Enclosure 4

Westinghouse WCAP-17400-NP

Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis

Non-Proprietary

149 pages follow

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Revision 0

July 2011

Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis

Westinghouse

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REVISION HISTORY

Revision	Description and Impact of the Change	Date
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LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS

2-D	Two-Dimensional
3-D	Three-Dimensional
400V+	400 Vantage +
422V+	422 Vantage +
AEG	Average Energy Group of Neutrons Causing Fission
AOA	Area of Applicability
at%	Atom Percent
B&W	The Babcock & Wilcox Company
BOL	Beginning of Life
DR	Debris Resistant
EALF	Energy of Average Lethargy causing Fission
ECT	Eddy Current Testing
En	Enrichment
ENC	Exxon Nuclear Fuel
ENDF/B	Evaluated Nuclear Data File
EOL	End of Life
EPU	Extended Power Uprate
FA	Fuel Assembly
FP	Fission Products
FRSB	Fuel Rod Storage Basket
GT	Guide Tube
GWd	Gigawatt days
HTC	Haut Taux de Combustion
HVFD	Hafnium Vessel Flux Depression
ID	Inner Diameter
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixing
ISG	Interim Staff Guidance
IT	Instrumentation Tube
k_{eff}	Effective neutron multiplication factor
KENO	SCALE Module KENO.V.a
LEU	Low Enriched Uranium
LV	Limiting Value
LWR	Light Water Reactor
MOX	Mixed Oxide Fuel
MTU	Metric Ton Uranium
MWd	Megawatt-day
MWt	Megawatts thermal
NFR	New Fuel Storage Rack
OD	Outer Diameter
OECD	Organization for Economic Co-operation and Development
OFA	Optimized Fuel Assembly
Optimized ZIRLO	Optimized ZIRLO®
ORNL	Oak Ridge National Lab

LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS (cont.)

Plexiglas	Plexiglas™
PNL	Pacific Northwest National Lab
ppm	parts per million
Prairie Island	Prairie Island Nuclear Generating Plant
psia	Pounds Per Square Inch Absolute
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assemblies
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SS	Stainless Steel
STD	Standard Fuel Assembly
TD	Theoretical Density
UT	Ultrasonic Testing
WABA	Wet Annular Burnable Absorber
Westinghouse	Westinghouse Electric Co. LLC
wt%	Weight Percent
Zirlo	Zirlo®

1 INTRODUCTION

The purpose of this report is to document the criticality safety analysis performed to support the proposed Extended Power Uprate (EPU) at Prairie Island Nuclear Generating Plant Units 1 and 2 (Prairie Island) and to address non-conservatisms in the previous analysis. The EPU impacts the criticality analysis because fuel depleted at EPU conditions is more reactive than fuel depleted under pre-EPU conditions at the same burnup. This is due to the higher fuel and moderator temperatures that harden the neutron spectrum, resulting in more plutonium production.

2 OVERVIEW

The existing racks are evaluated for the placement of fuel within new allowable storage arrays. This evaluation does not credit any Boraflex neutron absorber that might remain in the racks. 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 Spent Fuel Pool (SFP) and for the presence of full-length Rod Cluster Control Assemblies (RCCAs) placed in selected fuel assemblies.

2.1 ACCEPTANCE CRITERIA

The objective of this SFP criticality safety analysis is to ensure that the pool operates within the bounds discussed here.

1. All calculations of the effective neutron multiplication factor (k_{eff}) performed for permissible storage arrangements at a soluble boron concentration of 0 ppm yield results less than 1.0 including margin for all applicable biases and uncertainties with 95% probability at a 95% confidence level.

2. All calculations of the effective neutron multiplication factor (k_{eff}) performed for permissible storage arrangements at a soluble boron concentration of 400[1] ppm yield results less than 0.95 including margin for all applicable biases and uncertainties with 95% probability at a 95% confidence level.

3. The analysis demonstrates that k_{eff} is less than 0.95 under all postulated accident conditions with a soluble boron concentration of less than 1800 ppm. This criterion shall also be met 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 Prairie Island 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 Prairie Island SFP. Input selection is discussed in Section 3 of this analysis.

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 Fuel Categories 1 through 6 based on assembly reactivity (burnup, enrichment, and decay time). An assembly's fuel category determines which storage arrays are acceptable for the assembly in question. Fuel Category 1 defines the

[1] The limiting configuration requires a minimum soluble boron concentration of 350 ppm; however, to provide margin, the associated Technical Specifications (TS 4.3.1) may require any value between 350 ppm and the boron dilution endpoint. 400 ppm has been chosen as that value.

most reactive assemblies, i.e. representing a fresh 5.0 wt% ^{235}U assembly and Fuel Category 6 defines the least reactive assemblies, i.e. representing assemblies that can be stored in Array A (see Section 3.7.1). Fuel Category 7 is applicable solely to consolidated fuel (see Section 3.6.3.1).

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
Fuel Category 7	Consolidated Rod Storage Canister Category

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. Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
3. Fuel Categories 2 through 6 are determined from the coefficients provided in Section 5-1.
4. Fuel Category 7 applies only to consolidated rod storage canisters and is not ranked relative to the other fuel categories. Fuel Category 7 fuel may not be substituted for other fuel categories.

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 Reference 1, and its cross-section library based on Evaluated Nuclear Data File (ENDF/B) Version VI.3, and (2) SCALE Version 5.1, as documented in 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.

PARAGON is Westinghouse's 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 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 1). PARAGON has been chosen for this spent fuel criticality analysis because of its improvements (e.g., no lumped fission products) relative to the historically used PHOENIX-P (Reference 4) and because it has all the attributes needed for burnup credit applications. There are no SER 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 processing code (Reference 2). 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 pool, as well as finite rack module and pool representations to evaluate soluble boron requirements and postulated accident scenarios.

The SCALE package includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II (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 allows for accurate and detailed modeling of fuel assemblies in storage racks, either as infinite arrays or in actual pool 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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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 recommended SCALE library 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 (Reference 5).

3 ANALYSIS INPUT SELECTION

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 Prairie Island. Unless otherwise stated, all calculations assumed nominal characteristics for the fuel and the fuel storage cells. The effect of manufacturing tolerances is accounted for by combining the reactivity effects associated with manufacturing tolerances (rack, fuel, etc) with the other uncertainties as discussed in Section 4.1.2.

As discussed in Section 2.3.2, KENO is the criticality code used for the Prairie Island Monte Carlo calculations. KENO is used to determine the absolute reactivity of burned and fresh fuel assemblies. Additionally KENO is used to calculate the reactivity sensitivity to manufacturing tolerances and to perform the calculations needed to account for eccentric fuel positioning, the allowable temperature range of the SFP, and accident analyses.

All Monte Carlo calculations are performed using an explicit model of the fuel and storage cell geometry and employ periodic or reflective boundary conditions as appropriate. Modeling assumptions are described in the appropriate section below.

3.1 FUEL ASSEMBLY SELECTION

Prairie Island has operated using several different fuel assembly designs manufactured by Westinghouse Electric Co. LLC (Westinghouse) as well as fuel assembly designs manufactured by Exxon Nuclear Fuel[2] (ENC). The different fuel designs used at Prairie Island were analyzed to determine the limiting fuel assembly design with respect to criticality safety. The limiting fuel assembly design was used to develop the isotopic number densities and develop burnup limits.

The fuel assemblies for Prairie Island incorporate a 14x14 square array of 179 fuel rods with 16 control rod guide tubes (GT) and one instrumentation tube (IT). The fuel rod cladding is Zircaloy or ZIRLO® including Optimized ZIRLO® and other variants. Each standard fuel rod contains a column of enriched UO_2 fuel pellets. The pellets are pressed and sintered and are dished on both ends. Gadolinia bearing fuel rods, when part of the fuel assembly design, contain UO_2-Gd_2O_3 fuel pellets (Burnable absorber usage is discussed in Section 3.3.4).

Westinghouse manufactured the fuel assemblies for Cycles 1 through 4 for both units. The fuel assembly design was Westinghouse 14X14 STANDARD (STD).

ENC supplied fuel assemblies for Cycles 5 through 10 using three different designs. The ENC fuel designs are the ENC STANDARD, ENC HIGH BURNUP, and ENC TOPROD fuel designs.

Beginning with Cycle 11 for both units, the Westinghouse Optimized Fuel Assembly (OFA) design was used. Since then, Prairie Island has also incorporated the Westinghouse HIGH BURNUP-OFA, 400 VANTAGE + (400V+) and 422 VANTAGE + (422V+) fuel designs. The 422V+ fuel design is the current fuel design and there are no plans to transition to another fuel type in the foreseeable future.

[2] Exxon Nuclear Fuel has been acquired by AREVA and so the Exxon designs discussed here are the proprietary property of AREVA.

For the purposes of SFP criticality safety analyses, the Westinghouse fuel designs fall into two general categories: OFA and Standard. The OFA category consists of the Westinghouse OFA, HIGHBURNUP-OFA, and 400V+ fuel designs. The Standard category consists of the Westinghouse STD and 422V+ fuel designs. The differences between the OFA designs are based on changes in cladding and grid material. [

]a,c Table 3-1 summarizes the different OFA assemblies.

Table 3-1 OFA Fuel Assembly Specifications

Parameter	Value		
Assembly type	OFA	HIGHBURNUP-OFA	400V+
Rod array size	14x14	14x14	14x14
Rod pitch, inch	0.556	0.556	0.556
Active fuel length, inch	144	144	144
Stack density, % TD	[ ]a,c	[ ]a,c	[ ]a,c
Maximum enrichment, wt% ^{235}U	5.0	5.0	5.0
Total number of fuel rods	179	179	179
Fuel cladding OD, inch	0.400	0.400	0.400
Fuel cladding ID, inch	0.3514	0.3514	0.3514
Fuel cladding thickness, inch	0.0243	0.0243	0.0243
Pellet diameter, inch	0.3444	0.3444	0.3444
Number of guide/instrument tubes	16 / 1	16/1	16 / 1
Guide/instrument tube OD, inch	0.5260 / 0.399	0.5260 / 0.400	0.5260 / 0.400
Guide/instrument tube ID, inch	0.4920 / 0.3520	0.4920 / 0.3520	0.4920 / 0.3520
Guide/instrument tube thickness, inch	0.0170 / 0.0235	0.0170 / 0.0235	0.0170 / 0.0235

The Standard category consists of the Westinghouse Standard and 422V+ fuel types. The fuel designs in this category contain fuel pellets and fuel rods with the same diameter; the differences between designs include changes to cladding and grid material, and instrumentation and guide tube dimensions. [
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Table 3-2 contains descriptions of the STD and 422V+ fuel designs.

Table 3-2 Standard Fuel Assembly Specifications		
Parameter	**Value**	
Assembly type	STD	422V+
Rod array size	14x14	14x14
Rod pitch, inch	0.556	0.556
Active fuel length, inch	144	143.25
Stack density, % TD	[ ]a,c	[ ]a,c
Maximum enrichment, wt% ^{235}U	5.0	5.0
Total number of fuel rods	179	179
Fuel cladding OD, inch	0.4220	0.4220
Fuel cladding ID, inch	0.3734	0.3734
Fuel cladding thickness, inch	0.0243	0.0243
Pellet diameter, inch	0.3659	0.3659
Number of guide/instrument tubes	16 / 1	16 / 1
Guide/instrument tube OD, inch	0.5390 / 0.4220	0.5260 / 0.4220
Guide/instrument tube ID, inch	0.5050 / 0.3740	0.4920 / 0.3740
Guide/instrument tube thickness, inch	0.0170 / 0.0240	0.0170 / 0.0240

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Table 3-3 Westinghouse Fuel Design Comparison

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The ENC fuel assemblies that have been used at Prairie Island were also reviewed to determine whether an ENC fuel design may be more limiting. The three ENC fuel types used by Prairie Island (STANDARD, HIGHBURNUP and TOPROD) are summarized in Table 3-4. [

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[3] When performing a fuel design reactivity comparison, any reasonable burnup profile is acceptable for use so long as all the designs being compared contain the same profile. For example, it would be misleading to compare the OFA fuel in Table 3-3 to the ENC fuel in Table 3-5.

Table 3-4 ENC Fuel Assembly Specifications

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Table 3-5 Westinghouse – Exxon Fuel Design Comparison

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3.1.1 Fuel Assembly Modeling Assumptions

Fuel assemblies used in this analysis were modeled in PARAGON for depletion calculations and in KENO for reactivity calculations.

PARAGON performs heterogeneous, 2-D transport calculations to develop the spent fuel isotopic number densities. The calculations use the exact fuel assembly geometry and material composition from a cross-sectional slice of an assembly. [
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KENO is used to perform 3-D Monte Carlo calculations to determine reactivity as discussed in Section 2.3.2

In KENO, the following assumptions have been made with regard to the fuel assembly.

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Table 3-6 Design Basis Fuel Assembly Design Specifications		
Parameter	**Value (422V+)**	**Value (Analyzed)**
Assembly type	14x14 422 Vantage +	14x14 422 Vantage +
Rod array size	14x14	14x14
Rod pitch, inch	0.556 [ ]^{a,c}	0.556 [ ]^{a,c}
Active fuel length, inch	143.25	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	179	179
Fuel cladding OD, inch	0.422 [ ]^{a,c}	0.422 [ ]^{a,c}
Fuel cladding ID, inch	0.3734 [ ]^{a,c}	0.3734 [ ]^{a,c}
Fuel cladding thickness, inch	0.0243 [ ]^{a,c}	0.0243 [ ]^{a,c}
Pellet diameter, inch	0.3659 [ ]^{a,c}	0.3659 [ ]^{a,c}
Number of guide/instrument tubes	16 / 1	16 / 1
Instrument tube OD, inch	0.4220 [ ]^{a,c}	0.4220 [ ]^{a,c}
Instrument tube ID, inch	0.3740 [ ]^{a,c}	0.3740 [ ]^{a,c}
Instrument tube thickness, inch	0.0240 [ ]^{a,c}	0.0240 [ ]^{a,c}
Guide tube OD, inch	0.5260 [ ]^{a,c}	0.5260 [ ]^{a,c}
Guide tube ID, inch	0.4920 [ ]^{a,c}	0.4920 [ ]^{a,c}
Guide tube thickness, inch	0.017 [ ]^{a,c}	0.017 [ ]^{a,c}

3.2 FUEL STORAGE CELL & RACK DESCRIPTION

The storage racks utilized at Prairie Island are high density flux trap racks that were designed with Boraflex neutron absorber inserts. Any remaining Boraflex neutron absorber material is not credited in the criticality analysis. The fuel storage cell characteristics that are used in the criticality analysis are summarized in Table 3-7.

Table 3-7 Fuel Rack Specifications	
Parameter	**Value**
Rack material & density	SS304 7.94 g/cc
Cell inner dimension, inch	8.27 [ ]a,c
Cell wall thickness, inch	0.09 [ ]a,c
Cell pitch, inch	9.50 [ ]a,c
Poison cavity thickness, inch	0.125 [ ]a,c
Poison cavity width, inch	8.20 [ ]a,c
Sheathing thickness, inch	0.024 [ ]a,c

3.2.1 Fuel Storage Cell & Rack Modeling Assumptions

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3.3 FUEL DEPLETION PARAMETER SELECTION

3.3.1 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 operated in Cycles 1-4 of either unit. Note that the presence of Burnable Poison Rod Assemblies (BPRs) (only present in Cycles 1-4) is accounted for in the burnup requirements provided for fuel operated during Cycles 1-4. Section 3.3.4 discusses BPRs in more detail.

The methodology for generating isotopic number densities to support burnup credit in this analysis is based on in-reactor operation at bounding operating conditions. This is done through the depletion of 2-D axial nodes in infinite reactor geometry at the bounding core conditions discussed in Section 3.3.2. Once the fuel has been depleted to a desired burnup, it is allowed to decay to its most reactive state (100 hours) and any residual xenon is removed. The isotopic inventory from PARAGON is then brought to cold conditions, 68 °F and 14.7 psia. Credit is taken for the decay of ^{241}Pu to ^{241}Am.

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Based on Prairie Island fuel management, the fuel exposed to BPRs has isotopic number densities which are calculated at enrichments of 2.5, 3.0, and 3.4 wt% ^{235}U and at decay times of 0 and 20 years. Isotopic number densities have been generated for fuel that has not been exposed to BPRs at enrichments of 3.4, 4.0, 4.5, and 5.0 wt% ^{235}U and decay times of 0, 5, 10, 15, and 20 years. [

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3.3.2 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 pool. Reference 7 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 nominal operating parameters for Prairie Island are given in Table 3-11.

3.3.2.1 Soluble Boron Concentration

The soluble boron concentration in the reactor during operation impacts the reactivity of fuel being discharged to the pool. Boron is a strong thermal neutron absorber and so its presence hardens the neutron energy spectrum in the core, creating more plutonium. To adequately incorporate the impact of soluble boron into the generation of isotopics, it is important to account for reactor operation with respect to soluble boron.

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Table 3-8 Cycle Average Soluble Boron Concentrations

Cycle	Soluble Boron (ppm)	Cycle	Soluble Boron (ppm)
Unit 1 Cycle 19	697	Unit 2 Cycle 19	723
Unit 1 Cycle 20	804	Unit 2 Cycle 20	749
Unit 1 Cycle 21	816	Unit 2 Cycle 21	740
Unit 1 Cycle 22	825	Unit 2 Cycle 22	773
Unit 1 Cycle 23	770	Unit 2 Cycle 23	735
Unit 1 Cycle 24	804	Unit 2 Cycle 24	845
Unit 1 Cycle 25	713	Unit 2 Cycle 25	732
Unit 1 Cycle 26	698	Unit 2 Cycle 26	747
EPU Cycle 28	781	EPU Cycle X[4]	802
EPU Cycle 29	779	EPU Cycle Y[4]	811
EPU Cycle 30	795	EPU Cycle Z[4]	826

3.3.2.2 Fuel Temperature

The fuel temperature during operation impacts the reactivity of fuel being discharged to the pool. 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. Therefore, utilizing a higher fuel temperature is more conservative.

The parameters important to determining fuel temperature are power level, moderator temperature, and coolant flow rate. [

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[4] EPU Cycles X, Y, and Z are possible future equilibrium fuel cycles created during EPU fuel management representing several different lifetime/feed fuel assumptions. Therefore these are included as representative of what soluble boron concentrations might look like after the EPU transition cycles (Cycles 28-30).

3.3.2.3 Operating History and Specific Power

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3.3.3 Axial Profile Selection

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

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3.3.3.1 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 5.1.

As discussed in Reference 11, 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 carries 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 as such 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 and so 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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3.3.3.2 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 5.1.

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 7. An appropriate

moderator temperature ensures the impact of moderator density on the neutron spectral effects is bounded, building in a conservative amount of plutonium.

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3.3.3.3 Axial Burnup and Temperature Profiles

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Table 3-9 Axial Burnup and Temperature Profiles

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3.3.4 Burnable Poison Usage

Burnable poison usage at Prairie Island has been considered for this analysis and conservative assumptions have been utilized to bound the effects of burnable poisons on fuel isotopics. The burnable poisons that have seen use at Prairie Island include Burnable Poison Rod Assemblies (BPRs) and Gd_2O_3 Gadolinia (gad) integrated into fuel pins.

BPRs are boron-impregnated glass inserts that are inserted into assembly guide tubes to control power distribution by suppressing an assembly's reactivity during operation. BPRs can impact assembly reactivity after discharge. These inserts contain boron, which is a strong thermal neutron absorber, hardening the neutron spectrum of the assembly the BPR has been placed in. Additionally, BPRs displace

[5] Note that the 0-18 GWd/MTU shape normalizes to less than 1.0, providing additional conservatism.

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. BPR parameters are shown in Table 3-10.

Table 3-10 BPR Specifications

Parameter	BPR
Burnable poison material	Borosilicate Glass
Burnable poison ID, inch	0.2440
Burnable poison OD, inch	0.3890
Burnable poison clad material	SS
Burnable poison inner clad thickness, inch	0.0065
Burnable poison inner clad OD, inch	0.2365
Burnable poison outer clad thickness, inch	0.0188
Burnable poison outer clad OD, inch	0.4310
Burnable poison length, inch	142.7

At Prairie Island, BPRs were only used in the first four cycles of each unit when the plant operated using lower enrichment fuel. A review of past fuel management shows that the maximum enrichment of any assembly that contained a BPR is 3.4 wt% ^{235}U. [

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Prairie Island has transitioned from using BPRs to using gadolinia for power distribution control. As discussed in Reference 9, gadolinia is loaded as an integral part of the fuel matrix, forming rods containing UO_2-Gd_2O_3. The weight percent or loading of Gd_2O_3 in each gadolinia-bearing rod and the number of gadolinia rods within an assembly are both variable.

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3.3.5 Fuel Depletion Modeling Assumptions

The PARAGON code (Section 2.3.1) was used to develop the depleted fuel isotopics used in this analysis. To perform the depletion calculations, PARAGON used the depletion parameters and axial profiles discussed above to ensure that the isotopic number densities generated would be more conservative (reactive) than that of the fuel being discharged from the reactors at Prairie Island. Table 3-11 gives the nominal operating parameters for the Prairie Island reactors both pre and post proposed EPU[6] as well as the values assumed for the depletion calculations supporting this analysis.

Table 3-11 Parameters Used in Depletion Analysis

	Nominal Core Parameters		Depletion Analysis
Parameter	**Value (Pre-EPU)**	**Value (proposed EPU)**	**Value (Analyzed)**
Maximum soluble boron concentration, ppm	845	826	[ ]a,c
Rated thermal power, MWt	1683	1811	1811
Average assembly power, MWt	13.91	14.97	[ ]a,c
Core outlet moderator temperature, °F	602.8	610.6	[ ]a,c As described in Section 3.3.3.3
Core inlet moderator temperature, °F	535.5	541.2	[ ]a,c As described in Section 3.3.3.3
Minimum RCS flow rate (Thermal Design Flow), gpm	178,000	187,000	[ ]a,c
Fuel designs	Westinghouse STD, OFA & ENC STD, TOPROD & HIGHBURNUP	422V+	[ ]a,c
Burnable Poison[7]	BPR or gadolinia	gadolinia	[ ]a,c

[6] Note that in addition to the EPU, Prairie Island has instituted a measurement uncertainty recapture (MUR) uprate. The MUR uprate is bounded by the EPU conditions and so is not shown in the table.

[7] Two sets of isotopics were developed, one set included a 16 finger BPR and the other did not, all other depletion parameters are the same between the two sets of isotopics. The BPR isotopics are used to represent the fuel operated in Cycles 1-4.

3.4 RODDED OPERATION[8]

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As part of this analysis, past rodded operation at Prairie Island was considered. Past cycles were reviewed and the impact of rodded operation on both shape and neutron spectrum were studied. The conclusions of this analysis can be found in Section 5.2.

3.5 CREDIT FOR RCCAS

Array G is the only array where a poison insert is being credited to reduce burnup limits. Table 3-12 lists dimensions for the full length RCCA control insert. [ ]a,c

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Table 3-12 RCCA Specifications	
Parameter	**Value**
Material	Silver-Indium-Cadmium
Silver content, wt%	80 [ ]a,c
Indium content, wt%	15 [ ]a,c
Cadmium content, wt%	5 [ ]a,c
Poison OD, inch	0.3905 [ ]a,c
Clad ID, inch	0.3934 [ ]a,c
Clad OD, inch	0.4310 [ ]a,c
Clad material	SS
Poison density, gm/cm^3	10.17

3.6 NORMAL CONDITION DESCRIPTION

This section discusses normal conditions within the SFP besides the steady-state storage of fresh and spent assemblies. During normal operation, the SFP has a soluble boron concentration of greater than 1800 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:

3.6.1 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 BPRs.

3.6.2 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. Examples of these procedures include fuel assembly cleaning, inspection, reconstitution, and sipping.

3.6.3 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 and damaged fuel rod baskets, fuel rod consolidation storage canisters, movable in-core detectors, and miscellaneous maintenance equipment. Further descriptions of both consolidated and failed fuel storage are given below.

3.6.3.1 Consolidated Rod Storage Description

Rod consolidation is a process in which the fuel rods are removed from the fuel assembly structural material and packed into a storage can. Typically two assemblies worth of fuel rods are packed in a storage can in a close-packed triangular rod pitch as shown in Figure 3-1. The storage can is approximately the size of the fuel assembly and is placed in a fuel rack storage cell. The fuel assembly structural material is sheared into small pieces after the nozzles have been removed.

To optimize fuel storage at Prairie Island, consolidated rod storage was implemented as a trial. These consolidated rod storage canisters have been stored in a 2x2 array as shown in Section 3.7.1. The cells face-adjacent to the consolidated rod storage canisters contain the assembly structural components from the consolidation process.

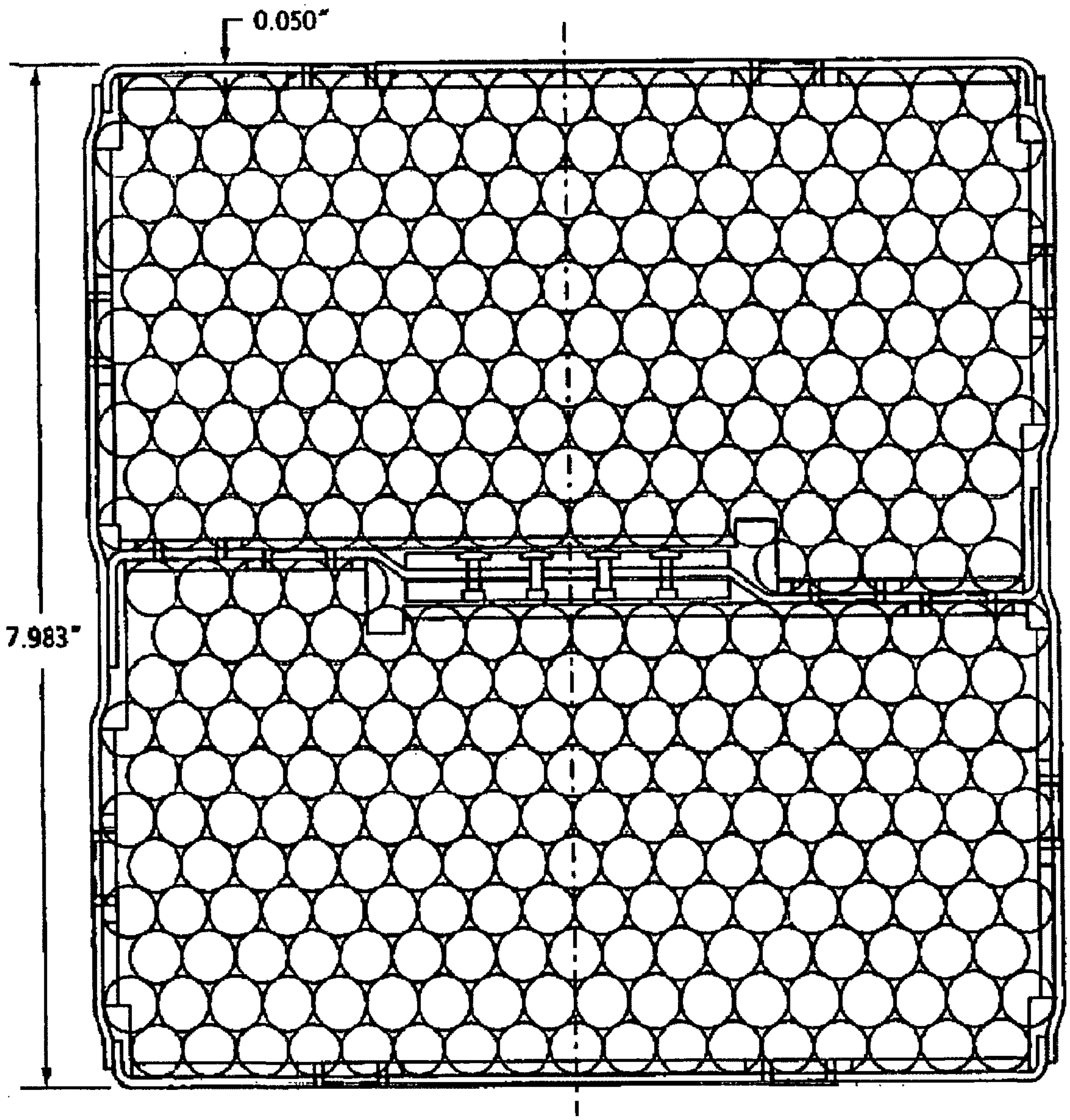

Figure 3-1 Consolidated Rod Storage Canister

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Table 3-13 Consolidated Rod Storage Canister Analysis

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3.6.3.2 Failed Fuel Basket Description

This section of the analysis discusses the assumptions made with regard to modeling the failed fuel baskets and analysis of the acceptable storage locations of these baskets. There are two different designs of failed fuel basket; failed fuel pin basket and fuel rod storage canister are shown as Figure 3-2 and Figure 3-3. Table 3-14 and Table 3-15 provide the nominal and analyzed parameters. Note if a statement applies to only one of these two baskets, the name of the individual basket will be used. When referring to both the failed fuel pin basket and the fuel rod storage canister at the same time, the term failed fuel basket will be used.

[9] Not using burnable poison is conservative because the analysis is performed with fresh fuel so there are no competing effects (spectral hardening, etc) which mitigate the poison's reactivity suppression.

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Figure 3-3 Fuel Rod Storage Canister

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Figure 3-4 Fuel Rod Storage Canister Cross Section

Table 3-15 Fuel Rod Storage Canister Specifications	
Parameter	**Value**
Small rod sleeve array size	8x8 – 4x4
Rod pitch, inch	.875
Small fuel rod sleeve OD, inch	0.755
Small fuel rod sleeve ID, inch	0.625
Total number of fuel rods in the small fuel rod sleeves	48
Large rod sleeve array size	3x3
Rod pitch, inch	1.00
Large fuel rod sleeve OD, inch	0.880
Large fuel rod sleeve ID, inch	0.750
Total number of fuel rods in the large fuel rod sleeves	9
Fuel rod storage canister inner dimension, inch	7.625
Active fuel length, inch	144
Stack density, % TD	[ ]a,c
Maximum enrichment, wt% ^{235}U	5.0[11]

3.6.4 Type 4 Normal Conditions

Type 4 conditions involve temporary installation of components on the rack periphery. Examples include fuel assembly inspection and maintenance platforms, fuel sipping cans, and dry fuel storage casks.

3.6.5 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.

3.7 KENO MODELING ASSUMPTIONS

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The methods used to model the fuel, consolidated fuel, and failed fuel basket storage arrays, as well as accident scenarios are discussed in the following sections.

Discussion of the fuel assembly and storage array assumptions can be found in Sections 3.1.1 and 3.2.1 respectively. [

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3.7.1 Array Descriptions

Descriptions of the fuel storage arrays allowable for use at Prairie Island are outlined below. Each storage array was modeled in KENO as an infinite series of arrays so that an infinite array was modeled.

The restrictions associated with arrays can be found in Section 5.1.

Array A

Fuel Category 6 assembly in every cell.

6	6
6	6

Array B

Fuel Category 3 assembly in three of every four cells; one of every four cells is empty (water filled).

3	3
3	X

Array C

Checkerboard pattern of Fuel Category 1 assemblies and empty (water-filled) cells.

1	X
X	1

Array D

Two Fuel Category 5 assemblies, one Fuel Category 1 assembly, and one empty cell in every four cells. The Fuel Category 1 and empty cell shall be face-adjacent.

5	5
X	1

Array E

2	4
X	2

Two Fuel Category 2 assemblies, one Fuel Category 4 assembly, and one empty cell in every four cells. The Fuel Category 4 assembly shall be diagonal to the empty cell.

Array F

7	NF
NF	7

Checkerboard pattern of Fuel Category 7 consolidated rod storage canisters and non-fuel cells.

Array G

5	5	5
5	5R	5
5	5	5

Nine Fuel Category 5 assemblies in every nine cells with a full length RCCA loaded in the center assembly.

Notes:

1. In all arrays, an assembly of lower reactivity may replace an assembly of higher reactivity.
2. Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
3. Fuel Categories 2 through 6 are determined from the coefficients provided in Section 5.1.
4. Fuel Category 7 is fuel stored in the consolidated rod storage canisters. Fuel Category 7 fuel may not be substituted for other fuel categories.
5. An **X** indicates an empty (water-filled) cell.
6. An **R** indicates the assembly must contain a full length RCCA.
7. Attributes for each array are as stated in the definition. Diagram is for illustrative purposes only.
8. An empty (water-filled) cell may be substituted for any fuel containing cell in any storage array.
9. An **NF** indicates a non-fuel cell. These cells may contain assembly structural materials, nozzles, grids, or other non-fuel components.

Figure 3-5 Allowable Storage Arrays

3.8 ACCIDENT DESCRIPTION

The following reactivity increasing accidents are considered in this analysis:

- Misloaded fresh fuel assembly into incorrect storage rack location
- Inadvertent removal of an RCCA
- SFP temperature greater than normal operating range (150 °F)
- Dropped & misplaced fresh fuel assembly

The inputs to the accident analysis are the results of the burnup limit calculations discussed in Section 4.3.

4 ANALYSIS DESCRIPTION & CALCULATIONS

This section describes the calculations that were done to support the results and conclusions provided in Section 5. This section discusses the methods used to determine the target k_{eff}, evaluate the impact of soluble boron on the analyzed configurations, and evaluate the impact of accidents in the SFP.

4.1 BURNUP LIMIT GENERATION

To ensure the safe operation of the Prairie Island 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 (MWd/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 3.3). 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 4.1.1 to account for manufacturing deviations.

4.1.1 Target k_{eff} Calculation Description

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4.1.2 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}. Examples include the pool temperature and code validation biases. Uncertainties account for allowable variations within the real model whether they are physical (manufacturing tolerances), analytical (depletion and fission product uncertainties) or measurement related (burnup measurement uncertainty).

4.1.2.1 Bias & Uncertainty Descriptions

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

4.1.2.1.1 Manufacturing Tolerances

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4.1.2.1.2 Burnup Measurement Uncertainty

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4.1.2.1.3 Depletion Uncertainty

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4.1.2.1.4 Fission Product Worth Uncertainty

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4.1.2.1.5 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 3.3.2.3 and Reference 7.

4.1.2.1.6 Eccentric Fuel Assembly Positioning

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Figure 4-1 Fuel Assembly Eccentric Positioning

4.1.2.1.7 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.

4.1.2.1.8 Pool Temperature Bias

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4.1.2.1.9 Borated and Unborated Biases and Uncertainties

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4.1.2.2 Storage Array Biases & Uncertainties Results

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The biases and uncertainties for each fuel storage array are given in Section 4.1.2.2.1 and Section 4.1.2.2.2 for fuel not operated during Cycles 1-4 and fuel operated during Cycles 1-4 respectively.

4.1.2.2.1 Storage Array Biases & Uncertainties for Fuel Not Operated During Cycles 1-4

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Table 4-1 Biases & Uncertainties for Array A: Fuel Not Operated During Cycles 1-4

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Table 4-2 Biases & Uncertainties for Array B: Fuel Not Operated During Cycles 1-4

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Table 4-3 Biases & Uncertainties for Array C

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Table 4-4 Biases & Uncertainties for Array D: Fuel Not Operated During Cycles 1-4

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Table 4-5 Biases & Uncertainties for Array E: Fuel Not Operated During Cycles 1-4

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Table 4-6 Biases & Uncertainties for Array G: Fuel Not Operated During Cycles 1-4

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4.1.2.2.2 Biases & Uncertainties for Fuel Operated During Cycles 1-4

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Table 4-7 Biases & Uncertainties for Array A: Fuel Operated During Cycles 1-4

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Table 4-8 Biases & Uncertainties for Array B: Fuel Operated During Cycles 1-4

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Table 4-9 Biases & Uncertainties for Array D: Fuel Operated During Cycles 1-4

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Table 4-10 Biases & Uncertainties for Array G: Fuel Operated During Cycles 1-4

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4.1.2.3 Consolidated Rod Storage Canister Biases & Uncertainties Results

The consolidated rod storage canister array has had explicit biases & uncertainties calculated for it. [

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Table 4-11 Biases & Uncertainties for Array F

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4.1.2.4 Failed Fuel Basket Biases & Uncertainties

No biases and uncertainties were calculated for the failed fuel baskets. The failed fuel baskets are significantly less reactive than a fuel assembly. Incorporating specifically calculated biases and uncertainties for the failed fuel baskets will not impact the results discussed in Section 5.4.3.

4.2 SOLUBLE BORON CREDIT

In this analysis, boron credit calculations assume natural boron, which is 19.9 at% ^{10}B. However, because the isotopic concentration of boron can vary, this analysis also accounts for boron isotopic concentrations as low as 19.4 at% ^{10}B.

The minimum soluble boron concentration in the Prairie Island SFP is assumed in this analysis to be 400 ppm, the value in proposed Technical Specification 4.3.1 associated with achieving $k_{eff} \leq 0.95$ for the limiting normal condition including biases, uncertainties, and administrative margin. For conservatism, this analysis ensures that a k_{eff} of 0.95 will be maintained assuming 400 ppm of soluble boron at 19.4 at% ^{10}B (389 ppm at 19.9 at% ^{10}B).

Table 4-12 presents the maximum k_{eff} values for normal conditions including biases and uncertainties at a boron concentration of 389 ppm at 19.9 at% ^{10}B (400 ppm at 19.4 at% ^{10}B). It is demonstrated that 389 ppm at 19.9 at% ^{10}B (400 ppm at 19.4 at% ^{10}B) is sufficient to comply with the acceptance criterion of $k_{eff} \leq 0.95$ under all normal conditions (see Table 4-12). Additionally, Table 4-12 identifies the limiting storage array and the soluble boron concentration required to meet a $k_{eff} \leq 0.95$.

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Table 4-12 [ ]^{a,c}

4.3 RODDED OPERATION

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[1] Calculations were performed at 389 ppm at 19.9 at% ^{10}B which equates to 400 ppm at 19.4 at% ^{10}B.

[2] The k_{eff}s provided in Table 4-12 should be compared to 0.945, as 0.005 Δk administrative margin is not included.

[3] Review of historical operation shows that for fuel not operated in Cycles 1-4, the maximum amount of rodded operation is less than 600 MWd/MTU of core average operation.

4.4 NORMAL CONDITIONS

4.4.1 Type 1 Normal Conditions

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Therefore, any components designed to be inserted into an assembly may be stored in a fuel assembly in the SFP.

4.4.2 Type 2 Normal Conditions

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 operations 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 of undamaged rods from a damaged assembly for insertion in a new assembly. In most cases, damaged rods will be replaced with stainless steel rods, but 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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4.4.3 Type 3 Normal Conditions

Type 3 normal conditions include the insertion of components other than intact fuel assemblies into storage cells. 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 3.6.3.1. Items containing fissile material and items without fissile material which can be stored differently than a fuel assembly are described below.

The consolidated fuel storage canister has been evaluated as Fuel Category 7 fuel in Array F (see Section 3.7.1). A target k_{eff} was calculated based on the biases and uncertainties given in Section 4.1.2.3. Array F was shown to meet the required target k_{eff} and therefore consolidated fuel storage canisters can be stored in this array. Additionally, an interface-specific analysis was performed which confirmed the acceptability of an interface between Array A and Array F. [

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In addition to the consolidated fuel storage canisters, Array F includes non-fuel cells that contain damaged or compacted fuel structural components. These have been analyzed as a part of Array F as described in Section 3.6.3.1. This analysis allows for the structural material (excluding top and bottom nozzles) of up to three assemblies to be stored within a non-fuel cell for Array F. Additionally, in Array F, it is acceptable to stack any number of top or bottom nozzles in the non-fuel cell.

Failed fuel baskets, unlike fuel assemblies, are not designed to maximize reactivity. Therefore, the baskets are significantly less reactive than a fuel assembly and as a result can be stored in any cell location where a fuel assembly is permitted. [

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Table 4-13 Failed Fuel Pin Basket Analysis

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Table 4-14 Fuel Rod Storage Canister Analysis

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As shown in Table 4-13 and Table 4-14, the failed fuel baskets are significantly less reactive than the least reactive fuel assemblies credited in this analysis (assemblies that meet the requirements of Fuel Category 6).

A damaged fuel rod storage basket is a consolidated rod storage canister containing up to 10 fuel rods too damaged to be inserted into either failed fuel basket. [

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The storage of a canister with movable incore detectors has been evaluated for storage in the SFP at Prairie Island. This canister holds both the movable incore detectors and the associated thimble tubes. Together these detectors do not possess enough fissile material to impact reactivity significantly and can be placed in any location where a fuel assembly of any fuel category may be stored.

A filter assembly is defined as a single storage cell up to completely full of cotton or polyester fibers and minor steel and plastic structural components. These filters may contain very small discrete particles of fissile material in the filter media. A filter assembly cannot contain enough fissile material to go critical and can be stored in any location where a fuel assembly of any fuel category may be stored.

4.4.4 Type 4 Normal Conditions

Type 4 normal conditions include temporary installation of non-fissile components on the rack periphery. Analysis 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.

4.4.5 Type 5 Normal Conditions

Type 5 normal conditions include miscellaneous activities not fully aligned with the first four types of normal operations.

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 needed to be stored as a fuel assembly (i.e. failed fuel basket, 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 prospective, all fuel handling tools are acceptable for their intended purpose.

4.5 ACCIDENTS

4.5.1 Assembly Misload into the Storage Racks

The misloaded fresh fuel accident scenario is analyzed by placing a 5.0 wt% ^{235}U fresh fuel assembly into the water-filled cell required in Arrays B, C, D, and E. A 6x6 model is utilized with periodic boundary conditions containing one misloaded fresh fuel assembly. Each array that credits a water-filled cell was analyzed; a misload into Array D is the most limiting. This accident requires 887 ppm of boron to maintain k_{eff} less than 0.95, (910 ppm assuming 19.4 at% ^{10}B) including biases and uncertainties. The results of Array D calculation is presented in Table 4-15.

4.5.2 Inadvertent Removal of an RCCA

This accident accounts for the removal of an RCCA from Array G. This accident has not been explicitly analyzed because it is bounded by the assembly misload accident discussed in Section 4.5.1. The insertion of a fresh 5.0 wt% ^{235}U assembly into Array D is the limiting misload. This creates an accident condition with two fresh 5 wt% ^{235}U assemblies, and two Fuel Category 5 assemblies in a 2x2 storage array. This condition is clearly bounding of the withdrawn of an RCCA from Array G, which creates an array containing solely Fuel Category 5 assemblies.

4.5.3 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. Due to the large volume of water in the SFP, boiling off of the pool water before remediation is not credible; therefore the lowest density of the water is the water density at boiling and atmospheric pressure, 0.96 gm/cm^3. Calculations are run with voiding and 887 ppm of soluble boron. To demonstrate conservatism, an additional case with a moderator density of 0.75 gm/cm^3 is performed. The results for these calculations are presented in Table 4-15.

4.5.4 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. In this case, there is significant separation between the dropped fuel assembly and the active regions of the rest of the fuel assemblies due to the top nozzle, fuel rod plenum, fuel rod end plug, and the separation between the fuel rod and the top nozzle. Due to the large physical separation between the active region of the fuel and the dropped assembly it is clear that the misloaded fresh fuel assembly described above is more limiting than a single assembly lying horizontally on top of other assemblies in the rack.

It is possible to misplace a fuel assembly in a location not intended for fuel. Any assembly placed outside of the racks is surrounded by water on at least two sides. The misloaded fresh assembly discussed above is surrounded by fuel on all four sides. The additional neutron leakage of the two sides not facing fuel ensures that this condition is bounded by the misload event.

4.5.4.1.1 Accident Results

Table 4-15 Results of the Accident Calculations at 887 ppm

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[5] The k_{eff}s provided in Table 4-15 should be compared to 0.945, as 0.005 Δk administrative margin is not included

5 ANALYSIS RESULTS

This section documents the results of the Prairie Island 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 restrictions placed on the various storage arrays such as placement of non-fuel items and an evaluation of normal SFP activities which are bounded by this analysis.

5.1 BURNUP LIMITS & RESTRICTIONS ON STORAGE ARRAYS

Assembly storage is controlled through the storage arrays defined in Section 3.7.1. An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity array. Fuel categories are defined by assembly burnup, enrichment and decay time as provided by Table 5-2 through Table 5-17.

Table 5-1 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
Fuel Category 7	Consolidated Rod Storage Canister Category

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. Fuel Category 1 is fuel up to 5.0 wt% ^{235}U; no burnup is required.
3. Fuel Categories 2 through 6 are determined from the coefficients provided.
4. Fuel Category 7 applies only to consolidated rod storage canisters and is not ranked relative to the other fuel categories. Fuel Category 7 fuel may not be substituted for other fuel categories.

5.1.1 Requirements for Fuel Not Operated During Cycles 1-4

Table 5-2 Fuel Category 2: Burnup Requirement Coefficients, Fuel Not Operated During Cycles 1-4

Decay Time (yr)	Coefficients			
	A_1	A_2	A_3	A_4
0	-0.669	9.018	-32.080	33.507

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 2.9 wt% ^{235}U and 5.0 wt% ^{235}U may be used.

Table 5-3 Fuel Category 2: Burnup Requirements (GWd/MTU), Fuel Not Operated During Cycles 1-4

Decay Time (yr)	Wt% ^{235}U				
	2.90	3.40	4.00	4.50	5.00
0	0	2.389	6.659	10.799	14.932

Notes:

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

Table 5-4 Fuel Category 3: Burnup Requirement Coefficients, Fuel Not Operated During Cycles 1-4				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	-0.120	1.300	5.006	-18.765
5	-0.167	1.766	3.085	-16.141
10	-0.218	2.249	1.405	-14.163
15	-0.281	2.949	-1.267	-10.873
20	-0.401	4.237	-5.881	-5.513

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 2.5 wt% ^{235}U and 5.0 wt% ^{235}U may be used.
3. Linear interpolation between decay times is permitted. However, an assembly with a decay time greater than 20 years must use the 20 years limits.

Table 5-5 Fuel Category 3: Burnup Requirements (GWd/MTU), Fuel Not Operated During Cycles 1-4					
Decay Time (yr)	**Wt% ^{235}U**				
	2.50	**3.40**	**4.00**	**4.50**	**5.00**
0	0	8.567	14.379	19.152	23.765
5	0	8.199	13.767	18.285	22.559
10	0	8.044	13.489	17.837	21.837
15	0	7.865	13.259	17.537	21.392
20	0	7.710	13.091	17.281	20.882

Notes:

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

Table 5-6 Fuel Category 4: Burnup Requirement Coefficients, Fuel Not Operated During Cycles 1-4				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	1.355	-14.866	62.715	-72.624

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 1.8 wt% ^{235}U and 5.0 wt% ^{235}U may be used.

Table 5-7 Fuel Category 4: Burnup Requirements (GWd/MTU), Fuel Not Operated During Cycles 1-4					
Decay Time (yr)	**Wt% ^{235}U**				
	1.80	**3.40**	**4.00**	**4.50**	**5.00**
0	0	22.013	27.100	32.031	38.676

Notes:

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

Table 5-8 Fuel Category 5: Burnup Requirement Coefficients, Fuel Not Operated During Cycles 1-4				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.569	-6.563	37.088	-47.854
5	0.302	-3.795	27.410	-37.964
10	0.151	-2.248	21.874	-32.204
15	-0.198	1.133	11.031	-21.713
20	-0.427	3.424	3.614	-14.522

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 1.75 wt% ^{235}U and 5.0 wt% ^{235}U may be used.
3. Linear interpolation between decay times is permitted. However, an assembly with a decay time greater than 20 years must use the 20 years limits.

Table 5-9 Fuel Category 5: Burnup Requirements (GWd/MTU), Fuel Not Operated During Cycles 1-4					
Decay Time (yr)	**Wt% ^{235}U**				
	1.75	**3.40**	**4.00**	**4.50**	**5.00**
0	0	24.741	31.906	37.991	44.636
5	0	23.230	30.284	36.052	41.961
10	0	22.116	28.988	34.467	39.841
15	0	21.108	27.867	32.827	37.017
20	0	20.564	27.390	32.167	35.773

Notes:

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

Table 5-10 Fuel Category 6: Burnup Requirement Coefficients, Fuel Not Operated During Cycles 1-4				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.567	-6.205	35.936	-45.944
5	0.923	-9.720	45.538	-53.858
10	0.728	-7.992	40.264	-48.929
15	0.343	-4.016	27.236	-36.380
20	0.283	-3.391	24.925	-33.963

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 1.7 wt% ^{235}U and 5.0 wt% ^{235}U may be used.
3. Linear interpolation between decay times is permitted. However, an assembly with a decay time greater than 20 years must use the 20 years limits.

Table 5-11 Fuel Category 6: Burnup Requirements (GWd/MTU), Fuel Not Operated During Cycles 1-4					
Decay Time (yr)	**Wt% ^{235}U**				
	1.70	**3.40**	**4.00**	**4.50**	**5.00**
0	0	26.794	34.808	41.785	49.486
5	0	24.886	31.846	38.341	46.207
10	0	24.194	30.847	36.760	43.591
15	0	23.279	30.260	36.114	42.275
20	0	22.705	29.593	35.320	41.262

Notes:

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

5.1.2 Burnup Requirements for Fuel Operated during Cycles 1-4

Table 5-12 Fuel Category 3: Burnup Requirement Coefficients, Fuel Operated During Cycles 1-4

Decay Time (yr)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.000	-0.722	14.272	-31.167
20	0.000	-1.944	20.494	-39.085

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 2.5 wt% ^{235}U and 3.4 wt% ^{235}U may be used.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 5-13 Fuel Category 3: Burnup Requirements (GWd/MTU), Fuel Operated During Cycles 1-4

Decay Time (yr)	Wt% ^{235}U		
	2.50	**3.00**	**3.40**
0	0	5.151	9.011
20	0	4.901	8.122

Notes:

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

Table 5-14 Fuel Category 5: Burnup Requirement Coefficients, Fuel Operated During Cycles 1-4				
Decay Time (yr)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.673	-8.242	44.607	-56.428
20	1.784	-16.297	60.035	-64.713

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 1.75 wt% ^{235}U and 3.4 wt% ^{235}U may be used.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 5-15 Fuel Category 5: Burnup Requirements (GWd/MTU) , Fuel Operated during Cycles 1-4				
Decay Time (yr)	**Wt% ^{235}U**			
	1.75	**2.50**	**3.00**	**3.40**
0	0	14.093	21.386	26.410
20	0	11.393	16.887	21.131

Notes:

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

Table 5-16 Fuel Category 6: Burnup Requirement Coefficients, Fuel Operated During Cycles 1-4				
Decay Time (yr)	Coefficients			
	A_1	A_2	A_3	A_4
0	1.097	-10.246	47.457	-56.456
20	1.820	-15.656	56.856	-60.351

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 1.7 wt% ^{235}U and 3.4 wt% ^{235}U may be used.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 5-17 Fuel Category 6: Burnup Requirements (GWd/MTU) , Fuel Operated during Cycles 1-4				
Decay Time (yr)	Wt% ^{235}U			
	1.70	2.50	3.00	3.40
0	0	15.290	23.320	29.571
20	0	12.377	18.453	23.509

Notes:

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

5.1.3 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.

5.2 RODDED OPERATION

Prairie Island has experienced load follow operation in the past as evaluated in Section 4.3. The study of past rodded operation has shown that those fuel assemblies which experienced rodded operation are bounded by the design basis assemblies used to develop the burnup requirements defined in Section 5.1.

For fuel not operated during Cycles 1-4, an assembly in a rodded location can be stored utilizing the burnup requirements provided in Section 5.1, provided that the assembly has been exposed to rodded operation for no more than 1 GWd/MTU of core average burnup. If an assembly experiences more than 1 GWd/MTU of core average rodded operation, the assembly shall either be treated as Fuel Category 1 or evaluated to determine which Fuel Category is appropriate for safe storage of the assembly.

5.3 INTERFACE CONDITIONS

The Prairie Island SFP uses racks that are all of the same physical configuration. Therefore, the only interface conditions that need to be addressed in this analysis are those between different fuel storage arrays. To ensure that the interface between fuel storage arrays is bounded by this analysis, a restriction requiring that each fuel assembly loaded in the SFP meets the requirements of a storage array, is placed upon fuel loading.

The requirement to check all possible arrays considers each storage cell as part of up to four arrays; one each where the cell is the top left, top right, bottom left and bottom right cell of the array. All four of these arrangements must match one of the acceptable storage arrays, but all arrangements will not necessarily match the same storage array. The application of this requirement will automatically establish rows or columns of boundary cells at the interface, if required. These boundary cells conservatively fulfill the requirements of cells on each side of the interface.

Following these rules ensures that every assembly is loaded in arrays that match an analyzed condition, and therefore no interface-specific analyses are required except for the Array A – Array F interface discussed in Section 4.4.3. Gaps between rack modules are conservatively neglected, i.e., cells located across a rack-to-rack gap are considered the same as cells directly facing each other within a rack.

5.4 NORMAL CONDITIONS

The purpose of this section is to summarize the conclusions of the evaluation of normal SFP operations performed in Section 4.3.

5.4.1 Type 1 Normal Conditions

Any component designed to reside in the guide tubes of a fuel assembly can be stored in an assembly's guide tubes in the SFP.

5.4.2 Type 2 Normal Conditions

Fuel assembly evolutions (fuel cleaning, inspection, reconstitution, and sipping) occurring outside of the storage racks is acceptable while there is at least one assembly pitch of water between the assembly in question, other assemblies and the storage racks. It is also acceptable to perform these actions above the top of the storage racks.

5.4.3 Type 3 Normal Conditions

Components in the spent fuel pool which do not contain any fissile material can be stored in any cell that would allow the storage of a fuel assembly of any fuel category. This analysis has confirmed the acceptable storage of the following non-fuel assembly components:

- Consolidated fuel can be stored in Array F as Fuel Category 7 fuel;
- Failed fuel baskets, damaged fuel storage baskets, and filter assemblies can be stored in any location where fuel assemblies can be stored;
- A non-fuel cell of Array F may contain damaged or compacted fuel assembly components provided that it is either less than three assemblies worth of structural material (excluding top and bottom nozzles), stacks of any number of top nozzles or bottom nozzles, or a cell can instead contain up to 50% (by volume) of components made of SS-304; and
- There are movable incore detectors being stored in the SFP together with their associated thimble tubes. These movable incore detectors may be stored together in any location where a fuel assembly can be stored.

5.4.4 Type 4 Normal Conditions

The placement of any non-fissile components or structures along the edge or top of the rack is acceptable.

5.4.5 Type 5 Normal Conditions

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.

Fuel handling tools may be used for their individual intended purposes without impacting the results and conclusions of this analysis.

5.5 SOLUBLE BORON CREDIT

Soluble boron is credited in the Prairie Island 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, 910 ppm of soluble boron is required to ensure $k_{eff} \leq 0.95$, this leaves significant margin to the required Technical Specification value of 1800 ppm.

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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 code system (Reference A1).

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The methodology validated in this section is based on the use of the CSAS25, NITAWL-II, BONAMI, and KENO V.a (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 or cask loading and other normal operations in the SFP. The validation is adequate to cover all present and anticipated (non-mixed-oxide) LWR fuel designs. The area of applicability is discussed in more detail in subsection A.5.4.

A.2 CALCULATIONAL METHOD

The analysis methodology employs SCALE Version 5.1, as documented in Reference A1, using a 44-group Evaluated Nuclear Data File Version 5 (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-II, and KENO.

Standard material compositions are employed in the SCALE analyses consistent with the material descriptions in References A3 through A8.

All calculations are performed on systems with the following hardware and 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

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

A.3.1 Determination of Bias and Bias Uncertainty

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A.3.2 Test for Normal Distribution

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A.3.3 Identify Trends in the Data

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A.4 CRITICAL EXPERIMENT DESCRIPTION

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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

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

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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-1 [ ]a,c

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

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Figure A-3 [ ]a,c

Table A-8 [ ]a,c

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Figure A-4 [ ]^{a,c}

Table A-9 [ ]^{a,c}

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Figure A-5 [ ]a,c

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Figure A-6 [ ]a,c

Table A-11 [ ]a,c

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Table A-12 [ ]a,c

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Figure A-8 [

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Figure A-10 [ ]^{a,c}

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Table A-18 [ ][a,c]

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Figure A-11 [ ]a,c

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Figure A-12 [ ]a,c

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Figure A-13 [ ]a,c

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Figure A-14 [ ]a,c

Table A-24 [ ]a,c

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Figure A-15 [ ]a,c

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Figure A-16 [ ]a,c

Table A-26 [ ]a,c

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Table A-27 [
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Table A-28 [ ]a,c

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A.5.3.4 Summary of Trend Results

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Table A-30 [ ]^{a,c}

A.5.4 Non-Parametric Treatment

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Table A-31 [ ]a,c

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A.5.5 Area of Applicability Definition

The area of applicability (AOA) of this benchmark is defined by the range of parameters in the validation suite. Table A-32 summarizes the Areas of Applicability.

Table A-32 Area of Applicability

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A.6 SUMMARY OF THE BIAS AND BIAS UNCERTAINTY

Table A-33 summarizes the results of the validation.

Table A-33 Summary of Bias and Bias Uncertainty

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