generated here will conservatively bound all fuel to be stored in the Comanche Peak SFP. Input selection is discussed in Section 4 of this analysis.

To proactively address potential future fuel design and operational changes, the design parameters which need to be confirmed are provided in Section 6.2. SFP operation restrictions are provided in Section 6.3. Reviewing Section 6.2 on a reload basis assures that future fuel designs and operation changes are reviewed with respect to the potential impacts on the storage requirements provided in this document.

The acceptability of the storage arrays developed in this analysis is ensured by controlling the assemblies that can be stored in each array. In this analysis, assemblies are divided into two Groups (Group F1 and F2) based on fuel rod outer diameter. Assemblies are then further divided into Fuel Categories 1 through

6 based on assembly reactivity determined as a function of assembly average burnup, initial enrichment[1], and decay time. An assembly's fuel category determines which storage arrays are acceptable for the assembly in question. Fuel Category 1 defines the most reactive assemblies, i.e. representing a fresh 5.0 Weight Percent (wt%) ^{235}U Group F2 assembly and Fuel Category 6 defines the least reactive assemblies, i.e. representing assemblies that can be stored in Array II-A (see Section 5.2).

Table 2-1 Fuel Categories Ranked by Reactivity

Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	
Fuel Category 5	
Fuel Category 6	Low Reactivity

Notes:

1. Fuel categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
2. For Group F1, Fuel Category 1 and 2 are fuel up to 3.5 wt% ^{235}U; no burnup is required.
3. For Group F1, Fuel Categories 3 and 5 are omitted.
4. For Group F1, Fuel Categories 4 and 6 are determined from the coefficients provided in Section 6.1.
5. For Group F2, Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
6. For Group F2, Fuel Categories 2 through 6 are determined from the coefficients provided in Section 6.1.

2.3 COMPUTER CODES

The analysis methodology employs the following computer codes and cross-section libraries: (1) the two-dimensional (2-D) transport lattice code PARAGON, as documented in WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (Reference 2), and its cross-section library based on Evaluated Nuclear Data File Version VI.3 (ENDF/B-VI.3), and (2) SCALE Version 5.1, as documented in ORNL/TM-2005/39, "SCALE: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation" (Reference 3), with the 44-group cross-section library based on ENDF/B-V.

2.3.1 Two-Dimensional Transport Code PARAGON

PARAGON is used for simulation of in-reactor fuel assembly depletion to generate isotopics for burnup credit applications in the SFP.

[1] Initial Enrichment is the nominal ^{235}U enrichment of the central zone region of fuel, excluding axial blankets, prior to reduction in ^{235}U content due to fuel depletion. If the fuel assembly contains axial regions of different ^{235}U enrichment values, such as axial blankets, the maximum Initial Enrichment value is to be utilized.

PARAGON is Westinghouse Electric Company LLC's (Westinghouse) state-of-the-art 2-D lattice transport code. It is part of Westinghouse's core design package and provides lattice cell data for three-dimensional (3-D) core simulator codes. This data includes macroscopic cross-sections, microscopic cross-sections for feedback adjustments, pin factors for pin power reconstruction calculations, and discontinuity factors for a 3-D nodal method solution of the diffusion equation. PARAGON uses the collision probability theory within the interface current method to solve the integral transport equation. Throughout the calculation, PARAGON uses the exact heterogeneous geometry of the assembly and the same energy groups as in the cross-section library to compute the multi-group fluxes for each micro-region location of the assembly. In order to generate the multi-group data, PARAGON goes through four steps of calculations: resonance self-shielding, flux solution, burnup calculation and homogenization. The 70-group PARAGON cross-section library is based on the ENDF/B-VI.3 basic nuclear data. It includes explicit multigroup cross-sections and other nuclear data for 174 isotopes, without any lumped fission products or pseudo cross-sections. PARAGON and its 70-group cross-section library are benchmarked, qualified, and licensed both as a standalone transport code and as a nuclear data source for a core simulator in a complete nuclear design code system for core design, safety, and operational calculations.

PARAGON is generically approved for depletion calculations (Reference 2). PARAGON has been chosen for this spent fuel criticality analysis because it has all the attributes needed for burnup credit applications. There are no Safety Evaluation Report limitations for the use of PARAGON in UO_2 criticality analysis.

2.3.1.1 PARAGON Cross-Section Library

The current PARAGON cross-section library uses ENDF/B as the basic evaluated nuclear data files. Currently the library has 70 neutron energy groups and 48 gamma energy groups. This library has been generated using the NJOY code documented in "NJOY91.118: A Code System For Producing Pointwise And Multigroup Neutron and Photon Cross Sections From EndF/B Evaluated Nuclear Data" (Reference 4). To account for the resonance self-shielding effect, the group cross-sections are tabulated as a function of both temperature and background scattering cross-section (dilution). The resonance self-shielding module of the code uses these resonance self-shielding tables to compute the isotopic self-shielded cross-section in the real heterogeneous situation. The library contains energy group cross-sections and transport-corrected P_0 scattering matrices as a function of temperature. The P_0 scattering matrices contain diagonal corrections for anisotropic scattering. The library also has temperature-dependent P_1 scattering matrices for all major moderator materials.

2.3.2 SCALE Code Package

The SCALE system was developed for the NRC to standardize the method of analysis for evaluation of nuclear fuel facilities and shipping package designs. In this SFP criticality analysis, the SCALE code package is used to calculate the reactivity of fissile systems in SFP conditions. Specifically, the SCALE package is used to analyze infinite arrays for all storage arrays in SFP, as well as finite rack module and SFP representations to evaluate interfaces, soluble boron requirements, and postulated accident scenarios.

The SCALE package includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-III (NITAWL), and KENO V.a (KENO). A brief description of each module is provided below:

- The BONAMI module utilizes Dancoff approximations to perform Bondarenko unresolved resonance self-shielding calculations. BONAMI solves problems in a one-dimensional multizone slab, cylindrical, or spherical geometry. Heterogeneous effects are accounted for through the use of the Dancoff factors evaluated for the geometry of the problem, defined as separate regions of fuel, cladding and moderator.
- The NITAWL module performs problem-dependent resonance shielding by applying the Nordheim Integral Treatment. The infinite-dilute multi-group cross-sections are adjusted by a correction value determined by NITAWL. The correction is calculated by first determining the infinite dilute contribution of each resonance to the group cross-section and then by calculating what the contribution would be if the resonance was shielded for the specific problem. The geometry type, materials, characteristic dimensions, and Dancoff factor are all passed to NITAWL for determining the details of the approximations used to self-shield the cross-sections. Additionally, NITAWL produces a problem-dependent working cross-section library that is used in the subsequent multigroup Monte Carlo simulations.
- The KENO module is a multigroup Monte Carlo criticality program used to calculate the k_{eff} of 3-D models. Flexible geometry features and the availability of various boundary condition prescriptions in KENO allow for accurate and detailed modeling of fuel assemblies in storage racks, either as infinite arrays or in actual SFP models. Anisotropic scattering is treated by using discrete scattering angles through the use of P_n Legendre polynomials. KENO uses problem-specific cross-section libraries, processed for resonance self-shielding and for the thermal characteristics of the problem.

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]^{a,c} The details of the validation are found in Appendix A. The validation shows that SCALE 5.1 is an accurate tool for calculation of k_{eff} for SFP applications. The benchmark calculations utilize the same computer platform and cross-section libraries as are used for the design basis calculations. [

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2.3.2.1 SCALE 44-Group Cross-Section Library

The 44-group ENDF/B-V library included in the SCALE package was developed for use in the analysis of fresh and spent fuel and radioactive waste systems. Collapsed from the finegroup 238-group ENDF/B-V cross-section library, this broad-group library contains all nuclides (more than 300) from the ENDF/B-V data files. Broad-group boundaries (22 fast and 22 thermal energy groups) were chosen as a subset of the parent 238-group ENDF/B-V boundaries, emphasizing the key spectral aspects of a typical Light Water Reactor (LWR) fuel package. The finegroup 238-group ENDF/B-V cross-sections were collapsed into this broad-group structure using a fuel-cell spectrum calculated based on a 17 × 17 Westinghouse Pressurized Water Reactor (PWR) assembly. Thus, the 44-group library performs well for LWR lattices

and is the SCALE library used here for criticality safety analysis of arrays of light-water-reactor-type fuel assemblies, as would be encountered in fresh or spent fuel transportation or storage environments as discussed in NUREG/CR-6686, "Experience With the SCALE Criticality Safety Cross-Section Libraries" (Reference 5).

3 COMANCHE PEAK NUCLEAR POWER PLANT

This section describes the physical characteristics of Comanche Peak that are important to SFP criticality safety. The reactor and its associated fuel designs and fuel management history are discussed in Section 3.1 and the physical characteristics of the SFP are discussed in Section 3.2.

3.1 REACTOR DESCRIPTION

Comanche Peak is a 4-loop Westinghouse PWR utilizing fuel with a 17x17 lattice. Comanche Peak has operated at licensed powers from 3411 MWt to 3612 MWt, using multiple fuel designs from several fuel vendors. All fuel assemblies used at Comanche Peak incorporate a 17x17 square array of 264 fuel rods with 24 control rod guide tubes (GT) and one instrumentation tube (IT). The fuel rod cladding material is Zircaloy and its variants such as ZIRLO® High Performance Fuel Cladding Material. Each fuel rod contains a column of enriched UO_2 fuel pellets. The pellets are pressed and sintered, and are dished on both ends.

Table 3-1 provides basic data on the type of reactor and the fuel types that comprise Comanche Peak. Section 3.1.1 describes the different fuel designs outlined in Table 3-1 including outlining neutronically important physical characteristics of each fuel design. Section 3.1.2 describes the non-mechanical fuel features which further subdivide mechanically identical fuel designs. Additionally, the fuel management history for both Comanche Peak units is provided in Table 3-4 and Table 3-5.

Table 3-1 Reactor General Specifications

Parameter	Value
Reactor type	Westinghouse 4-Loop PWR
Reactor power (MWt)	3411 - 3612
Fuel lattice	17x17
Fuel design 1	Westinghouse Standard Fuel (STD)
Fuel design 2	Westinghouse Vantage 5-Hybrid (V5H)
Fuel design 3	Siemens/Framatom ANP (ANP)
Fuel design 4	Westinghouse Optimized Fuel Assembly (OFA)
Fuel design 5	Westinghouse Vantage + (V+)

Notes:
1. Comanche Peak's current licensed operating power is 3612 MWt.

3.1.1 Fuel Designs

This section outlines the neutronically important mechanical features of the five (5) fuel designs outlined in Table 3-1. The fuel designs are divided into two groups. The first group (Group F1) consists of fuel designs which incorporate a nominal rod diameter of 0.374 inches. The second group (Group F2) consists of fuel designs which incorporate a nominal rod diameter of 0.360 inches.

Table 3-2 Group F1 Fuel Assembly Mechanical Specifications		
Parameter	**Value**	
Assembly type	STD	V5H
Rod array size	17x17	17x17
Rod pitch, inch	0.496	0.496
Active fuel length, inch	144	144
Total number of fuel rods	264	264
Fuel cladding Outer Dimension (OD), inch	0.374	0.374
Fuel cladding Inner Dimension (ID), inch	0.329	0.329
Fuel cladding thickness, inch	0.0225	0.0225
Pellet diameter, inch	0.3225	0.3225
Number of guide/instrument tubes	24 / 1	24 / 1
Guide/instrument tube OD, inch	0.4820 / 0.4820	0.4740 / 0.4740
Guide/instrument tube thickness, inch	0.0160 / 0.0160	0.0160 / 0.0160

As shown in Table 3-2, the neutronically important mechanical properties of the STD and V5H fuel designs are essentially identical. Therefore, STD and V5H can be treated as a single fuel design with only deviations in non-mechanical fuel features being considered when determining the limiting fuel features to be analyzed when developing isotopics for Group F1.

Table 3-3 Group F2 Fuel Assembly Mechanical Specifications			
Parameter	**Value**		
Assembly type	OFA	V+	ANP
Rod array size	17x17	17x17	17x17
Rod pitch, (in)	0.496	0.496	0.496
Active fuel length, (in)	144	144	144
Total number of fuel rods	264	264	264
Fuel cladding OD, (in)	0.360	0.360	0.360
Fuel cladding ID, (in)	0.315	0.315	0.310
Fuel cladding thickness, (in)	0.0225	0.0225	0.025
Pellet diameter, (in)	0.3088	0.3088	0.3035
Number of guide/instrument tubes	24 / 1	24 / 1	24 / 1
Guide/instrument tube OD, (in)	0.4740 / 0.4740	0.4740 / 0.4740	0.4800 / 0.4800
Guide/instrument tube thickness, (in)	0.0160 / 0.0160	0.0160 / 0.0160	0.0160 / 0.0160

As shown in Table 3-3, the neutronically important mechanical properties of the OFA and V+ fuel designs are identical but the ANP fuel design is different. Therefore, the OFA and V+ fuel designs can be considered a single design, only considering deviations in the non-mechanical fuel features for OFA and V+. The ANP design is considered separately when determining the limiting fuel features to be analyzed when developing isotopics for Group F2.

3.1.2 Fuel Management History

This section discusses the non-mechanical fuel features which are important to criticality safety and how they impact the number of distinct fuel designs to be considered in this analysis. Specifically, the presence and enrichment of axial blankets are taken into account, as well as the presence and amount of Burnable Absorber (BA) being used such as Integral Fuel Burnable Absorber (IFBA) or Wet Annular Burnable Absorber (WABA). Table 3-4 and Table 3-5 provide the fuel management history for Comanche Peak Units 1 and 2 respectively. This table provides information on the mechanical fuel design and the fuel features of the assemblies that are loaded fresh for each cycle.

Table 3-4 Comanche Peak Unit 1 Fuel Management History

Cycle	Power (MWt)	Fuel Design	Blankets	Blanket Enrich	Blanket Length (in)	BA Type	Max. BA Loading
1	3411	STD	No	N/A	N/A	Pyrex	24-Finger
2	3411	STD	No	N/A	N/A	WABA	12-Finger
3	3411	V5H	Yes – Solid	Natural	6	WABA	12-Finger
4	3411	ANP	Yes – Solid	Natural	6	Solid B_4C	20-Finger
5	3411	ANP	Yes – Solid	Natural	6	Solid B_4C	24-Finger
6	3411	ANP	Yes – Solid	Natural	6	Solid B_4C	24-Finger
7	3411	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
8	3411	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
9	3411	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
10	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
11	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
12	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
13	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
14	3612	V+	Yes – Mixed	2.6	6	IFBA/WABA	128-Pin/16-Finger
15	3612	V+	Yes – Mixed	2.6	6	IFBA/WABA	128-Pin/16-Finger
16	3612	V+	Yes – Mixed	2.6	6	IFBA/WABA	128-Pin/16-Finger

Notes:

1. For the mixed blankets, the solid blankets are present in fuel pins without IFBA. Annular blankets are present in fuel pins with IFBA.

Table 3-5 Comanche Peak Unit 2 Fuel Management History

Unit 2 Fuel Management							
Cycle	**Power (MWt)**	**Fuel Design**	**Blankets**	**Blanket Enrich**	**Blanket Length (in)**	**BA Type**	**Max. BA Loading**
1	3411	OFA	No	N/A	N/A	WABA	20-Finger
2	3411	OFA	Yes – Solid	Natural	6	WABA	24-Finger
3	3411	ANP	Yes – Solid	Natural	6	Solid B_4C	24-Finger
4	3411	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
5	3445	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
		OFA	Yes – Solid	Natural	6	None	N/A
6	3458	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
7	3458	ANP	Yes – Solid	2.0	6	Solid B_4C	24-Finger
		V+	Yes – Solid	2.0	6	WABA	24-Finger
8	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
9	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
10	3458	V+	Yes – Annular	2.6	6	IFBA/WABA	128-Pin/16-Finger
11	3458	V+	Yes – Annular	2.6	6	IFBA	156-Pin
12	3612	V+	Yes – Mixed	2.6	6	IFBA	156-Pin
13	3612	V+	Yes – Mixed	2.6	6	IFBA	156-Pin

Notes:

1. For the mixed blankets, the solid blankets are present in fuel pins without IFBA. Annular blankets are present in fuel pins with IFBA.
2. A combination of 80-Pin/24-Fingers was used in Unit 2 Cycle 8. [ ]^{a,c}

As shown in Table 3-4 and Table 3-5, there are several different combinations of fuel design, blankets and maximum BAs that need to be considered in performing the analysis. These different fuel types are outlined for Group F1 in Table 3-6 and for Group F2 in Table 3-7.

Table 3-6 Group F1 Fuel Design Parameters a,c

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Table 3-7 Group F2 Fuel Design Parameters a,c

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3.2 SPENT FUEL POOL DESCRIPTION

Comanche Peak has two SFPs. The physical characteristics of these SFPs are described in this section. Each SFP has two different storage regions (Region I and Region II). These regions are differentiated by the spent fuel storage rack designs. Each SFP is surrounded by a concrete wall at least 24 inches in thickness with a $3/16^{th}$ inch stainless steel liner. The SFPs are connected by a transfer canal, allowing fuel from either unit to be stored in either SFP. Diagrams of the Unit 1 & 2 SFPs are provided in Figure 3-1 and Figure 3-2 respectively.

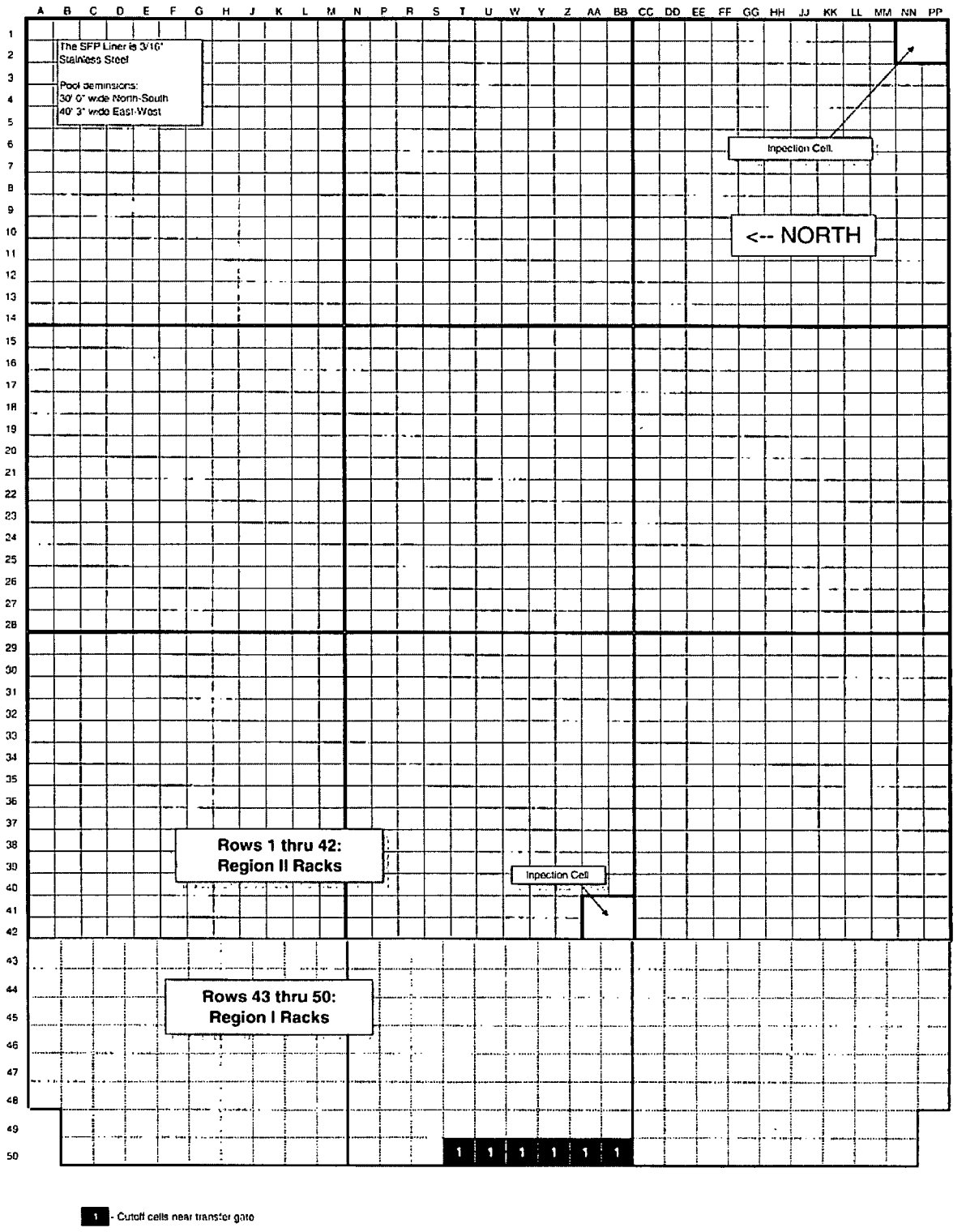

Figure 3-1 Comanche Peak Unit 1 SFP

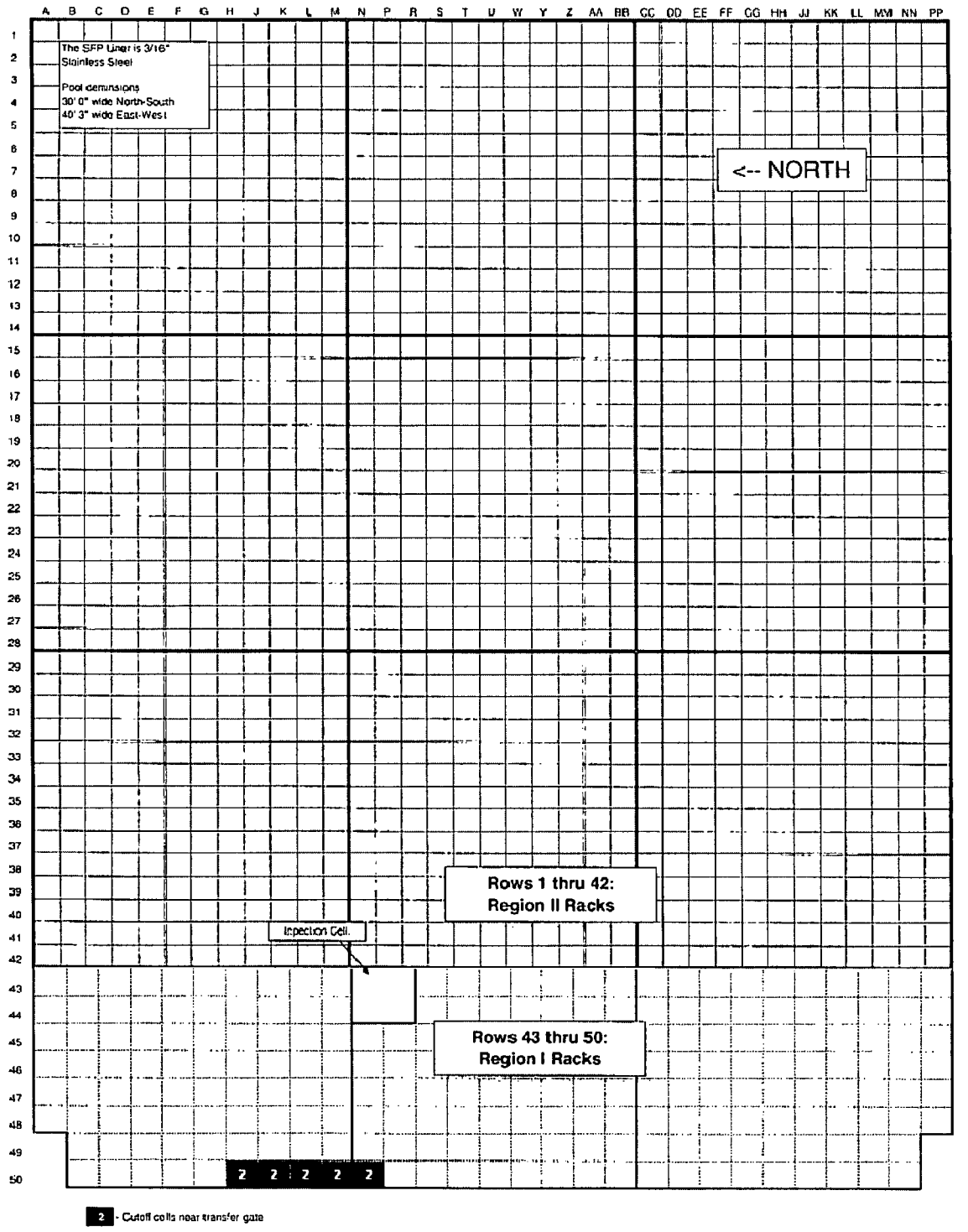

Figure 3-2 Comanche Peak Unit 2 SFP

3.2.1 Region I Fuel Rack Description

The first region in each SFP (Region I) contains storage racks designed to hold fresh fuel of up to 5 wt% ^{235}U. The Region I racks are based on a "flux trap" design which uses two BORAL® (BORAL) panels between each assembly to reduce the neutronic interaction between adjacent assemblies. The specifications for these racks are given in Table 3-8.

Table 3-8 Region I Fuel Storage Rack Specifications

Parameter	Value	Tolerance
Rack cell pitch (N-S/E-W) (in)	11.05 / 10.65	[ ]^{a,c}
Cell inner dimension (in)	8.8	[ ]^{a,c}
Cell wall thickness (in)	0.075	[ ]^{a,c}
Flux trap gap width (N-S/E-W) (in)	1.83 / 1.43	[ ]^{a,c}
Neutron absorber type	BORAL	N/A
Neutron absorber loading (g/cm^2)	[ ]^{a,c}	[ ]^{a,c}
Neutron absorber thickness (in)	0.101	N/A
Neutron absorber cavity width (in)	7.5	[ ]^{a,c}
Neutron absorber cavity height (in)	147	[ ]^{a,c}
Neutron absorber wrapper thickness (in)	0.0235	[ ]^{a,c}
Min./Max. distance to SFP wall (in)	5.75 / 19.45	N/A

Notes:
1. The neutron absorber thickness tolerance is accounted for in the areal density treatment of the BORAL.

3.2.2 Region II Fuel Rack Description

The second region in each SFP (Region II) contains storage racks designed to maximize the number of storage cells available (minimize storage cell pitch). The Region II racks are based on a high density rack design which was intended to use one Boraflex panel between each assembly to reduce the neutronic interaction of adjacent assemblies. However, because the issues with Boraflex degradation were discovered before Region II racks were installed at Comanche Peak, the Boraflex panels have been removed from the Region II racks. In the SFP of one unit, the Boraflex panels were removed from the wrappers, but in the other unit's SFP, the wrappers were never installed on the fuel racks. The specifications for the racks which include the Boraflex wrapper are given in Table 3-9. The specifications for the racks without the wrapper are given in Table 3-10.

Table 3-9 Region II Fuel Storage Rack Specifications –With Boraflex Wrapper		
Parameter	**Value**	**Tolerance**
Rack cell pitch (in)	9.0	± 0.06
Cell inner dimension (in)	8.83	± 0.05
Cell wall thickness (in)	0.075	± 0.004
Flux trap gap width (in)	N/A	N/A
Neutron absorber type	N/A	N/A
Neutron absorber loading (g/cm^2)	N/A	N/A
Neutron absorber cavity width (in)	N/A	N/A
Neutron absorber cavity height (in)	N/A	N/A
Neutron absorber wrapper thickness (in)	0.020	N/A
Min./Max. distance to SFP wall (in)	5.75 / 19.32	N/A

Table 3-10 Region II Fuel Storage Rack Specifications – No Boraflex Wrapper		
Parameter	**Value**	**Tolerance**
Rack cell pitch (in)	9.0	± 0.06
Cell inner dimension (in)	8.83	± 0.05
Cell wall thickness (in)	0.075	± 0.004
Flux trap gap width (in)	N/A	N/A
Neutron absorber type	N/A	N/A
Neutron absorber loading (g/cm^2)	N/A	N/A
Neutron absorber cavity width (in)	N/A	N/A
Neutron absorber cavity height (in)	N/A	N/A
Neutron absorber wrapper thickness (in)	N/A	N/A
Min./Max. distance to SFP wall (in)	5.75 / 19.32	N/A

As shown above, the only difference between the two separate sets of Region II racks is the presence of the Boraflex wrappers. Because the wrappers displace moderator and are made of stainless steel which absorbs neutrons, this analysis conservatively ignores the presence of the Boraflex wrappers.

4 DEPLETION ANALYSIS

The depletion analysis is a vital part of any SFP criticality analysis which uses burnup credit. The isotopic inventory of the fuel as a function of burnup is generated through the depletion analysis and therefore the inputs used need to be carefully considered. This section describes the methods used to determine the appropriate inputs for the generation of isotopic number densities and the Monte Carlo simulations to conservatively bound fuel depletion and storage at Comanche Peak. [

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4.1 DEPLETION MODELING ASSUMPTIONS

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4.2 FUEL DEPLETION PARAMETER SELECTION

4.2.1 Fuel Isotopic Generation

This section outlines how parameters are selected for use in the fuel depletion calculations to generate isotopic number densities. For the purposes of this analysis, the isotopic number densities generated are differentiated by fuel enrichment, decay time after discharge, and whether the fuel was categorized as Group F1 or Group F2.

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Based on Comanche Peak fuel management, the Group F1 fuel has isotopic number densities which are calculated at enrichments of 1.6, 2.4, and 3.5 wt% ^{235}U and at decay times of 10, 15, and 20 years. Isotopic number densities have been generated for Group F2 fuel at enrichments of 3.0, 4.0, and 5.0 wt% ^{235}U and decay times of 0, 5, 10, 15, and 20 years. [

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4.2.2 Reactor Operation Parameters

The reactivity of the depleted fuel in the SFP is determined by the in-reactor depletion conditions. The conditions experienced in the reactor impact the isotopic composition of fuel being discharged to the SFP. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel" (Reference 6) provides discussion on the core operation parameters important to SFP criticality. This section outlines the parameters used in generating the fuel isotopics and why they are appropriate for use in this analysis. The operating conditions of Group F1 and Group F2 fuel are provided in Table 4-7 and Table 4-10 respectively. Each table provides both the nominal values and the values assumed in the analysis.

4.2.2.1 Soluble Boron Concentration

The soluble boron concentration in the reactor during operation impacts the reactivity of fuel being discharged to the SFP. Because boron is a strong thermal neutron absorber its presence hardens the neutron energy spectrum in the core, creating more plutonium. To ensure this impact is adequately accounted for in the isotopic generation it is important to account for the presence of soluble boron during the reactor operation.

Based on guidance from Reference 6, a constant cycle average soluble boron concentration which assumes 19.9 at% ^{10}B (Equation 4-1) may be modeled in place of a soluble boron letdown curve. [

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Table 4-1 Cycle Average Soluble Boron Concentrations			
Unit 1	**Soluble Boron (ppm)**	**Unit 2**	**Soluble Boron (ppm)**
Cycle 1	497.3	Cycle 1	474.8
Cycle 2	529.4	Cycle 2	612.1
Cycle 3	509.6	Cycle 3	688.4
Cycle 4	615.9	Cycle 4	702.3
Cycle 5	707.5	Cycle 5	703.2
Cycle 6	694.1	Cycle 6	669.2
Cycle 7	686.3	Cycle 7	741.7
Cycle 8	643.3	Cycle 8	752.2
Cycle 9	671.2	Cycle 9	770.9
Cycle 10	849.9	Cycle 10	743.8
Cycle 11	794.6	Cycle 11	856.0
Cycle 12	698.1	Cycle 12	843.2
Cycle 13	769.6	Cycle 13	868.3
Cycle 14	795.5		
Cycle 15	806.2		
Cycle 16	810.8		

4.2.2.2 Fuel Temperature

The fuel temperature during operation impacts the reactivity of fuel being discharged to the SFP. Increasing fuel temperature increases resonance absorption in ^{238}U due to Doppler broadening which leads to increased plutonium production, increasing the reactivity of the fuel. [

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4.2.2.3 Operating History and Specific Power

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]a,c Interim Staff Guidance (ISG) DSS-ISG-2010-1, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools" (Reference 8) states:

> "It may be physically impossible for the fuel assembly to simultaneously experience two bounding values (i.e., the moderator temperature associated with the "hot channel" fuel assembly and the minimum specific power). In those cases, the application should maximize the dominate parameter and use the nominal value for the subordinate parameter."

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4.2.2.4 Maximum Average Assembly Power

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4.2.3 Axial Profile Selection

This section discusses the selection of bounding axial burnup and moderator temperature profiles. [

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4.2.3.1 Axial Burnup Profile Selection

This section describes the methods used to determine the limiting distributed axial burnup profiles. These profiles will be used along with the uniform axial burnup profile to calculate the burnup limits provided in Section 6.1.

As discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses" (Reference 9), as fuel is operated in the reactor, the center of each assembly generates more power than the ends. This leads to the burnup of each assembly varying along its length. Since the middle of each assembly generates most of the power, the burnup in the middle of the assembly is greater than the assembly average. At the same time, the ends of the assembly are less burned than the assembly average. When the burnup difference between the middle and end of an assembly is large enough, reactivity becomes driven by the end of the assembly rather than the middle, as the under depletion of the ends overcomes the reactivity loss due to the neutron leakage.

Without considering neutron leakage, a uniformly enriched assembly's reactivity will be driven by the lowest burnup section of the fuel. At the beginning of life, there is no axial burnup variation (fresh fuel) and thus the fuel is axially isoreactive unless neutron leakage is considered. When neutron leakage is considered, the axial location which has the least leakage will be most reactive, thus the center of the assembly drives reactivity. As the burnup of the assembly becomes more distributed, the impact of neutron leakage decreases and the assembly's reactivity is increasingly driven by the under-depleted ends of the fuel. Therefore, as a general rule, with increasing burnup the axial location driving assembly reactivity moves from the center towards the top of an assembly.

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4.2.3.2 Axial Moderator Temperature Profile Selection

This section describes the methods used to determine the limiting axial moderator temperature profiles. These profiles will be used together with axial distributed and uniform burnup profiles to calculate the isotopics used in generating the burnup limits provided in Section 6.1.

Selecting an appropriate moderator temperature profile is important as it impacts the moderator density and therefore the neutron spectrum during depletion as discussed in Reference 6. An appropriate moderator temperature ensures the impact of moderator density on the neutron spectral effects is bounded, conservatively biasing the isotopic inventory of the fuel.

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4.2.4 Burnable Absorber Usage

Burnable absorber usage at Comanche Peak has been considered for this analysis and conservative assumptions have been utilized to bound the effects of BAs on fuel isotopics. The BAs that have been used at Comanche Peak include both discrete and integral BAs and the history of BA usage is available in Table 3-4 and Table 3-5.

The discrete BAs which have been used at Comanche Peak are Pyrex, Solid B_4C rods, and WABA. Discrete BAs can impact assembly reactivity after discharge. These inserts contain boron, which is a strong thermal neutron absorber, hardening the neutron spectrum of the assembly where the insert resides. Additionally, these inserts displace moderator that would normally be in the guide tubes of an assembly. The moderator displacement also acts to harden the neutron spectrum of the fuel assembly. These two phenomena cause increased plutonium production in the assembly, leading to the assembly being more reactive when discharged to the SFP. The insert parameters are shown in Table 4-2.

Table 4-2 Discrete Insert Specifications

Parameter	Pyrex	Solid B_4C	WABA
BA material	Borosilicate Glass	Al_2O_3-B_4C	Al_2O_3-B_4C
BA ID, (in)	0.1900	N/A	0.2780
BA OD, (in)	0.3360	0.304	0.3180
BA clad material	SS	Zircaloy-4	Zircaloy
BA inner clad thickness, (in)	0.0070	N/A	0.0210
BA inner clad OD, (in)	0.1810	N/A	0.2670
BA outer clad thickness, (in)	0.0185	0.026	0.0260
BA outer clad OD, (in)	0.3810	0.381	0.3810
BA length, (in)	144	120 – 132	120-144
BA Insertion, (GWd/MTU)	15.650	29.073	30.685

In addition to discrete absorbers used at Comanche Peak, the integral absorber IFBA has been used. As with the discrete absorbers, IFBA contains boron which causes increased plutonium production in the assembly, leading to the assembly being more reactive when discharged to the SFP. The IFBA parameters are shown in Table 4-3.

Table 4-3 IFBA Specifications	
Parameter	**IFBA**
BA material	ZrB_2
BA ^{10}B loading, mg/in	[ ]^{a,c}
BA Thickness, mils	[ ]^{a,c}
BA length, (in)	120-132

The impact of BAs on fuel design selection is discussed in Section 4.3.

4.3 FUEL DESIGN SELECTION

To develop conservative storage requirements at Comanche Peak, the different fuel designs and the conditions in which those designs were operated needs to be considered. All of the fuel designs to be considered for Comanche Peak are discussed in Section 3.1.1.

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4.3.1 Group F1 Fuel Design Selection

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Table 4-4 Group F1 Design Basis Parameters a,c

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4.3.2 Group F2 Fuel Design Selection

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Table 4-5 Group F2 Fuel Design Parameters a,c

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Table 4-6 [ ]^{a,c} a,c

4.4 FINAL DEPLETION PARAMETERS

This section outlines the parameters used in the final depletion calculations for both Group F1 and Group F2. The depletion parameters outlined in this section are:

- Core Operation Parameters
- Fuel Assembly Dimensions
- Axial Burnup and Moderator Temperature Profiles

4.4.1 Group F1 Final Depletion Parameters

The fuel isotopics used in the Group F1 reactivity calculations were generated based on the data presented in Table 4-7 through Table 4-9.

Table 4-7 Parameters Used in Depletion Analysis for Group F1		
Parameter	**Nominal Values**	**Depletion Analysis**
Maximum cycle average soluble boron concentration, ppm	707.5	[ ]^{a,c}
Rated thermal power, MWt	3411	[ ]^{a,c}
Average assembly power, MWt	17.68	[ ]^{a,c}
Core outlet moderator temperature, °F	622.1	[ ]^{a,c}
Core inlet moderator temperature, °F	559.6	[ ]^{a,c}
Minimum RCS flow rate (Thermal Design Flow), gpm	101,000	[ ]^{a,c}
Fuel designs	Westinghouse STD, V5H	[ ]^{a,c}
Fuel assembly blanket	No Blankets, Natural Blankets	[ ]^{a,c}
Fuel density, % TD	95.5	[ ]^{a,c}
BA	Pyrex, WABA	[ ]^{a,c}
BA Lengths (in)	144, 120-144	[ ]^{a,c}

Table 4-8 Group F1 Fuel Assembly Mechanical Specifications	
Parameter	**Value**
Assembly type	STD
Rod array size	17x17
Rod pitch, inch	0.496
Active fuel length, (in)	144
Total number of fuel rods	264
Fuel cladding OD, (in)	0.374
Fuel cladding ID, (in)	0.329
Fuel cladding thickness, (in)	0.0225
Pellet diameter, (in)	0.3225
Number of guide/instrument tubes	24 / 1
Guide/instrument tube OD, (in)	0.4820 / 0.4820
Guide/instrument tube thickness, (in)	0.0160 / 0.0160

Table 4-9 Axial Burnup and Temperature Profiles for Group F1 a,c

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4.4.2 Group F2 Final Depletion Parameters

The fuel isotopics used in the Group F2 reactivity calculations were generated based on the data presented in Table 4-10 through Table 4-12.

Table 4-10 Parameters Used in Depletion Analysis for Group F2		
Parameter	**Nominal Values**	**Depletion Analysis**
Maximum cycle average soluble boron concentration, ppm	868.3	[ ]^{a,c}
Rated thermal power, MWt	≤ 3612	[ ]^{a,c}
Average assembly power, MWt	18.72	[ ]^{a,c}
Core outlet moderator temperature, °F	623.8	[ ]^{a,c},
Core inlet moderator temperature, °F	558.0	[ ]^{a,c}
Minimum RCS flow rate (Thermal Design Flow), gpm/loop	95700	[ ]^{a,c}
Fuel designs	OFA, V+, ANP	[ ]^{a,c}
Fuel assembly blanket	No Blankets, Natural Blankets, 2.0 wt% ^{235}U, 2.6 wt% ^{235}U	[ ]^{a,c}
Fuel density, % TD	95.5	[ ]^{a,c}
BA	IFBA, WABA, Solid B_4C	[ ]^{a,c}
BA Lengths (in)	120 – 132, 120 – 144, 120 – 132	[ ]^{a,c}

Notes:

1. The 2.6 wt% ^{235}U fuel blankets are constructed either entirely of annular pellets or a mix with annular pellets above IFBA bearing fuel rods and solid pellets above the non-IFBA rods.

Table 4-11 Group F2 Fuel Assembly Mechanical Specifications	
Parameter	**Value**
Assembly type	OFA
Rod array size	17x17
Rod pitch, inch	0.496
Active fuel length, (in)	144
Total number of fuel rods	264
Fuel cladding OD, (in)	0.360
Fuel cladding ID, (in)	0.315
Fuel cladding thickness, (in)	0.0225
Pellet diameter, (in)	0.3088
Number of guide/instrument tubes	24 / 1
Guide/instrument tube OD, inch	0.4740 / 0.4740
Guide/instrument tube thickness, inch	0.0160 / 0.0160

Table 4-12 Limiting Axial Burnup and Temperature Profiles for Group F2

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5 CRITICALITY ANALYSIS

This section describes the reactivity calculations and evaluations performed in developing the burnup requirements for fuel storage at Comanche Peak and confirming continued safe SFP operation during both normal and accident conditions.

5.1 KENO MODELING ASSUMPTIONS

As discussed in Section 2.3.2, KENO is the criticality code used to support this analysis. KENO is used to determine the absolute reactivity of burned and fresh fuel assemblies loaded in storage arrays.
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5.1.1 Description of Fuel Assembly and Storage Rack for KENO

This section outlines the dimensions and tolerances of the design basis fuel assemblies for Group F1 and Group F2 and the fuel storage racks. These dimensions and tolerances are the basis for the KENO models used to determine the burnup requirements for each fuel storage array and to confirm the safe operation of the SFP under normal and accident conditions.

5.1.1.1 Fuel Assembly Dimensions and Tolerances

This section provides the dimensions and tolerances for the design basis fuel for Group F1 and Group F2. Selection of these fuel designs is discussed in Section 4.3.

Table 5-1 Group F1 Design Basis Fuel Assembly Design Specifications

Parameter	Value (Group F1)	Value (Analyzed)
Assembly type	17x17 STD	17x17 STD
Rod array size	17x17	17x17
Rod pitch, (in)	0.496 [ ]^{a,c}	0.496 [ ]^{a,c}
Active fuel length, (in)	144	144
Stack density, (% TD)	[ ]^{a,c}	[ ]^{a,c}
Maximum enrichment, (wt% ^{235}U)	4.95	5.0
Enrichment tolerance, (wt% ^{235}U)	[ ]^{a,c}	[ ]^{a,c}
Total number of fuel rods	264	264
Fuel cladding OD, (in)	0.374 [ ]^{a,c}	0.374 [ ]^{a,c}
Fuel cladding thickness, (in)	0.0225 [ ]^{a,c}	0.0225 [ ]^{a,c}
Pellet diameter, (in)	0.3225 [ ]^{a,c}	0.3225 [ ]^{a,c}
Number of guide/instrument tubes	24 / 1	24 / 1
Instrument tube OD, (in)	0.4820 [ ]^{a,c}	0.4820 [ ]^{a,c}
Guide tube OD, (in)	0.4820 [ ]^{a,c}	0.4820 [ ]^{a,c}

Notes:
1. All Group F1 fuel has a TD of less than 96.05 %TD at 95/95 confidence interval.

Table 5-2 Group F2 Design Basis Fuel Assembly Design Specifications		
Parameter	**Value (Group F2)**	**Value (Analyzed)**
Assembly type	17x17 OFA	17x17 OFA
Rod array size	17x17	17x17
Rod pitch, (in)	0.496 [ ]^{a,c}	0.496 [ ]^{a,c}
Active fuel length, (in)	144	144
Stack density, (% TD)	[ ]^{a,c}	[ ]^{a,c}
Maximum enrichment, (wt% ^{235}U)	4.95	5.0
Enrichment tolerance, (wt% ^{235}U)	[ ]^{a,c}	[ ]^{a,c}
Total number of fuel rods	264	264
Fuel cladding OD, (in)	0.360 [ ]^{a,c}	0.360 [ ]^{a,c}
Fuel cladding thickness, (in)	0.0225 [ ]^{a,c}	0.0225 [ ]^{a,c}
Pellet diameter, (in)	0.3088 [ ]^{a,c}	0.3088 [ ]^{a,c}
Number of guide/instrument tubes	24 / 1	24 / 1
Instrument tube OD, (in)	0.4740 [ ]^{a,c}	0.4740 [ ]^{a,c}
Guide tube OD, (in)	0.4740 [ ]^{a,c}	0.4740 [ ]^{a,c}

5.1.1.2 Fuel Storage Cell & Rack Dimensions and Tolerances

The storage racks utilized at Comanche Peak are described in Section 3.2. The fuel storage cell characteristics as they are modeled in the criticality analysis are summarized in this section.

Table 5-3 Region I Fuel Storage Rack Specifications

Parameter	Value	Value (Analyzed)
Rack cell pitch (N-S/E-W) (in)	11.05 / 10.65 [ ]^{a,c}	11.05 / 10.65 [ ]^{a,c}
Cell inner dimension (in)	8.8 [ ]^{a,c}	8.8 [ ]^{a,c}
Cell wall thickness (in)	0.075 [ ]^{a,c}	0.075 [ ]^{a,c}
Flux trap gap width (N-S/E-W) (in)	1.83 / 1.43 [ ]^{a,c}	1.83 / 1.43 [ ]^{a,c}
Neutron absorber type	BORAL	BORAL
Neutron absorber loading (g/cm^2)	[ ]^{a,c}	[ ]^{a,c}
Neutron absorber thickness (in)	0.101	0.101
Neutron absorber cavity width (in)	7.5 [ ]^{a,c}	7.5 [ ]^{a,c}
Neutron absorber cavity height (in)	147 [ ]^{a,c}	147 [ ]^{a,c}
Neutron absorber wrapper thickness (in)	0.0235 [ ]^{a,c}	0.0235 [ ]^{a,c}
Min./Max. distance to SFP wall (in)	5.75 / 19.32	N/A
[ ]^{a,c}		

Table 5-4 Region II Fuel Storage Rack Specifications – No Boraflex Wrapper		
Parameter	**Value**	**Value (Analyzed)**
Rack cell pitch (in)	9.0 ± 0.06	9.0 [ ]^{a,c}
Cell inner dimension (in)	8.83 ± 0.05	8.83 [ ]^{a,c}
Cell wall thickness (in)	0.075 ± 0.004	0.075 [ ]^{a,c}
Flux trap gap width (in)	N/A	N/A
Neutron absorber type	N/A	N/A
Neutron absorber loading (g/cm^2)	N/A	N/A
Neutron absorber cavity width (in)	N/A	N/A
Neutron absorber cavity height (in)	N/A	N/A
Neutron absorber wrapper thickness (in)	N/A	N/A
Min./Max. distance to SFP wall (in)[1]	5.75 / 19.32	3.0
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5.1.2 The Impact of Structural Materials on Reactivity

Over the years, different fuel types have been developed to meet the needs of the utilities. Differences between the fuel types include changes in pin pitch, fuel rod dimensions such as pellet and cladding dimensions, and structural components such as grid material and volumes.

Each of the fuel types present at the plant (in the SFP and in the core) need to be considered. For Comanche Peak, the determination of the bounding fuel assembly design for the analysis has been performed as outlined in Section 4.3. [

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5.1.2.1 Composition of Structural Materials

Various zirconium-based materials and stainless-steel have traditionally been used as structural materials for fuel assembly designs. [

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5.1.2.2 Top and Bottom Nozzles

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5.1.2.3 Grids and Sleeves

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5.1.2.4 Impact of Potential BORAL Blistering

The industry has encountered incidences of the BORAL neutron absorbing material forming blisters in the SFP. There exists a growing concern with the viability of long term use of BORAL in SFPs as illustrated in EPRI Report 1021053, "Reaction-path and Capillary-effects Models of BORAL™ Blistering" (Reference 11) and EPRI Report 1025204, "Strategy for Managing the Long-Term Use of Boral® in Spent Fuel Storage Pools" (Reference 12). To address this concern, a BORAL blistering reactivity confirmation study has been performed.

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Table 5-5 [ ]^{a,c} a,c

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Table 5-6 [ ]^{a,c} a,c

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5.2 ARRAY DESCRIPTIONS

Assembly storage is controlled through the storage arrays defined in Section 5.2. An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity fuel category. Fuel categories are defined by assembly burnup, initial enrichment, and decay time as provided by Table 6-2 through Table 6-15.

Table 5-7 Fuel Categories Ranked by Reactivity

Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	
Fuel Category 5	
Fuel Category 6	Low Reactivity

Notes:

1. Fuel categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
2. For Group F1, Fuel Category 1 and 2 are fuel up to 3.5 wt% ^{235}U; no burnup is required.
3. For Group F1, Fuel Categories 3 and 5 are omitted.
4. For Group F1, Fuel Categories 4 and 6 are determined from the coefficients provided in Section 6.1.
5. For Group F2, Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
6. For Group F2, Fuel Categories 2 through 6 are determined from the coefficients provided in Section 6.1.

Descriptions of the fuel storage arrays allowable for use at Comanche Peak are outlined below.

The restrictions associated with storage in the SFP can be found in Section 6.3.

Array I-A Category 1 assemblies in all cells.	1	1
	1	1

Array Definition	Illustration			
Array II-A Category 6 assembly in every cell. Only valid for two rows adjacent to the SFP wall. The two rows adjacent to Array II-A must be Array II-B, and the empty cell in Array II-B must be adjacent to Array II-A.	WALL	6	6	Array II-B
		6	6	

Array II-B Category 4 assemblies in 3 out of 4 cells, with empty cell in the fourth cell.	4	4
	X	4

Array II-C Checkerboard pattern of two diagonally-opposed Category 5 assemblies with an empty cell and a Category 3 assembly.	5	3
	X	5

Array II-D Checkerboard pattern of two diagonally-opposed Category 2 assemblies with two diagonally-opposed empty cells.	2	X
	X	2

Array II-E 1 out of 4 storage array, with 3 empty cells.	X	X
	X	1

Notes:

1. In all arrays, an assembly of lower reactivity may replace an assembly of higher reactivity.
2. In Group F1, Fuel Category 1 and 2 include fuel up to 3.5 wt% ^{235}U; no burnup is required.
3. In Group F1, Fuel Categories 3 and 5 are omitted.
4. In Group F1, Fuel Categories 4 and 6 are determined from the coefficients provided in Section 6.1.
5. In Group F2, Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
6. In Group F2, Fuel Categories 2 through 6 are determined from the coefficients provided in Section 6.1.
7. An **X** indicates an empty (water-filled) cell.
8. Attributes for each array are as stated in the definition. Diagram is for illustrative purposes only.
9. An empty (water-filled) cell may be substituted for any cell in any storage array.

Figure 5-1 Allowable Storage Arrays

5.3 BURNUP LIMIT GENERATION

To ensure the safe operation of the Comanche Peak SFP, this analysis defines fuel storage arrays which dictate where assemblies can be placed in the SFP based on each assembly's enrichment (wt% ^{235}U), assembly average burnup (GWd/MTU), and decay time (years) since discharge.

Each assembly in the reactor core depletes under slightly different conditions and therefore can have a different reactivity at the same burnup. This is accounted for in the analysis by using a combination of depletion parameters that together produce a bounding isotopic inventory throughout life (see Section 4). Additionally, while fuel manufacturing is a very tightly controlled process, assemblies are not identical. Reactivity margin is added to the KENO reactivity calculations for the generation of burnup limits as discussed in Section 5.3.1 to account for manufacturing deviations.

5.3.1 Target k_{eff} Calculation Description

As discussed in Section 2.1, this analysis provides burnup limits such that the Comanche Peak SFP remains subcritical in unborated conditions. [

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5.3.2 Biases & Uncertainties Calculations

Reactivity biases are known variations between the real and analyzed system and their reactivity impact is added directly to the calculated k_{eff}. [

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5.3.2.1 Bias & Uncertainty Descriptions

The following sections describe the biases and uncertainties that are accounted for in this analysis.

5.3.2.1.1 Manufacturing Tolerances

The reactivity effect of manufacturing tolerances is included in the criticality analysis. [

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Figure 5-2 [**]a,c**

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Figure 5-3 [ ]a,c

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Figure 5-4 [ ]a,c

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Figure 5-5 [ ]a,c

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5.3.2.1.2 Burnup Measurement Uncertainty

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5.3.2.1.3 Depletion Uncertainty

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5.3.2.1.4 Fission Product Worth Uncertainty

A common approach to the validation of cross-sections is by benchmarking critical experiments that are designed to closely represent the configurations of the desired criticality application. [

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5.3.2.1.5 Operational Uncertainty

The operational uncertainty included in the rackup tables accounts for the uncertainty in specific power histories from reactor operation as discussed in Section 4.2.2.3 and Reference 6.

5.3.2.1.6 Eccentric Fuel Assembly Positioning

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5.3.2.1.7 Other Uncertainties

An uncertainty in the predictive capability of SCALE 5.1 and the associated cross-section library is considered in the analysis. The uncertainty from the validation of the calculational methodology is discussed in detail in Appendix A.

5.3.2.1.8 SFP Temperature Bias

The Comanche Peak SFP does not have a nominal temperature; instead it operates within an allowable range. [

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5.3.2.1.9 Borated and Unborated Biases and Uncertainties

Comanche Peak Technical Specifications require the SFP k_{eff} to be $\leq$ 0.95 under borated conditions accounting for all applicable biases and uncertainties. [

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5.3.2.2 Storage Array Biases & Uncertainties Results

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The biases and uncertainties for each fuel storage array are given in Section 5.3.2.2.1 and Section 5.3.2.2.2 for Group F1 fuel and Group F2 fuel respectively.

5.3.2.2.1 Storage Array Biases & Uncertainties for Group F1

Table 5-8 Biases & Uncertainties for Array I-A Group F1 a,c

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Table 5-9 Biases & Uncertainties for Array II-A Group F1

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Table 5-10 Biases & Uncertainties for Array II-B Group F1

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Table 5-11 Biases & Uncertainties for Array II-D Group F1 a,c

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Table 5-12 Biases & Uncertainties for Array II-E Group F1 a,c

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5.3.2.2.2 Storage Array Biases & Uncertainties for Group F2

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Table 5-13 Biases & Uncertainties for Array I-A Group F2

Table 5-14 Biases & Uncertainties for Array II-A Group F2 a,c

Table 5-15 Biases & Uncertainties for Array II-B Group F2 a,c

Table 5-16 Biases & Uncertainties for Array II-C Group F2 a,c

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Table 5-17 Biases & Uncertainties for Array II-D Group F2 a,c

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Table 5-18 Biases & Uncertainties for Array II-E Group F2 a,c

5.4 INTERFACE CONDITIONS

Interfaces are the locations where there is a change in either the storage racks or the storage requirements of the fuel in question. At Comanche Peak each SFP has two regions made up of storage racks of a single design (one design in Region I and one design in Region II). There are two types of interfaces to be considered here, intra-region interfaces, and inter-region interfaces.

5.4.1 Intra-Region Interfaces

Within each region, the only interface conditions that need to be addressed in this analysis are those between different fuel storage arrays. [

]^{a,c} Because the burnup requirements for Array II-A were developed with an interface with Array II-B explicitly included, Array II-A can only interface with Array II-B. Array II-B can interface with any other storage array provided the conditions discussed in this section are met.

5.4.2 Inter-Region Interface

The inter-region interface is the interface between Region I and Region II in the SFPs. This interface is shown in Figure 3-1 and Figure 3-2. The interface between Region I and Region II has been reviewed and storage along the interface is acceptable. There are no restrictions on the orientation of the arrays which interface Region I.

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Table 5-19 Inter-Region Interface Results a,c

5.5 NORMAL CONDITION DESCRIPTION

This section discusses normal conditions within the SFP in addition to the steady-state storage of fresh and spent assemblies. During normal operation, the SFP has a soluble boron concentration of greater than 2400 ppm and a moderator temperature ≤ 150 °F. Beyond the storage of fuel assemblies, there are five major types of normal conditions covered in this analysis. These five conditions are:

5.5.1 Type 1 Normal Conditions

Type 1 conditions involve placement of components in or near the intact fuel assemblies while normally stored in the storage racks. This also includes removal and reinsertion of these components into the fuel when stored in the rack positions using specifically designed tooling. Examples of these evolutions include: control rods, thimble plugging devices, and discrete BA inserts.

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]^{a,c} Therefore, any components designed to be inserted into an assembly may be stored in a fuel assembly in the SFP.

5.5.2 Type 2 Normal Conditions

Type 2 conditions involve evolutions where the fuel assembly is removed from the normal storage rack location for a specific procedure and reinserted after the procedure's completion. The Type 2 normal conditions include removal of an assembly from a storage location to perform fuel assembly cleaning, inspection, reconstitution, or sipping. Descriptions of each of these items are given below along with the evaluation of the impact on this criticality safety analysis.

Fuel assembly cleaning is defined as placing cleaning equipment adjacent to a single assembly and either jetting water from or into a nozzle. The cleaning equipment will displace water adjacent to the assembly and can use demineralized (unborated) water to clean assemblies. The demineralized water used in this process is not confined to a particular volume, but would be readily dispersed into the bulk water of the SFP. In all cases, only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies. The large delta between the Technical Specification required boron concentration and the boron concentration credited in this analysis and the relatively small volume of demineralized water used for this operation guarantees that the addition of unborated water does not constitute a significant dilution event.

Fuel assembly inspection is defined as placing non-destructive examination equipment against at least one face of an assembly. Periscopes and underwater cameras can be placed against all four faces of the assembly simultaneously and will displace water. In all cases, only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies.

Fuel assembly reconstitution is defined as either pulling damaged fuel rods out of an assembly and reinserting intact rods with less reactivity compared to the damaged rod or removing undamaged rods from a damaged assembly for insertion in a new assembly. In most cases, damaged rods will be replaced with stainless steel rods. Natural uranium rods may also be used. In all cases, only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies.

Fuel assembly sipping is defined as placing up to two fuel assemblies in the sipping equipment. The fuel assemblies are separated by at least one assembly pitch via equipment design. While the sipping equipment can be placed within one assembly pitch adjacent to a storage rack loaded with fuel, the fuel assemblies loaded into the sipping equipment are more than one assembly pitch removed from the fuel located in the storage racks. During this operation, demineralized water may be introduced to the sipping container, exposing the assembly(s) to an unborated environment.

Fuel assembly cleaning, inspection, reconstitution, and sipping are bounded by this criticality analysis.
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5.5.3 Type 3 Normal Conditions

Type 3 conditions involve insertion of components that are not intact fuel assemblies, into the fuel storage rack cells. Examples include failed fuel rod baskets, movable in-core detectors, and miscellaneous maintenance equipment. Any components that do not contain fissile materials can replace a fuel assembly of any fuel category in one of the approved storage configurations described in Section 5.2. Items containing fissile material are restricted to storage in Region I.

5.5.4 Type 4 Normal Conditions

Type 4 normal conditions include temporary installation of non-fissile components on the rack periphery. Analyses of the storage arrays contained within this criticality analysis assume an infinite array of storage cells. This assumption bounds the installation of components on the periphery of racks.

5.5.5 Type 5 Normal Conditions

Type 5 conditions involve miscellaneous conditions that do not fit into the first four normal condition types. Examples include usage of fuel handling tools for their intended purpose, miscellaneous debris under the storage racks, and damaged storage cells.

A damaged storage cell is defined as a cell where the cell liner is out of tolerance or the entry channel has been damaged. These cells should not be used to store fuel assemblies but they may be used to store

items that need to be stored as a fuel assembly (i.e. non-fissile material in Region II or a failed fuel basket in Region I, etc) or credited as an empty cell.

Insertion of handling tools into the top of fuel assemblies or other components occurs frequently in the SFP environment. The insertion of handling tools into the top of an assembly is bounded by the storage of inserts in fuel assemblies and therefore, from a criticality perspective, all fuel handling tools are acceptable for their intended purpose.

Comanche Peak contains "fuel inspection cells" as shown in Figure 3-1 and Figure 3-2. These storage cells are each the size of a 2x2 array. The inspection cells are in both Region I and Region II. The Region I fuel inspection cells can be used for fuel handling evolutions without restriction except that only one fuel assembly can be within the inspection cell at a time. The Region II fuel inspection cells can be used for fuel handling evolutions provided that the storage cells adjacent to the fuel inspection cell do not contain fuel assemblies and that the inspection cell contains only one fuel assembly at a time. The cells adjacent to the fuel inspection cell can contain non-fissile material without compromising the function of the fuel inspection cell.

5.6 SOLUBLE BORON CREDIT

The minimum soluble boron concentration in the Comanche Peak SFP to maintain $k_{eff} \leq 0.95$ for the limiting normal condition including biases, uncertainties, and administrative margin is 320 ppm. [

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Table 5-20 [	]a,c

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5.7 ACCIDENT DESCRIPTION

The following reactivity increasing accidents are considered in this analysis:

- Assembly Misload
- SFP temperature greater than normal operating range (150 °F)
- Dropped & misplaced fresh fuel assembly
- Seismic Accident

5.7.1 Assembly Misload into the Storage Racks

This section addresses the potential for an assembly or assemblies to be placed in a location which is not allowed by the burnup requirements in Section 6.1. This analysis addresses multiple assemblies being misloaded in series into unacceptable storage locations and the misload of a single assembly.

5.7.1.1 Multiple Assembly Misload

A multiple assembly misload is a hypothetical accident where assemblies are misloaded in series due to a common cause. [

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Table 5-21 [ ]a,c a,c

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5.7.1.2 Single Assembly Misload

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]a,c This accident requires 1867 ppm of boron (19.4 at% ^{10}B) to maintain $k_{eff} \leq 0.95$.

5.7.2 Spent Fuel Temperature Outside Operating Range

The SFP is to be operated at less than 150 °F. However, under accident conditions this temperature could be higher. [

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5.7.3 Dropped & Misplaced Fresh Assembly

During placement of the fuel assemblies in the racks, it is possible to drop the fuel assembly from the fuel handling machine. The dropped assembly could land horizontally on top of the other fuel assemblies in the rack. [

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It is possible to misplace a fuel assembly in a location not intended for fuel. [

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5.7.4 Seismic Event

In the event of an earthquake or similar seismic event, the SFP storage racks can shift position. This can cause the rack modules to slide together eliminating the space between modules and between modules and the SFP wall. [

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5.7.5 Accident Results

Table 5-22 [ ]^{a,c}

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5.8 RODDED OPERATION

Comanche Peak, like the vast majority of nuclear power plants in the US, operates at hot full power conditions almost exclusively. While standard operation is performed unrodded, it is allowable to operate at hot full power with rods inserted to the insertion limits. Operating with Rod Cluster Control Assemblies (RCCA) inserted into the core impacts the assemblies in the rodded locations. The insertion of an RCCA into an assembly during operation has several effects.

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6 ANALYSIS RESULTS & CONCLUSIONS

This section documents the results of the Comanche Peak criticality safety analysis. Included in this section are the burnup requirements for the fuel storage arrays documented in this analysis. This section also contains the AOA and restrictions of this analysis. Note that the AOA of the validation suite is discussed in A.5.5.

6.1 BURNUP LIMITS FOR STORAGE ARRAYS

Assembly storage is controlled through the storage arrays defined in Section 5.2. An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity fuel category. Fuel categories are defined by assembly average burnup, initial enrichment[1] and decay time as provided by Table 6-2 through Table 6-15.

Table 6-1 Fuel Categories Ranked by Reactivity

Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	
Fuel Category 5	
Fuel Category 6	Low Reactivity

Notes:

1. Fuel categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
2. For Group F1, Fuel Category 1 and 2 are fuel up to 3.5 wt% ^{235}U; no burnup is required.
3. For Group F1, Fuel Categories 3 and 5 are omitted.
4. For Group F1, Fuel Categories 4 and 6 are determined from the coefficients provided in Section 6.1.
5. For Group F2, Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
6. For Group F2, Fuel Categories 2 through 6 are determined from the coefficients provided in Section 6.1.

[1] Initial Enrichment is the nominal ^{235}U enrichment of the central zone region of fuel, excluding axial blankets, prior to reduction in ^{235}U content due to fuel depletion. If the fuel assemby contains axial regions of different ^{235}U enrichment values, such as axial blankets, the maximum Initial Enrichment value is to be utilized.

6.1.1 Burnup Requirements for Group F1 Fuel

Table 6-2 Fuel Category 4: Burnup Requirement Coefficients for Group F1				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
10	-0.0444	-1.3474	22.7039	-28.0852
15	0.2015	-2.6257	24.1016	-28.2473
20	0.4646	-4.1432	26.3891	-29.2170

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 1.35 wt% ^{235}U and 3.50 wt% ^{235}U may be used. Below 1.35 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-3 Fuel Category 4: Burnup Requirements (GWd/MTU) for Group F1				
Decay Time (yr)	**Wt% ^{235}U**			
	1.35	**1.60**	**2.40**	**3.50**
10	0	4.609	18.029	32.969
15	0	4.418	17.258	32.582
20	0	4.301	16.674	32.310

Notes:

1.This table is included as an example, the burnup limits will be calculated using the coefficients provided.

Table 6-4	Fuel Category 6: Burnup Requirement Coefficients for Group F1			
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
10	1.4351	-17.3247	73.3805	-67.4585
15	1.7078	-18.7916	74.6322	-67.2637
20	0.5289	-9.9969	53.7741	-52.6302

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 1.25 wt% ^{235}U and 3.50 wt% ^{235}U may be used. Below 1.25 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-5	Fuel Category 6: Burnup Requirements (GWd/MTU) for Group F1			
Decay Time (yr)	**Wt% ^{235}U**			
	1.25	**1.60**	**2.40**	**3.50**
10	0	11.477	28.703	38.675
15	0	11.036	27.222	36.973
20	0	9.982	26.157	35.793

Notes:

1. This table is included as an example, the burnup limits will be calculated using the coefficients provided.

6.1.2 Burnup Requirements for Group F2

Table 6-6 Fuel Category 2: Burnup Requirement Coefficients for Group F2				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0	1.6738	-8.5396	9.2206

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 3.55 wt% ^{235}U and 5.00 wt% ^{235}U may be used. Below 3.55 wt% ^{235}U, burnup credit is not required.

Table 6-7 Fuel Category 2: Burnup Requirements (GWd/MTU) for Group F2			
Decay Time (yr)	**Wt% ^{235}U**		
	3.55	**4.0**	**5.00**
0	0	1.843	8.367

Notes:

1. This table is included as an example, the burnup limits will be calculated using the coefficients provided.

Table 6-8 Fuel Category 3: Burnup Requirement Coefficients for Group F2				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.5317	-6.1006	32.7118	-36.2263
5	0.5228	-5.9434	31.2846	-34.4602
10	0.5432	-6.1075	31.1578	-33.9933
15	0.5206	-5.8897	30.1768	-32.9600
20	0.5158	-5.7796	29.4050	-32.0577

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 1.45 wt% ^{235}U and 5.00 wt% ^{235}U may be used. Below 1.45 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-9 Fuel Category 3: Burnup Requirements (GWd/MTU) for Group F2				
Decay Time (yr)	**Wt% ^{235}U**			
	1.45	**3.00**	**4.00**	**5.00**
0	0	21.359	31.040	41.280
5	0	20.018	29.043	38.727
10	0	19.179	27.682	37.008
15	0	18.619	26.830	35.756
20	0	18.067	26.099	34.952

Notes:

1. This table is included as an example, the burnup limits will be calculated using the coefficients provided.

Table 6-10	Fuel Category 4: Burnup Requirement Coefficients for Group F2			
Decay Time (yr)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.2553	-3.9826	30.6152	-36.7967
5	0.2366	-3.6430	28.2160	-33.9749
10	0.4387	-5.6018	33.3609	-37.9327
15	0.5450	-6.6302	36.0760	-40.0315
20	0.6327	-7.4663	38.2724	-41.7257

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 1.45 wt% ^{235}U and 5.00 wt% ^{235}U may be used. Below 1.45 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-11	Fuel Category 4: Burnup Requirements (GWd/MTU) for Group F2			
Decay Time (yr)	Wt% ^{235}U			
	1.45	3.00	4.00	5.00
0	0	26.098	38.281	48.626
5	0	24.274	35.743	45.605
10	0	23.578	33.958	43.664
15	0	23.239	33.069	42.718
20	0	22.977	32.395	42.066

Notes:

1. This table is included as an example, the burnup limits will be calculated using the coefficients provided.

Table 6-12 Fuel Category 5: Burnup Requirement Coefficients for Group F2				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.9373	-11.2553	54.7226	-54.1769
5	0.6169	-8.1494	44.7801	-45.7968
10	0.5380	-7.1852	40.7044	-41.9545
15	0.5385	-7.0180	39.2299	-40.3213
20	0.5200	-6.7906	38.0244	-39.0979

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 1.30 wt% ^{235}U and 5.0 wt% ^{235}U may be used. Below 1.30 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-13 Fuel Category 5: Burnup Requirements (GWd/MTU) for Group F2				
Decay Time (yr)	**Wt% ^{235}U**			
	1.30	**3.00**	**4.00**	**5.00**
0	0	34.000	44.615	55.216
5	0	31.855	42.414	51.481
10	0	30.017	40.331	49.187
15	0	28.745	38.774	47.690
20	0	27.899	37.630	46.259

Notes:

1. This table is included as an example, the burnup limits will be calculated using the coefficients provided.

Table 6-14 Fuel Category 6: Burnup Requirement Coefficients for Group F2				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.5789	-7.4498	42.4056	-41.1591
5	0.5247	-6.8992	39.7676	-38.6927
10	0.2701	-4.4306	31.9841	-32.4674
15	0.3105	-4.5582	31.1825	-31.3916
20	0.2374	-3.8754	28.89	-29.4975

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the nominal ^{235}U enrichment. Any enrichment between 1.20wt% ^{235}U and 5.00 wt% ^{235}U may be used. Below 1.20 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-15 Fuel Category 6: Burnup Requirements (GWd/MTU) for Group F2				
Decay Time (yr)	**Wt% ^{235}U**			
	1.20	**3.00**	**4.00**	**5.00**
0	0	34.639	46.316	56.986
5	0	32.684	43.571	53.252
10	0	30.902	41.865	50.450
15	0	29.515	40.279	49.378
20	0	28.703	39.249	47.742

Notes:

1. This table is included as an example, the burnup limits will be calculated using the coefficients provided.

6.1.3 Decay Time Interpolation

This analysis has provided burnup requirements at discrete decay times. However, it is acceptable to interpolate between these decay times to determine burnup limits at alternate decay times. Using linear interpolation between two already analyzed decay times will give a conservative burnup requirement for the decay time in question. This is acceptable because isotopic decay is an exponential function which means assembly reactivity will decay faster than calculations using linear interpolation would predict.

6.2 ANALYSIS AREA OF APPLICABILITY

This section summarizes the data which needs to be confirmed for each cycle of operation to assure that the results presented here remain valid. [

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Table 6-16 [ ]^{a,c} a,c

6.2.1 [ ]^{a,c}

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6.2.1.1 [ ]^{a,c}

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Table 6-17 [ ][a,c] a,c

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6.3 ANALYSIS RESTRICTIONS

The purpose of this section is to summarize the restrictions of the fuel being stored at Comanche Peak.

- All fuel to be stored in the Comanche Peak SFPs must be compared to the Analysis AOA provided in Table 6-16 and any fuel which does not fall within the bounds provided must be specifically analyzed before burnup credit can be taken. See Section 6.2.1.
- Analysis of outlier fuel must use the computer codes and analysis methodology provided in this WCAP.
- Fuel assembly evolutions (fuel cleaning, inspection, reconstitution, and sipping) in Region II must occur with at least one assembly pitch of water between the assembly in question and other assemblies. It is also acceptable to perform these actions above the top of the storage racks.
- An inspection can occur within the Region II storage racks without restriction if it does not involve unborated water and nothing occurs within the assembly envelope or below the top of the active fuel.
- Fissile material that is not fuel in a fuel assembly must be stored in Region I.
- Any storage cells considered damaged (outside of their allowable tolerances) cannot be used to store fuel assemblies without further evaluation. These damaged cells may be used to store non-fuel assembly components such as failed fuel baskets or credited as empty cells in a storage array.
- The fuel inspection cells in Region II can only be used if no fuel is stored in the adjacent fuel storage cells.

6.4 SOLUBLE BORON CREDIT

Soluble boron is credited in the Comanche Peak SFP to keep $k_{eff} \leq 0.95$ under all normal and credible accident scenarios. Under normal conditions, this requires less than 400 ppm of soluble boron. Under accident conditions except for the multiple misload, 1867 ppm of soluble boron is required to ensure $k_{eff} \leq 0.95$, this leaves significant margin to the proposed Technical Specification value of 2400 ppm. The multiple misload accident requires 2400 ppm of soluble boron [ ]^{a,c}

6.5 BORAL NEUTRON ABSORBERS

BORAL neutron absorbers are present in the Region I spent fuel storage racks as discussed in Section 3.2.1. Because of the potential degradation of BORAL discussed in References 11 and 12, Westinghouse performed the study documented in Section 5.1.2.4. [

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APPENDIX A VALIDATION OF SCALE 5.1

A.1 INTRODUCTION

This section summarizes the validation of the SCALE Version 5.1 (SCALE) code system documented in ORNL/TM-2005/39, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation" (Reference A1) for use with the Comanche Peak criticality safety analysis performed in this document. This validation suite develops methodology biases and bias uncertainties through evaluation of critical experiments which were selected based on their applicability to Comanche Peak. [

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In order to validate the SCALE code system, the NRC publication NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (Reference A2) was followed and, as recommended in Reference A2, NEA/NCS/DOC(95)03/IV, 2009 Edition , "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (Reference A3), has been used as the primary source of critical benchmarks for the validation. The following documents were also used as sources of critical benchmarks.

- NUREG/CR-6361, "Criticality Benchmark Guide for Light Water Reactor Fuel in Transportation and Storage Packages" (Reference A4)
- NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data" (Reference A5)
- DSU/SEC/T/2005-33/D.R. "Programme HTC – Phase 1: Réseaux de crayons dans l'eau pure (Water-moderated and reflected simple arrays) Réévaluation des experiences" (Reference A6)

The methodology validated in this section is based on the use of the CSAS25, NITAWL, BONAMI, and KENO modules of the SCALE code system.

This validation is applicable to fresh and spent fuel storage. It also covers the criticality analysis for any movement of fuel from the SFP to the core and other normal operations in the SFP. The validation is adequate to cover all present and anticipated (non-mixed-oxide) LWR fuel designs. The validation suite AOA is discussed in more detail in subsection A.5.5.

A.2 CALCULATIONAL METHOD

The analysis methodology employs SCALE, as documented in Reference A1, using a 44-group ENDF/B-V neutron cross-section library.

The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. The SCALE version that is utilized here is a code system that runs on LINUX clusters and includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL, and KENO.

Standard material compositions are employed in the SCALE analyses consistent with the material descriptions in References A3 through A6.

All calculations are performed on systems with the following software characteristics:

- SCALE Version 5.1
- SUSE Linux 9.0 and 10.0

Correct installation and operation of the SCALE code system is verified by performing test cases on the platforms described above. The validation and verification were performed on the same platform as the safety analysis.

A.3 VALIDATION METHOD

Validation of SCALE to perform criticality safety calculations has been performed following the methodology outlined in Reference A2. Validation includes quantification of the difference between calculated and experimental neutron effective multiplication factors (k_{eff}), called the bias. The bias and the uncertainty associated with the bias will be used in combination with additional subcritical margin and other bias and uncertainty calculations to establish a target k_{eff} as described in Section 5.3.1.

Statistical analysis is performed to determine whether trends exist in the bias. The range of benchmark experiment parameters used in the validation defines the validation suite AOA, which establishes the limits of the systems that can be analyzed using the validation presented here.

As described in Reference A2, the basic methodology used in the validation is:

1. Define the range of parameters to be validated
2. Select critical experiment data
3. Model experiments
4. Analyze the data

This validation is intended to cover the range of parameters associated with this analysis. [

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A description of the critical experiments modeled is contained in Section A.4. The results and AOA are contained in Section A.5.

A.3.1 Determination of Bias and Bias Uncertainty

The statistical analysis presented in Section 2.4.1 of Reference A2 is followed here. This approach involves determining a weighted mean that incorporates the uncertainty from both the measurement (σ_{exp})

and the calculation method (σ_{calc}). For the benchmark experiments chosen from References A3, A5, and A6, the experimental uncertainty presented in References A3, A5, and A6 are used. No experimental uncertainty is presented for the experiments contained in Reference A4 so the average value of experimental uncertainties of similar experiments documented in Reference A3 is used. This is consistent with the recommendation in Reference A2 that engineering judgment be used to approximate typical experimental uncertainties rather than assume no experimental uncertainty.

If the critical experiment being modeled is at a state other than critical (i.e. $k \neq 1.0$) then an adjustment is made to the calculated value of k_{eff}. This adjustment is done by normalizing the calculated eigenvalue to the experimental value. This normalization assumes that the inherent bias in the calculation is not affected by the normalization, which is valid for small differences in k_{eff}. To normalize k_{eff}, the following formula is used:

$$k_{norm} = \frac{k_{calc}}{k_{experiment}}$$

The normalized k_{eff} values are used in the subsequent determination of the bias and bias uncertainty, therefore all subsequent instances of k_{eff} should be taken to mean the normalized k_{eff} value.

The Monte Carlo and experimental uncertainties are root-sum-squared to create a combined uncertainty for each experiment:

$$\sigma_t = \sqrt{\sigma_{calc}^2 + \sigma_{experiment}^2}$$

A weighted mean k_{eff} ($\bar{k}_{eff}$) is calculated by using the weighting factor $1/\sigma_t^2$. The use of this factor reduces the "weight" of the data with high uncertainty. Within a set of data, the "i^{th}" member of that set is shown with a subscript "i". Henceforth, unless otherwise specified, the uncertainty for an "i^{th}" k_{eff} is shown as σ_i and is taken to mean the combined calculational and experimental uncertainty, shown above as σ_t. The weighted equation variables for the single sided lower tolerance limit are presented below:

Variance about the mean:

$$s^2 = \frac{\left(\frac{1}{n-1}\right)\Sigma\frac{1}{\sigma_i^2}\left(k_{eff_i} - \bar{k}_{eff}\right)^2}{\frac{1}{n}\Sigma\frac{1}{\sigma_i^2}}$$

Average total uncertainty:

$$\bar{\sigma}^2 = \frac{n}{\Sigma\frac{1}{\sigma_i^2}}$$

The weighted mean k_{eff} value:

$$\overline{k}_{eff} = \frac{\sum \frac{1}{\sigma_i^2} k_{eff_i}}{\sum \frac{1}{\sigma_i^2}}$$

The square root of the pooled variance:

$$S_P = \sqrt{s^2 + \overline{\sigma}^2}$$

Where:

s^2 = variance about the mean

n = number of critical experiments used in the validation

$\overline{\sigma}$ = average total uncertainty

Bias is determined by the relation:

Bias = $\overline{k_{eff}}$ - 1 if $\overline{k_{eff}}$ is less than 1.0; otherwise Bias = 0

A.3.1.1 Test for Normal Distribution

The statistical evaluation performed must be appropriate for the distribution of the data. If the data is normally distributed then a technique such as a one-sided tolerance limit can be used to determine the appropriate bias and bias uncertainty. [

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A.3.1.2 Identify Trends in the Data

Trends are determined through the use of regression fits to the calculated results. Based on a visual inspection of the data plots, it is determined that a linear fit is sufficient to determine a trend in the bias. The data plots are shown in Section A.5. In the equations below, "x" is the independent variable representing some parameter (e.g., enrichment). The variable "y" represents k_{eff}. Variables "a" and "b" are coefficients for the function where "b" is the slope and "a" is the intercept.

Per Reference A2, the equations used to produce a weighted fit of a straight line to the data are given below.

$$Y(x) = a + bx$$

$$a = \frac{1}{\Delta}\left[\sum \frac{x_i^2}{\sigma_i^2} \sum \frac{y_i}{\sigma_i^2} - \sum \frac{x_i}{\sigma_i^2} \sum \frac{yx_i}{\sigma_i^2}\right]$$

$$b = \frac{1}{\Delta}\left[\sum\frac{1}{\sigma_i^2}\sum\frac{yx_i}{\sigma_i^2} - \sum\frac{x_i}{\sigma_i^2}\sum\frac{y_i}{\sigma_i^2}\right]$$

$$\Delta = \sum\frac{1}{\sigma_i^2}\sum\frac{x_i^2}{\sigma_i^2} - \left(\sum\frac{x_i}{\sigma_i^2}\right)^2$$

Once the data has been fit to a line, a determination as to the "goodness of fit" must be made. Per Reference A2, there are two steps that should be employed when determining the goodness of fit. The first is to plot the data against the independent variable which allows for a visual evaluation on the effectiveness of the regression fit. The resulting plots are shown in Section A.5.

The second step is to numerically determine a goodness of fit after the linear relations are fit to the data. This adds a useful measure because visual inspection of the data plot will not necessarily reveal just how good the fit is to the data. Per Reference A2, the linear correlation coefficient is one standard method used to numerically measure the goodness of fit.

The linear-correlation coefficient is a quantitative measure of the degree to which a linear relationship exists between two variables. For weighted data, the linear correlation coefficient is:

$$r = \frac{\sum\frac{1}{\sigma_i^2}(x_i - \bar{x})(y_i - \bar{y})}{\sqrt{\sum\frac{1}{\sigma_i^2}(x_i - \bar{x})^2}\sqrt{\sum\frac{1}{\sigma_i^2}(y_i - \bar{y})^2}}$$

Where the weighted mean for the independent parameter is:

$$\bar{x} = \frac{\sum\frac{1}{\sigma_i^2}x_i}{\sum\frac{1}{\sigma_i^2}},$$

The weighted mean for the dependent parameter ($\bar{y}$) is $\bar{k}_{eff}$, shown in Section A.3.1.

The value of the linear correlation coefficient is often expressed as a squared term (r^2). The closer r^2 approaches the value of 1, the better the fit of the data to the linear equation.

In addition to the linear correlation coefficient, the Student t-test is used to determine if the linear fit of the data is statistically significant.

The Student t-test is a common statistical test used to determine if the null hypothesis is supported. For the purposes of this validation suite, the null hypothesis is that the slope of the regression fit (b) is equal to some specified value (b_0). When b_0 is set to zero the null hypothesis is that x and y are unrelated,

where x is an independent variable and y is k_{eff}. In order to determine if the null hypothesis is supported, t_{score} is calculated and compared to the Student t-distribution.

$$t_{score} = \frac{(b - b_0)\sqrt{n-2}}{\sqrt{\frac{SSE}{\sum(x_i - \bar{x})^2}}}$$

Where SSE is the sum of the squares of residuals.

$$SSE = \sum [k_{eff_i} - (a + bx_i)]^2$$

Per Reference A2, when a relationship between a calculated k_{eff} and an independent variable can be determined, a one-sided lower tolerance band may be used. This is a conservative method that provides a fitted curve above which the true population of k_{eff} is expected to lie. The equation for the one-sided lower tolerance band of a Normal distribution is:

$$K_L = K_{fit}(x) - S_{P_{fit}} \left\{ \sqrt{2F_a^{(2,n-2)} \left[\frac{1}{n} + \frac{(x - \bar{x})^2}{\sum(x_i - \bar{x})^2} \right]} + z_{2P-1} \sqrt{\frac{(n-2)}{\chi^2_{1-\Upsilon,n-2}}} \right\}$$

$K_{fit}(x)$ is the function derived in the trend analysis described above. Because a positive bias may be nonconservative, the equation below must be used for all values of x where $K_{fit}(x) > 1$:

$$K_L = 1 - S_{P_{fit}} \left\{ \sqrt{2F_a^{(2,n-2)} \left[\frac{1}{n} + \frac{(x - \bar{x})^2}{\sum(x_i - \bar{x})^2} \right]} + z_{2P-1} \sqrt{\frac{(n-2)}{\chi^2_{1-\Upsilon,n-2}}} \right\}$$

Where:

p = The desired confidence level (0.95)

$F_a^{(fit,n-2)}$ = The F distribution percentile with degree of fit, n-2 degrees of freedom. The degree of fit is 2 for a linear fit.

n = The number of critical experiment k_{eff} values

x = The independent fit variable

x_i = The independent parameter in the data set corresponding to the i^{th} k_{eff} value

$\bar{x}$ = The weighted mean of the independent variables

z_{2P-1} = The symmetric percentile of the normal distribution that contains the P fraction

$\Upsilon = \frac{1-p}{2}$

$\chi^2_{1-\Upsilon,n-2}$ = The upper Chi-square percentile

For a weighted analysis:

$$\sum (x_i - \bar{x})^2 = \frac{\Sigma \frac{1}{\sigma_i^2}(x_i - \bar{x})^2}{\frac{1}{n}\Sigma \frac{1}{\sigma_i^2}}$$

$$\bar{x} = \frac{\Sigma \frac{1}{\sigma_i^2} x_i}{\Sigma \frac{1}{\sigma_i^2}}$$

$$S_{P_{fit}} = \sqrt{s_{fit}^2 + \bar{\sigma}^2}$$

$$\bar{\sigma}^2 = \frac{n}{\Sigma \frac{1}{\sigma_i^2}}$$

$$s_{fit}^2 = \frac{\left(\frac{1}{n-2}\right)\Sigma\left\{\frac{1}{\sigma_i^2}\left[k_{eff_i} - K_{fit}(x_i)\right]^2\right\}}{\frac{1}{n}\Sigma \frac{1}{\sigma_i^2}}$$

Within the equation for K_L:

$$\text{Bias} = K_{fit} - 1 \quad \text{if } K_{fit} < 1.0; \text{ otherwise Bias} = 0$$

And the uncertainty in the bias is:

$$S_{P_{fit}}\left\{\sqrt{2F_a^{(2,n-2)}\left[\frac{1}{n} + \frac{(x-\bar{x})^2}{\Sigma(x_i-\bar{x})^2}\right]} + z_{2P-1}\sqrt{\frac{(n-2)}{\chi^2_{1-\gamma,n-2}}}\right\}$$

A.3.1.3 Treatment for Non-Normal Distributions

If a dataset fails a normality test, the non-parametric treatment described in Reference A2 can be applied to that data subset.

Per Reference A2, to calculate K_L, the lower limit of the 95/95 tolerance band:

K_L = k_{eff}^{min} – uncertainty for k_{eff}^{min} – NPM

Where:

k_{eff}^{min} = minimum normalized k_{eff},

uncertainty for k_{eff}^{min} is the spooled Monte Carlo and experimental uncertainty , and

NPM is the non-parametric margin.

The degree of confidence for 95% of a population can be generalized as

$$\beta = 1 - \sum_{j=0}^{m-1} \frac{n!}{j!(n-j)!}(1-q)^j q^{n-j},$$

Where:

q = desired population fraction,

n = number of data in one data sample,

m = rank order indexing from the smallest sample to the largest.

Using a population fraction of 95% and a rank order of 1, the equation reduces to

$$\beta = 1 - q^n = 1 - 0.95^n.$$

If enough data is not available to achieve a 95/95 tolerance level, the NPM is utilized as a correction factor, which can be found in Table 2.2 of Reference A2.

A.4 CRITICAL EXPERIMENT DESCRIPTION

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The Organization for Economic Co-operation and Development's (OECD) International Handbook of Evaluated Criticality Safety Benchmark Experiments, Reference A3, is the primary source for most of the experiments used in this suite; References A4 through A6 are used to supplement the selected experiments to ensure coverage of all the important parameters. Each set of experiments is described in detail below. All experiments used in this validation were determined to be acceptable as benchmark data.

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A.4.9 Experiments from NUREG/CR-6361

NUREG/CR-6361 (Reference A4) is intended to serve as a guide for performing criticality benchmark calculations for light-water-reactor (LWR) fuel applications. It documents 180 critical experiments and includes recommendations for selecting suitable experiments and determining the calculational bias and bias uncertainty. [

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A.4.10 HTC Experiments

The Haut Taux de Combustion (HTC) experiments are a series of experiments performed with mixed oxide rods designed to have a U and Pu isotopic composition equal to that of U(4.5%)O_2 PWR fuel with 37.500 GWd/MTU burnup. No fission products are included in the composition. Up to this point all the experiments modeled in this suite represent fresh fuel; the HTC experiments are included to ensure the validation suite covers spent fuel as well. The HTC critical experiment set is organized into four phases:

- Phase 1 – Water-Moderated and Reflected Simple Arrays (Reference A6)
- Phase 2 – Reflected Simple Arrays Moderated by Water Poisoned with Gadolinium or Boron
- Phase 3 – Pool Storage
- Phase 4 – Shipping Cask

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A.5 RESULTS

This section presents the results of the validation analysis. This includes the raw calculational results, the calculation of the bias and bias uncertainty, the detailed statistical trending results, and the definition of the AOA.

A.5.1 Raw Calculational Results

Table A-1 shows the raw calculational results for each of the critical experiments considered in this validation.

Table A-1 Raw Calculational Results a,c

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A.5.2 Determination of Bias and Bias Uncertainty and Normality Check

This validation suite is intended to be used for fresh and spent fuel storage at Comanche Peak. [

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Table A-2 [ ]^{a,c}

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Table A-3 [ ]a,c

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Table A-4 [ ]a,c

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A.5.3 Trending Analysis

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Figure A-2 [ ]a,c

Table A-6 [ ]a,c

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Figure A-3 [**]^{a,c}**

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Figure A-4 [ ][a,c]

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Figure A-5 [**]a,c**

Table A-9 [**]a,c**

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Figure A-6 [**]a,c**

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Table A-11 [ ][a,c]

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Figure A-7 [ ]a,c

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Figure A-9 [ ]a,c

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Figure A-10 [**]a,c**

Table A-16 [ ]^{a,c} a,c

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Per Reference A2, when a relationship between a calculated k_{eff} and an independent variable can be determined, a one-sided lower tolerance band may be used. This is a conservative method that provides a fitted curve above which the true population of k_{eff} is expected to lie. The one sided lower tolerance band method is described in Section A.3 and representative results are shown in Table A-17 and Table A-18.
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Table A-17 [ ]a,c a,c

Table A-18 [ ]a,c a,c

Table A-19 [ ]^{a,c} a,c

A.5.3.3 Summary of Trend Results

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Table A-20 [ ]a,c	a,c

A.5.4 Non-Parametric Treatment Results

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Table A-21 [ ]a,c

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A.5.5 Validation Suite Area of Applicability Definition

The AOA of this benchmark is defined by the range of parameters in the validation suite. Table A-22 summarizes the Areas of Applicability.

Table A-22 Validation Suite Area of Applicability a,c

A.6 SUMMARY OF THE BIAS AND BIAS UNCERTAINTY

Table A-23 summarizes the results of the validation.

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Table A-23 Summary of Bias and Bias Uncertainty

REFERENCES

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