ENCLOSURE

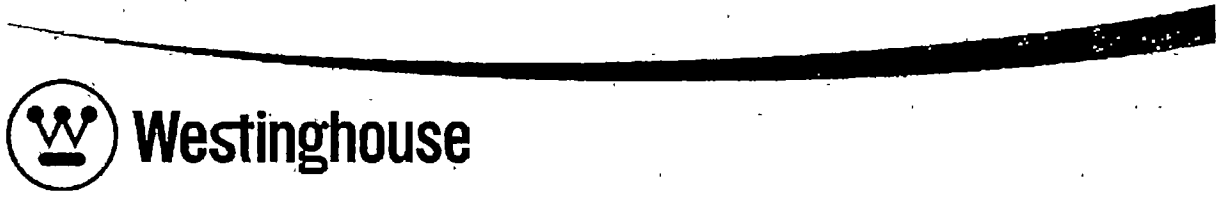
Attachment 5

WCAP-18414-NP
J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis
(Non-proprietary Version)

WCAP-18414-NP
Revision 0

September 2019

J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis



Westinghouse

WCAP-18414-NP
Revision 0

J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis

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September 2019

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REVISION HISTORY

Revision	Description and Impact of the Change	Date
0	Original Issue	09/2019

TRADEMARK NOTICE

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LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS

1-D	One-Dimensional
2-D	Two-Dimensional
3-D	Three-Dimensional
AEG	Average Energy Group of Neutrons Causing Fission
AoA	Area of Applicability
at%	Atom Percent
B&U	Sum of Biases and Uncertainties
B&W	Babcock and Wilcox
BA	Burnable Absorber
BONAMI	Bondarenko AMPX Interpolator
Boraflex	Neutron Absorber Material Comprised of Silicone Polymer and Boron Carbide Powder
C.E.A.	Commissariat à l'Energie Atomique et aux Energies Alternatives
Decay time	Post-irradiation cooling time
EALF	Energy of Average Lethargy causing Fission
En	Enrichment
ENDF/B	Evaluated Nuclear Data File
EPRI	Electric Power Research Institute
FHE	Fuel Handling Equipment
FOSAR	Foreign Object Search and Retrieval
FRSC	Fuel Rod Storage Canister
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
LWR	Light Water Reactor
MTU	Metric Ton Uranium
MWt	Megawatts-thermal
NPM	Non-Parametric Margin
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Lab
Optimized ZIRLO	Optimized ZIRLO® High Performance Cladding Material
PNNL	Pacific Northwest National Laboratory
ppm	parts per million
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
SFP	Spent Fuel Pool
SRSC	Service de Recherche en Sûreté Criticité, now called Service de Recherche en Neutronique et Sûreté Criticité
SS	Stainless Steel

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

STD	Standard Fuel Assembly
TD	Percentage of Theoretical Density
WABA	Wet Annular Burnable Absorber
Westinghouse	Westinghouse Electric Company LLC
wt%	Weight Percent
ZIRLO	ZIRLO® High Performance Fuel Cladding Material

1 INTRODUCTION

The purpose of this report is to document the criticality safety analysis performed to support the operation of the J. M. Farley Nuclear Power Plant Units 1 and 2 (hereafter, Farley Units 1 & 2) spent fuel pools (SFPs). The report considers past, current, and planned future operating history and fuel design of Farley Units 1 & 2.

The main report details the SFP criticality safety analysis. Appendix A details the validation of the code used for pool eigenvalue calculations.

2 OVERVIEW

The existing SFP storage racks are evaluated for the placement of fuel within the storage arrays described in Section 5.2.1. Credit is taken for the negative reactivity associated with burnup and post-irradiation cooling time (decay time) for assemblies which have been operated in the reactor. Fuel assemblies which have not been operated in the reactor may take credit for the presence of zirconium diboride (IFBA) (hereafter referred to as IFBA). While the Farley Units 1 & 2 SFP storage racks may contain Boraflex® absorber inserts, no credit is taken for the presence of Boraflex absorber. Additionally, credit is taken for the presence of soluble boron in the SFPs.

2.1 ACCEPTANCE CRITERIA

This SFP criticality safety analysis ensures that the SFPs operate within the bounds 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 a margin for all applicable biases and uncertainties with 95 percent probability at a 95 percent confidence level.

2. The k_{eff} of all permissible storage arrangements when crediting soluble boron shall yield results not exceeding 0.95, including a margin for all applicable biases and uncertainties with 95 percent probability at a 95 percent confidence level.

3. The k_{eff} when crediting soluble boron shall not exceed 0.95 under all postulated accident conditions, including a margin for all applicable biases and uncertainties with 95 percent probability at a 95 percent confidence level.

2.2 DESIGN APPROACH

For the SFPs, compliance is demonstrated by establishing the minimum burnup requirements as a function of enrichment and decay time and minimum number of unirradiated IFBA rods as a function of enrichment for storage arrays A, B, C, and D seen in more detail in Figure 5-2. The fuel storage arrays have been analyzed to determine separate burnup requirements for two major fuel designs considered, the Standard Fuel Assembly (STD), and the Optimized Fuel Assembly (OFA). Note that the burnup requirements developed for the STD fuel design are applicable to the Robust Fuel Assembly (RFA) design since their neutronically important characteristics are the same.

A conservative combination of best estimate and bounding values have been selected as input for modeling in this analysis to ensure that fuel represented by the proposed Farley Units 1 & 2 SFP storage Technical Specifications is less reactive than the fuel modeled for this analysis. Therefore, burnup requirements generated here will conservatively bound all fuel to be stored in the Farley Units 1 & 2 SFPs.

The acceptability of the storage arrays developed in this analysis is ensured by controlling the assemblies that can be stored in each array. Assemblies are divided into Fuel Categories 1 through 4, and D (assemblies meeting the requirements of the damaged fuel array, Array D), based on assembly reactivity

determined as a function of assembly average burnup, initial enrichment1, IFBA loading2, and decay time. An assembly's fuel category determines in which storage arrays it may be stored. Fuel Category 1 defines the most reactive assemblies, i.e. fresh 5 weight percent (wt%) ^{235}U assemblies without IFBA and Fuel Category 4 defines the least reactive assemblies, i.e., representing low reactivity assemblies that can be stored in Array B (see Table 5-3).

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 Version 1.2.0, as documented in WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (Reference 1) and its cross-section library based on Evaluated Nuclear Data File Version VI.3 (ENDF/B-VI.3), and (2) Scale Version 6.2.3, as documented in ORNL/TM-2005/39, "Scale: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation" (Reference 2), with the ENDF/B-VII 238-group cross-section library.

2.3.1 Two-Dimensional Transport Code PARAGON

PARAGON is used in this application to simulate in-reactor fuel assembly depletion to generate isotopics for burnup credit. PARAGON is the Westinghouse Electric Company LLC state-of-the-art 2-D lattice transport code for pressurized water reactor (PWR) applications. It is part of the Westinghouse 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 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. The list of fuel isotopes modeled in PARAGON and subsequently modeled in the criticality analysis are given in Table 2-1.

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1. Initial enrichment is the enrichment of the central zone region of fuel, excluding axial cutbacks\blankets and 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 cutbacks or low enriched blankets, the maximum initial enrichment value is to be used.
2 IFBA loading restrictions only apply to fresh fuel being stored as Fuel Category 2.

Table 2-1 Isotopes Used in the Nuclear Criticality Safety Analysis

a,c

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Additional qualification of PARAGON for use in spent fuel pool applications has been performed at Westinghouse. The Electric Power Research Institute (EPRI) has developed PWR reactivity depletion benchmarks using a large set of measured flux data (flux maps) in EPRI Report 3002010613, "Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty" (Reference 3).

A guide for application of the EPRI depletion benchmarks for use in burnup credit calculations is given by way of example in Reference 4, with this methodology repeated by Westinghouse in "EPRI Depletion Benchmark Calculations Using PARAGON" (Reference 5). Results of this analysis provide additional confidence in the usage of PARAGON for SFP reactivity calculations, and provide a sound basis for usage of the 5% decrement approach for depletion uncertainty (See Section 5.2.3.1.5), showing that the depletion isotopics generated with PARAGON, and input into CSAS5 input models in Scale is conservative for determining depletion uncertainty.

PARAGON is generically approved for depletion calculations (Reference 1). 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.

2.3.2 Scale Code Package

The Scale system was developed for the U.S. Nuclear Regulatory Commission (NRC) to standardize the method of analysis for evaluation of nuclear fuel facilities and shipping package designs (Reference 2). 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 the SFPs, finite rack modules and SFP representations to evaluate interfaces, soluble boron requirements, and postulated accident scenarios to demonstrate that the requirements in Section 2.1 are met.

The Scale package includes the control module Criticality Safety Analysis Sequence with KENO V.a (CSAS5), which provides reliable and efficient means of performing k_{eff} calculations for systems that are routinely encountered in engineering practice, especially in the calculation of k_{eff} of 3-D system models. Updated structurally from prior versions, CSAS5 implements the modern material and cross section processing module XSProc to process material input and provide a temperature resonance-corrected cross section library based on the physical characteristics of the problem being analyzed. XSProc calls several lower level functional modules, some of which perform simple functions that were not called out as separate from CSAS5 in past versions.

XSProc was developed for the Scale 6.2 release to prepare data for continuous-energy and multigroup calculations. XSProc expands material input from Standard Composition Library definitions into atom number densities (calling the integrated MixMacros module) and, for multigroup calculations, performs cross section resonance self-shielding, energy group collapse, and spatial homogenization. XSProc implements capabilities for problem-dependent temperature interpolation, calculation of Dancoff factors (calling the integrated Dancoff module), resonance self-shielding using Bondarenko factors with full-range intermediate resonance treatment, as well as use of continuous energy resonance self-shielding in the resolved resonance region. XSProc integrates and enhances the capabilities previously implemented independently in BONAMI, CENTRM, PMC, WORKER, ICE, and XSDRNPM, along with some additional capabilities that were provided by MIPLIB and SCALELIB in prior Scale release. For this work XSProc utilizes the following modules in addition to MixMacros and Dancoff (CENTRM and PMC are called via the CentrmPmc module):

- BONAMI: The BONAMI module is used to perform Bondarenko calculations for resonance self-shielding. BONAMI obtains problem-independent cross sections and Bondarenko shielding factors from a multigroup (MG) AMPX master library, and it creates a MG AMPX working library of self-shielded, problem-dependent cross sections. Several options may be used to compute the background cross section values using the narrow resonance or intermediate resonance approximations, with and without Bondarenko iterations. A novel interpolation scheme is used that avoids many of the problems exhibited by other interpolation methods for the Bondarenko factors. BONAMI is most commonly used in automated SCALE sequences and is fully integrated within the Scale cross section processing module, XSProc. During the execution of a typical Scale computational sequence using XSProc, Dancoff factors for uniform lattices of square- or triangular-pitched units are calculated automatically for BONAMI by numerical integration over the chord length distribution. Heterogeneous effects are treated using equivalence theory based on an "escape cross section" for arrays of slabs, cylinders, or spheres.

- CENTRM: CENTRM computes continuous-energy neutron spectra for infinite media, 1-D) systems, 2-D unit cells in a lattice, by solving the Boltzmann transport equation using a combination of pointwise and multigroup nuclear data. CENTRM is primarily used to calculate problem-specific fluxes on a fine energy mesh to generate self-shielded multigroup cross sections

for subsequent radiation transport computations. Several calculation options are available, including a slowing-down computation for homogeneous infinite media, 1-D discrete ordinates in slab, spherical, or cylindrical geometries; a simplified two-region solution; and 2D method of characteristics for a unit cell within a square-pitch lattice.

- PMC: PMC generates problem-dependent multigroup cross-sections from an existing AMPX multigroup cross-section library, a point wise nuclear data library, and a pointwise neutron flux file produced by the CENTRM continuous-energy transport code. In the Scale sequences, PMC is used primarily to produce self-shielded multigroup cross-sections over a specified energy range such as the resolved resonance energy range of individual nuclides in the system of interest. The self-shielded cross-sections are obtained by integrating the point wise nuclear data using the CENTRM problem-specific, continuous-energy flux as a weight function for each spatial zone in the system.

- KENO: The KENO module is a Monte Carlo criticality program used to calculate the k_{eff} of 3-D models using continuous energy or multigroup cross-sections and is called by CSAS5 once XSProc is complete. 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. The version used in this work, KENO V.a, contains a simplified geometry package appropriate for use here. Anisotropic scattering is treated by using discrete scattering angles using P_n Legendre polynomials. KENO uses problem-specific cross-section libraries, processed for resonance self-shielding and for the thermal characteristics of the problem.

For this work, the option parm=centrm is used as input, for which the CENTRM/PMC modules are executed to process shielded multi-group cross sections using continuous energy flux spectra calculated with the recommended type of continuous energy transport solver for the designated type of cell. An infinite homogeneous medium calculation is used for those materials not called out for special processing, uses 2-D Method of Characteristics for a LATTICECELL consisting of cylindrical fuel rods in a square lattice, and uses 1-D discrete S_n transport for all other LATTICECELLs and MULTIREGION cells.

The criticality sequence of Scale 6.2.3 is validated using fresh UO_2 critical experiments and Haut Taux de Combustion (HTC) critical experiments to form an experiment benchmark suite applicable to fresh and spent fuel criticality calculations. See NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data" (Reference 16) for an overview of the HTC criticals. Additional details of the validation are found in Appendix A. The validation shows that Scale 6.2.3 is an accurate tool for calculation of k_{eff} for SFP applications. The benchmark calculations use the same computer platform and cross-section libraries that are used for the design basis calculations. The validation considers both fresh UO_2 and fuel with plutonium designed to have an actinide composition similar to burned fuel.

2.3.3 Scale 238 Group Cross-Section Library

The 238-group ENDF/B-VII library included in the Scale package is available for general purpose criticality analyses. The group structure is the same as the 238-group ENDF/B-V and ENDF/B-VI

libraries in Scale, and the same weighting spectrum as for the ENDF/B-VI. As with the 238-group ENDF/B-VI library, the ENDF/B-VII library cannot be used with the NITAWL-III module for resonance self-shielding calculations in the resolved range.

The 238-group and continuous-energy ENDF/B-VII libraries have 417 nuclides that include 19 thermal-scattering moderators. The validation of the ENDF/B-VII 238-group library with the Scale Version 6.2.3 CSAS5 module is documented in Appendix A.

3 FARLEY UNITS 1 & 2 NUCLEAR POWER PLANT

This section describes the physical characteristics of Farley Units 1 & 2 that are important to SFP criticality safety. Pertinent reactor characteristics and associated fuel design and fuel management history are discussed in Section 3.1. The physical characteristics of the SFPs are discussed in Section 3.2.

3.1 REACTOR DESCRIPTION

The Farley Units 1 & 2 Nuclear Power Plant is a Westinghouse PWR utilizing fuel with a 17 x 17 lattice. Farley Units 1 & 2 have used multiple fuel designs from Westinghouse. All fuel assemblies used at Farley Units 1 & 2 incorporate a 17 x 17 square array of 264 fuel rods with 24 guide tubes (GT) and 1 instrument tube (IT). The fuel rod cladding material is Zircaloy cladding 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.

Section 3.1 provides data on the design and operation of Farley Units 1 & 2 as well as the fuel designs and fuel management of the plant. Table 3-1 provides basic data on the type of reactor and the fuel types that comprise Farley Units 1 & 2. The neutronically important mechanical features of the three fuel designs are listed in Table 3-2.

Table 3-1 Reactor General Specifications

Reactor type	Westinghouse
Historic & current reactor power[1] (MWt)	2652-2775
Fuel lattice	17 x 17
Fuel design 1	Westinghouse Standard Fuel Assembly
Fuel design 2	Westinghouse Optimized Fuel Assembly
Fuel design 3[2]	Westinghouse Robust Fuel Assembly

Note:

1. Reactor power will be analyzed in this work up to 2831 MWt with current fuel management strategy to address future operation.
2. The RFA fuel design has not been used at Farley, and there are no current plans to transition to this fuel design. However, it is included in this analysis to support potential future use provided the RFA design is operated within the analysis area of applicability.

Table 3-2 Fuel Design Mechanical Specifications			
Assembly type	STD	RFA	OFA
Rod array size	17 x 17	17 x 17	17 x 17
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 outer dimension (OD), in	0.374	0.374	0.36
Fuel cladding inner dimension (ID), in	0.329	0.329	0.315
Fuel cladding thickness, in	0.0225	0.0225	0.0225
Pellet diameter, in	0.3225	0.3225	0.3088
Number of GT/IT	24/1	24/1	24/1
GT/IT OD, in	0.482	0.482	0.474
GT/IT ID, in	0.450	0.442	0.442
Percent theoretical density, nominal	95.0 – 96.5	95.0 – 96.5	95.0 – 96.5

Non-mechanical fuel features which are important to criticality safety and how they impact the number of distinct fuel designs are considered in this analysis. Operational characteristics of every cycle operated at Farley Units 1 & 2 were reviewed. All cycles can be categorized conservatively into one of the following Criticality Fuel Designs. Table 3-3 outlines the key non-mechanical features and fuel management history of each of the fuel designs.

Table 3-3 Non-Mechanical Specifications and Operating History							
Criticality Fuel Design	**1**	**2**	**3**	**4**	**5**	**6**	**7**
Assembly type	STD	STD	STD	OFA	OFA	OFA	OFA
Max. TD[1]	96.5	96.5	98.0	96.5	96.5	96.5	98.0
Max. operating power, MWt	2652	2652	2831	2775	2652	2775	2831
Axial blanket enrichment	No Blanket	No Blanket	Annular, Fully Enriched	No Blanket	No Blanket	No Blanket	Annular, Fully Enriched
Axial blanket length, in	N/A	N/A	6	N/A	N/A	N/A	6
Burnable absorber (BA) Type	Pyrex	WABA	IFBA	IFBA / WABA	IFBA / WABA	IFBA	IFBA
BA material	B_2O_3-SiO_2	B_4C	ZrB_2	ZrB_2 / B_4C	ZrB_2/ B_4C	ZrB_2	ZrB_2
BA max. loading	12.5 wt%	6.03 mg^{10}B/cm	1.50X	1.00X / 6.03 mg^{10}B/cm	1.00X / 6.03 mg^{10}B/cm	1.50X	1.50X
Max BA length, in	144	134	132	IFBA: 132 WABA: 132	IFBA: 132 WABA: 134	132	132
Maximum number of rods / fingers	24	20	200	156 / 8	104 / 12	156	200

Note:

1. Percentage of Theoretical Density (TD).
2. The Max. TD for Criticality Fuel Designs 3 and 7 are chosen to bound potential future operation and are greater than current experienced at Plant Farley as seen in Table 3-2.

Criticality Fuel Design 3 and 7 are selected as the design basis fuel designs for STD/RFA and OFA fuel designs respectively. See Section 4.3.2 for methodology details for determination of the design basis criticality fuel designs.

3.2 FUEL STORAGE DESCRIPTION

The physical characteristics of the Farley Units 1 & 2 SFPs are described in this section. The SFPs are made up of one fuel storage rack design (region). The Farley Units 1 & 2 SFPs each consist of two 6 x 7, nineteen 7 x 7, and seven 7 x 8 storage racks. The storage racks are of flux trap style with an uncredited Boraflex neutron absorber panel on every side (in the x and y-axis directions) of each storage cell. This results in a flux trap between any two assembly storage locations. A schematic layout of the unit 1 and unit 2 SFP is given in Figure 3-1 and Figure 3-2, respectively. See Section 5.1.1 for modeling details. The specifications for the storage racks are given in Table 3-4.

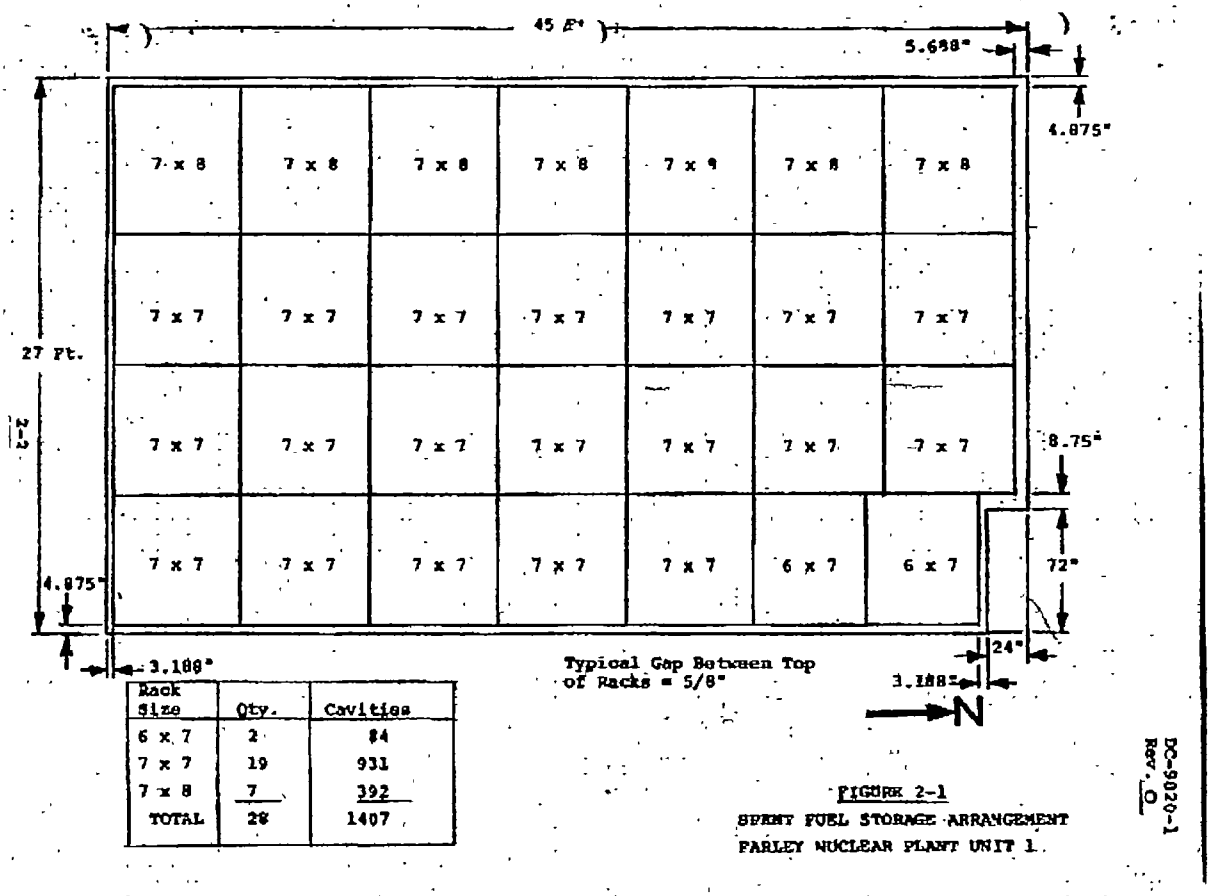

Figure 3-1 Farley SFP (Unit 1) Layout

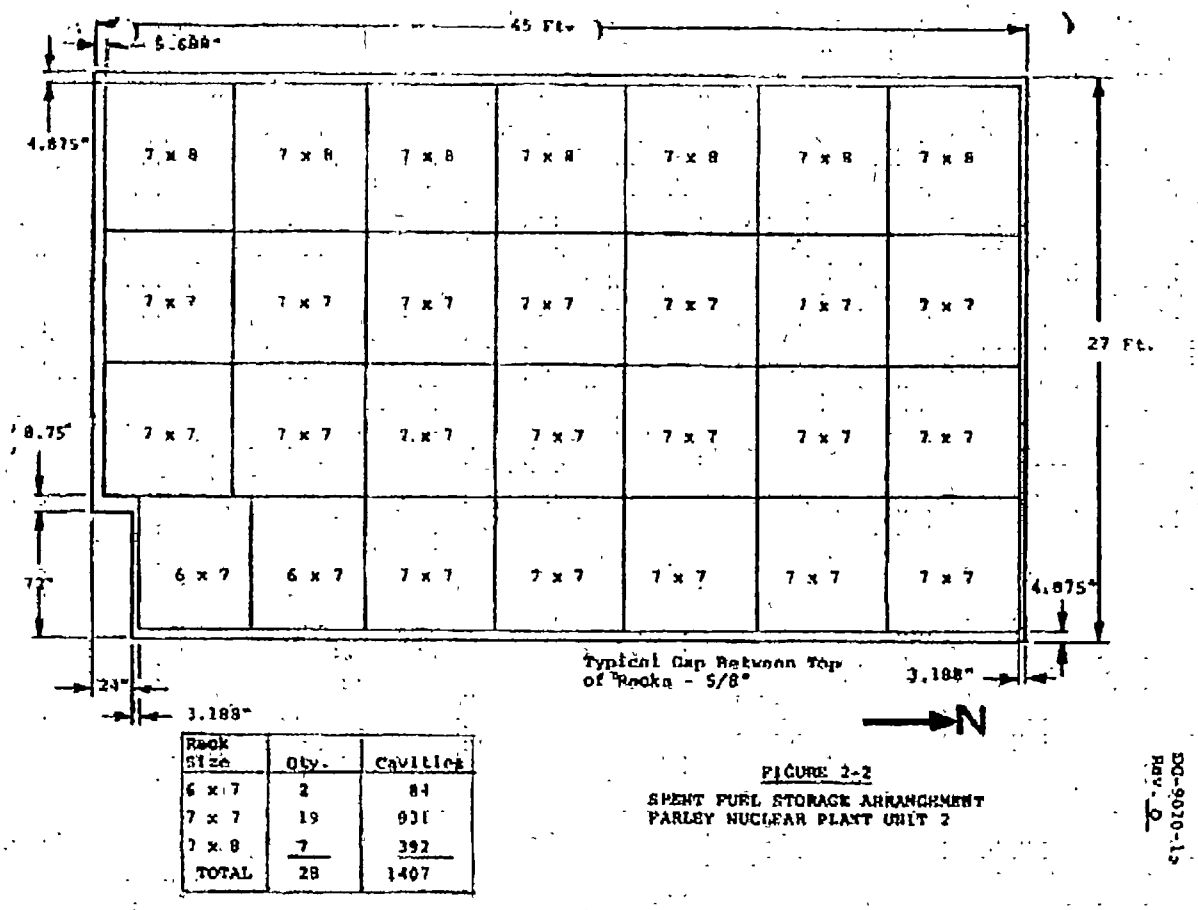

Figure 3-2 Farley SFP (Unit 2) Layout

Table 3-4 Fuel Storage Rack Specifications		
	Value	**Tolerance**
Cell pitch, in	10.75	±0.06
Cell ID, in	8.9	±0.045
Cell wall thickness, in	0.12	± 0.012
BA[1] Type	Boraflex	N/A
BA cavity width, in	8	±0.06
BA cavity thickness, in	0.07	N/A
BA wrapper thickness, in	0.024	±0.003

Note:

1. No credit is taken for the presence of any residual Boraflex. The BA cavity is assumed to be filled with water of the same composition as the water elsewhere in the storage racks.

Also present in the SFPs are fuel rod storage canisters (FRSCs) and loose pellet transport canisters (LPTCs). The fuel rod storage canister at Farley Units 1 & 2 is a rectangular lattice of storage tubes for failed fuel rods arranged in an 8 x 8 pattern. Design details of the FRSCs are given in Table 3-5. Section 5.4.3 contains additional modeling details.

The loose pellet transport canisters (LPTCs) are stainless steel (SS) canisters designed to store up to 5000 loose fuel pellets within individual loose pellet canisters stored within the LPTCs. Design details are given in Table 3-6. Section 5.4.3 contains additional modeling details.

Table 3-5 Fuel Rod Storage Canister	
Maximum rod loading of FRSC	52 rods with fresh 5 wt% ^{235}U loading
Length of FRSC, in	155.75
Lattice of storage locations	8 x 8
Fuel rod storage tubes per row	4, 6, 8, 8, 8, 8, 6, 4
Fuel rod storage tube OD, in	0.750
Fuel rod storage tube thickness, in	0.120
Fuel rod storage tube material	SS-304 (not modeled)
Fuel rod pitch, in	0.937

Table 3-6 Loose Pellet Transport Canister	
Maximum Loading of LPTC	5000 pellets at 5 wt%, no burnup
Length of LPTC, in	258
Outer shell material	SS-304
Outer shell thickness, in	0.375
Loose pellet canister outer dimensions, in	7 x 5
Loose pellet canister material	SS-304
Loose pellet canister SS thickness, in	0.078125

4 DEPLETION ANALYSIS

This section describes the methods used to determine the conservative and bounding inputs for the generation of isotopic number densities, which are then used in subsequent Monte Carlo simulations.
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]^{a,c}

4.1 DEPLETION MODELING SIMPLIFICATIONS & ASSUMPTIONS

There are several different combinations of fuel designs including differing mechanical designs, operating conditions, and BA types that need to be considered when performing the analysis. To facilitate the analysis, two bounding design basis fuel assembly types are determined, one for fuel designs with a nominal rod outer diameter of 0.374 inches (STD) and one with a nominal rod outer diameter of 0.360 inches (OFA).

[

]^{a,c}

- Depletion isotopics for use in the Criticality Analysis are generated every 2000 MWd/MTU.

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 design, fuel enrichment, burnup, and decay time after discharge.

[

]^{a,c}

Based on the Farley Units 1 & 2 fuel management, the fuel has isotopic number densities which are calculated at enrichments of 3, 4, and 5 wt% ^{235}U and decay times of 0, 5, 10, 15, and 20 years. Fresh fuel modeled in this analysis conservatively excludes ^{234}U and ^{236}U.

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. NEI-12-16, Revision 3, "Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants" (Reference 14) provides practical guidance for criticality safety analysts in line with current recommendations. 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 the fuel selected for modeling are provided in Table 4-5, which 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.

Based on guidance from Reference 6, "establishment of a bounding value for the maximum average boron per cycle based on boron let-down curves would enable more straightforward application of the depletion analyses," a constant cycle average soluble boron concentration (Equation 4-1) which assumes 19.9 at% ^{10}B in place of a soluble boron letdown curve is considered appropriately conservative. To determine the maximum cycle average soluble boron concentration, fuel management strategies for

Farley Units 1 & 2 have been reviewed. Table 4-1 provides the cycle average soluble boron concentration for cycles 1 through 29 of Unit 1 and for cycles 1 through 26 of Unit 2.

$$\left[\quad\right]^{a,c} \qquad \text{Equation 4-1}$$

[

]^{a,c}

Table 4-1 Cycle Average Soluble Boron Concentration (ppm)		
Cycle #	**Unit 1**	**Unit 2**
Cycle 1	567.3	547.5
Cycle 2	491.6	482.6
Cycle 3	507.8	653.3
Cycle 4	493.3	703.7
Cycle 5	461.3	685.8
Cycle 6	669.6	806.7
Cycle 7	783.0	769.3
Cycle 8	720.1	724.3
Cycle 9	777.1	851.7
Cycle 10	777.2	694.1
Cycle 11	789.9	856.0
Cycle 12	762.5	807.9
Cycle 13	754.4	722.3
Cycle 14	828.0	684.8
Cycle 15	771.6	783.7
Cycle 16	714.8	767.1
Cycle 17	755.9	800.7
Cycle 18	785.3	771.1
Cycle 19	797.0	803.9
Cycle 20	785.3	751.1
Cycle 21	783.9	801.1
Cycle 22	765.4	805.4
Cycle 23	783.2	777.5
Cycle 24	746.0	768.2
Cycle 25	786.0	798.8
Cycle 26	777.9	788.9
Cycle 27	771.3	N/A
Cycle 28	787.1	N/A
Cycle 29	780.6	N/A

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 discharged fuel. Therefore, utilizing a higher fuel temperature is more conservative.

The temperature input for this analysis is calculated by the FIGHTH code documented in Westinghouse WCAP-9522, "FIGHTH – A Simplified Calculation of Effective Temperatures in PWR Fuel Rods for Use in Nuclear Design" (Reference 7), which determines the fuel temperatures used as input to PARAGON for depletion calculations. FIGHTH calculates the steady state radial temperature distribution at each burnup, given the local value of the heat generation rate in the rod, the moderator temperature, and coolant flow rate. The FIGHTH model accounts for radial variations of the heat generation rate, thermal conductivity, thermal expansion in the fuel pellet, elastic deflection for the cladding, and pellet-clad gap conductance. The FIGHTH code is used in the development of cross-sections for in-core calculations as part of the standard reload methodology.

As discussed, the important input parameters used by FIGHTH for determining fuel temperature are power level, moderator temperature, and coolant flow rate. [

]^{a,c} Selection of moderator temperature is performed as discussed in Section 4.2.3.2. [

]^{a,c}

4.2.2.3 Operating History and Specific Power

The analysis assumes constant full power operation consistent with a bounding assembly average power. For fission product credit analyses, the conservative direction for specific power varies with burnup (see Reference 6). However, assuming a bounding assembly average power (therefore high specific power) ensures high fuel temperatures which is conservative throughout life. Interim Staff Guidance (ISG) DSS-ISG-2010-001 (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."

As anticipated by the ISG and consistent with sensitivity study results reported in Reference 6, the fuel temperature impact on reactivity is greater than the impact from specific power. Guidance in Reference 14 corroborates this assessment. This makes the selection of a high operating power, and therefore specific power to maximize fuel temperature, appropriate as the subordinate parameter is more conservative than nominally chosen and the 0.002 Δk bias recommended for a bounding treatment in Reference 6 is unnecessary. For additional conservatism, a 0.002 Δk uncertainty is taken on operational history.

4.2.2.4 Maximum Average Assembly Power

[

]a,c

Table 4-2 Fuel Design Operating Power	
Parameter	**Value**

a,c

[

]a,c

4.2.3 Axial Profile Selection

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

]^{a,c}

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 as one of the conservative input parameters to develop isotopics to ultimately calculate the minimum burnup requirements 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 axial center of each assembly generates more power than the ends. This leads to the burnup of each assembly varying along its length. Because the axial center of each assembly generates most of the power, the burnup in the axial center 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 axial center and end of an assembly is large enough, reactivity becomes driven by the end of the assembly rather than the axial center, as the under depletion of the ends (the end-effect) overcomes the reactivity loss due to neutron leakage.

As driven by the end-effect, the following methodology was used to ensure that the appropriate axial burnup profiles were selected for this analysis. Fuel management calculations containing readily available data from 25 cycles of operation were utilized to develop a database of axial burnup profiles specific to Farley Units 1 & 2. [

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[

]a,c.

For the reasons discussed above, it is typical for fuel modeled assuming a uniform axial burnup profile to be more reactive early in life than fuel modeled with a distributed profile. To address this, isotopics were created for fuel assuming both a uniform profile and distributed profile (for the design basis fuel). These isotopics were used during the Monte Carlo calculations to determine the minimum burnup requirements to ensure the limiting profile (uniform vs design basis distributed) has been used.

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 requirements 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 Farley Units 1 & 2 has been considered for this analysis and conservative assumptions have been used to bound the effects of BAs on fuel isotopics. The BAs that have been evaluated include both discrete and integral BAs. The BA rod parameters are shown in Table 4-3.

Table 4-3 Burnable Absorber Specifications

Parameter[1]	Pyrex	WABA	IFBA
BA material	B_2O_3-SiO_2	Al_2O_3-B_4C	ZrB_2
BA type	Discrete	Discrete	Integral
B_4C TD, %	N/A	70	N/A
Boric Oxide Content, wt%	12.5	N/A	N/A
^{10}B abundance or loading	19.9 at%	19.9 at%	2.35 (STD/RFA) / 2.25 (OFA) mg/in
BA thickness, in	0.073	0.02	0.0002[2]
BA ID, in	0.1900	0.2780	N/A
BA OD, in	0.3360	0.3180	N/A
BA clad material	Stainless Steel	Zirc-4	N/A
BA inner clad OD, in	0.1810	0.2670	N/A
BA inner clad thickness, in	0.0070	0.0210	N/A
BA outer clad OD, in	0.3810	0.3810	N/A
BA outer clad thickness, in	0.0185	0.0260	N/A
BA length, in	See Table 3-3	See Table 3-3	N/A
Max. BA exposure, MWd/MTU[3]	40000	40000	N/A

Notes:

1.) Additional BA information is contained in Table 3-3 for each Criticality Fuel Design considered. The maximum BA length and loading are modeled for each Criticality Fuel Design.

2.) Coating on the fuel pellet. IFBA depletion input captures the desired absorber per unit length with 0.2 mils coating thickness. Specific criticality analysis input is given in Table 5-1.

3.) Pyrex and WABA exposure is conservatively modeled to 40000 MWd/MTU. IFBA residual ^{10}B is removed from the fuel for spent fuel pool criticality calculations.

Criticality Fuel Designs 2, 4, and 5 contain WABA. The WABA length and number of WABA rodlets present will impact the final assembly reactivity, as they are directly related to the amount and location of absorber present within the assembly. For past operation with these designs, a minimum of a 5" cutback is observed, however less conservative 6" cutbacks are utilized in the depletion calculations. The maximum number of WABA rodlets used for past operating cycles falling within these designs is 20 yet 24 rodlets were analyzed. The additional rodlets ensures any reactivity impact lost from the additional inch of

WABA during operation is accounted for. For conservatism, discrete BAs (Pyrex and WABA) were not removed from the core until 40 GWd/MTU of operation.

4.2.5 Fuel Assembly Physical Changes with Depletion

Reference 14 discusses fuel assembly physical changes with depletion, and specifically calls out the need to address the potential reactivity impact from fuel rod changes (clad creep, fuel densification/swelling) and material dependent grid growth. Appendix B of Reference 14 is based on Westinghouse methodology and indicates that holistically the impact of fuel rod changes with depletion are conservative. Fuel assembly grid growth impact has been shown to be negligible during depletion and is conservatively addressed for the in-pool storage reactivity impact in Section. 5.2.3.1.11.

4.3 DESIGN BASIS FUEL SELECTION

4.3.1 Fuel Design and Management Modeling Considerations

To develop conservative storage requirements for the Farley Units 1 & 2 SFPs, the different unique fuel designs (criticality fuel designs) and the conditions in which those designs were operated or are planned to be operated were considered. All the criticality fuel designs considered are discussed in Section 3.1. [

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Table 4-4 Design Basis Comparison Modeled Fuel Design Parameters		
	Fuel Design	
Parameter	STD/RFA	OFA
Rod pitch, in	0.496	0.496
Active fuel length, in	144	144
Total number of fuel rods	264	264
Pellet OD, in	0.3225	0.3088
Clad OD, in	0.374	0.360
Clad ID, in	0.329	0.315
Number of GT/IT	24/1	24/1
GT/IT OD, in	0.482	0.474
GT/IT ID, in	0.442	0.442

Notes:
1. Blanket/Cutback, BA type and loading information is given in Table 3-2
2. RFA fuel model GT ID is used. Section 6.2 discusses the applicability of the RFA fuel design.

[

]^{a,c}

At Farley Units 1 & 2, the STD/RFA and OFA fuel designs have used the Pyrex, WABA and IFBA BAs. As discussed in NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for the PWR Burnup Credit" (Reference 11), the presence of discrete burnable absorbers such as Pyrex and WABA displace water and absorb thermal neutrons, thereby hardening the neutron spectrum and creating more plutonium isotopes. Therefore, Pyrex and WABA must be modeled in the depletion analysis. As discussed in NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit" (Reference 10), the presence of integral absorbers during depletion hardens the neutron spectrum, resulting in lower ^{235}U depletion and higher production of plutonium isotopes. As a result, the IFBA integral absorber must also be modeled in the depletion analysis for determination of the bounding criticality fuel design.

4.3.2 Reactivity Comparison Methodology

The final bounding assembly design is the determination of a limiting combination of fuel type and conservative depletion input parameters, denoted as a criticality fuel design. The design basis conservatively covers past, current, and expected future spent fuel operation for Farley Units 1 & 2. This

section outlines the methodology used to determine a bounding assembly design, including the selected criticality fuel type. For this analysis, a bounding fuel type is determined for the STD/RFA and OFA fuel designs. The potentially limiting axial burnup profiles identified using the methodology approach described in Section 4.2.3.1 are implemented together with the limiting depletion parameters for each Criticality Fuel Design.

Reactivity comparisons were performed across all burnup bins at 3, 4, and 5 wt% ^{235}U with all potentially limiting axial burnup profiles and the limiting moderator temperature profiles for each criticality fuel design with STD/RFA and OFA fuel assemblies. Comparisons were performed for both Array B and Array C (See Section 5.2.1 for a description of storage arrays). Based on the reactivity comparison Criticality Fuel Design 1 was chosen as the limiting STD/RFA criticality fuel design, and Criticality Fuel Design 7 was chosen as the limiting OFA criticality fuel design. Note that when necessary, reactivity comparisons focused on a range of reactivity of interest to spent fuel pool criticality to ensure that the limiting design was appropriate. The two largest differences were 0.00058 Δk at 30 GWd/MTU and 0.00031 Δk at 35 GWd/MTU for Array B comparisons of Criticality Fuel Design 7 with 3 wt% ^{235}U fuel with two different potentially limiting axial burnup profiles.

Criticality Fuel Design 1 was selected as limiting over Criticality Fuel Design 2 and 3 for STD (RFA) fuel. It is the combination of Criticality Fuel Design 1 input (including the burnable absorber usage during operation) that leads to the overall bounding nature, despite the lower fuel percent of theoretical density (96.5 vs 98.0) when compared with Criticality Fuel Design 3. See Section 6.2 for the analysis area of applicability.

4.4 FINAL DEPLETION PARAMETERS

This section outlines the parameters used in the final depletion calculations. The depletion parameters discussed in this section are:

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

The fuel isotopics used in the reactivity calculations were generated based on the data presented in Table 4-4, Table 4-5, and Table 4-6.

Table 4-5 Parameters Used in Depletion Analysis

Parameter	Nominal Values	Depletion Analysis
Maximum cycle average soluble boron concentration, ppm	461.3 – 856.0	[ ]^{a,c}
Rated thermal power[1], MWt	2652 - 2775	[ ]^{a,c}
Average assembly power, MWt	16.90 – 18.04 [2]	[ ]^{a,c}
Soluble boron ^{10}B atom percent, %	19.9	[ ]^{a,c}
Minimum core loading, kg U	72443 (STD), 66417 (OFA)	[ ]^{a,c}
System pressure, psia	2250	[ ]^{a,c}
Core outlet moderator temperature, °F	618.9 (max)	[ ]^{a,c}
Core inlet moderator temperature, °F	541.1 (max)	[ ]^{a,c}
Minimum RCS flow rate (thermal design flow), gpm/coolant pump	86000	[ ]^{a,c}
Fuel designs	STD/RFA, OFA	[ ]^{a,c}
Fuel assembly cutback/blanket region	See Table 3-3	[ ]^{a,c}
Blanket type	See Table 3-3	[ ]^{a,c}
TD	94.0 – 96.5	[ ]^{a,c}
BA	Pyrex, WABA, IFBA	[ ]^{a,c}
Max BA lengths, in	144 (STD), 134 (STD/OFA), 132 (STD/OFA)	[ ]^{a,c}

Notes:
1. The current rated thermal power is 2775 MWt, with a proposed uprate power of 2831 MWt.
2. This number is calculated by dividing the rated thermal power by the number of fuel assemblies.
3. [ ]^{a,c}
4. See Section 6.2 for overall applicability of fuel designs covered by the analysis area of applicability.

Table 4-6 Limiting Axial Burnup and Moderator Temperature Profiles a,c

5 CRITICALITY ANALYSIS

This section describes the reactivity calculations and evaluations performed in developing the burnup requirements for fuel storage in the Farley Units 1 & 2 SFPs. The section also confirms continued safe SFP operation during both normal and accident conditions.

5.1 KENO MODELING APPROACH, SIMPLIFICATIONS & 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. Additionally, KENO is used to determine the reactivity sensitivity of these storage arrays to effects such as manufacturing tolerances, fuel depletion, eccentric positioning, and the allowable temperature range of the SFPs. KENO is also used to model accident scenarios and confirm there is sufficient soluble boron to meet the requirements of Section 2.1.

The methods used to model the fuel in normal and accident scenarios are discussed in the following sections. [

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Table 5-1 IFBA Criticality Modeling Specifications			
Parameter	**1.00X**	**1.25X**	**1.50X**
[			]a,c

- [

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- Acceptable storage arrays are described in Section 5.2.1. Figure 5-1 shows a planar view (x-y) of each storage array as modeled (periodic boundary conditions applied in the x-y directions).

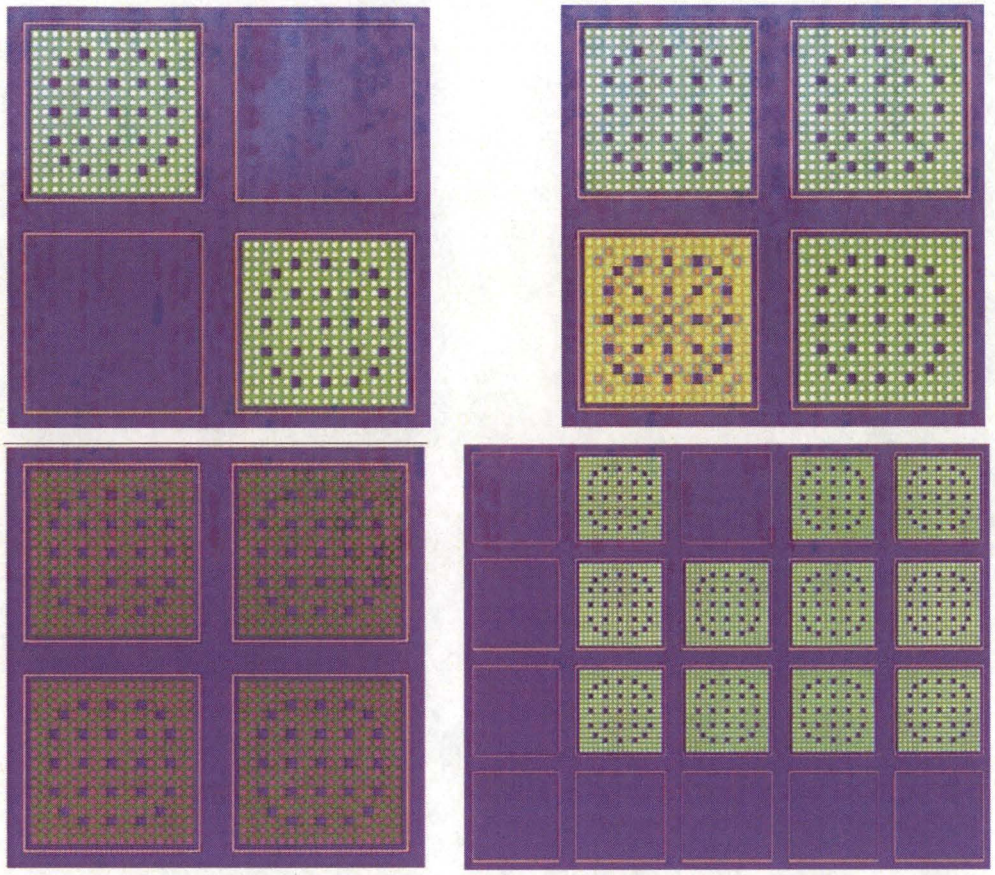

Figure 5-1 KENO Array Rack Model Planar (x-y) View: Top Left Array A, Top Right Array B, Bottom Left Array C, Bottom Right Array D (All Models: x-y Periodic Boundary Conditions)

5.1.1 Description of Fuel Assembly and Storage Racks for KENO

This section outlines the dimensions and tolerances of the design basis fuel assembly designs and the fuel storage racks. These dimensions and tolerances are the input basis for the KENO models used to determine the burnup requirements for each fuel storage array and to confirm the safe operation of the SFPs 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 assembly designs. Table 5-2 provides this data for STD/RFA and OFA fuel as modeled. Selection of these fuel designs is discussed in Section 4.3. As identified in Section 4.3.2, Criticality Fuel Design 1 was selected as limiting over Criticality Fuel Design 2 and 3 for STD (RFA) fuel. It is the combination of Criticality Fuel Design 1 input (including the burnable absorber usage during operation) that leads to the overall bounding nature, despite the lower fuel percent of theoretical density (96.5 vs 98.0) when compared with Criticality Fuel Design 3. See Section 6.2 for the analysis area of applicability.

Table 5-2 Design Basis Fuel Assembly Design Modeling Parameters		
STD (RFA) Fuel Assembly		
Parameter	**Value**	**Tolerance**
Rod array size	17 x 17	N/A
Rod pitch, in	0.496	[ ]^{a,c}
Active fuel length, in	144	N/A
Nominal fuel theoretical density, % TD	96.5	N/A
Maximum pellet enrichment, wt% ^{235}U	5	[ ]^{a,c}
Total number of fuel rods	264	N/A
Fuel cladding OD, in	0.374	[ ]^{a,c}
Fuel cladding ID, in	0.329	[ ]^{a,c}
Pellet diameter, in	0.3225	[ ]^{a,c}
Number of GT/IT	24/1	N/A
GT/IT OD, in	0.482	[ ]^{a,c}
GT/IT ID, in	0.442	[ ]^{a,c}
OFA Fuel Assembly		
Parameter	**Value**	**Tolerance**
Rod array size	17 x 17	N/A
Rod pitch, in	0.496	[ ]^{a,c}
Active fuel length, in	144	N/A
Nominal fuel theoretical density, % TD	98.0	N/A
Maximum pellet enrichment, wt% ^{235}U	5	[ ]^{a,c}
Total number of fuel rods	264	N/A
Fuel cladding OD, in	0.360	[ ]^{a,c}
Fuel cladding ID, in	0.315	[ ]^{a,c}
Pellet diameter, in	0.3088	[ ]^{a,c}
Number of GT/IT	24/1	N/A
GT/IT OD, in	0.474	[ ]^{a,c}
GT/IT ID, in	0.442	[ ]^{a,c}

Notes:
1. [ ]a,c
2. The maximum pellet enrichment tolerance is used for all enrichments evaluated as identified in Section 5.2.3.1.2.

5.1.1.2 Fuel Storage Cell Rack Dimensions and Tolerances

The storage racks used at Farley Units 1 & 2 SFPs are described in Section 3.2. The fuel storage cell characteristics, as they are modeled in the criticality analysis, are shown in Section 3.2. Dimensions including tolerances are given in Table 3-4. Tolerance models were created and the reactivity impacts were accounted for in the form of uncertainties added to the final reactivity calculation as shown in Table 5-4 through Table 5-10.

5.1.2 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 rod pitch, fuel rod dimensions such as pellet and cladding dimensions, and structural components such as grid material and volumes.

Each of the fuel types which have been or are planned to be operated at the plant need to be considered. For Farley Units 1 & 2, the determination of the design basis fuel assembly designs for the analysis has been performed as outlined in Section 4.3. The structural materials of each fuel type do not need to be considered in the determination of the bounding fuel assembly design as discussed in Reference 14 in regards to grid material where 50 ppm is added to soluble boron requirements as recommended to neglect modeling grids[1].

5.1.2.1 Composition of Structural Materials

Various zirconium-based materials and SS 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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[1] Reference 14 indicates this 50 ppm is also sufficient to offset the change in reactivity effect of tolerances under borated conditions (if modeling only unborated conditions for bias and uncertainty calculations).

5.1.2.3 Grids and Sleeves

A generic study determined the impact of grids and sleeves being present in an assembly both during core operation and storage in the SFP shows that the impact of grids and sleeves is negligible. The study was based on the [

]^{a,c} The study incorporated a variety of depletion parameters and several different [ ]^{a,c} Additionally, Reference 14 indicates for depletion that 50 ppm additional soluble boron in determination of normal and accident condition soluble boron requirements is acceptable if grids are not explicitly modeled in the SFP criticality analysis. An additional 50 ppm was used.

5.2 KENO MODELING ANALYSIS

KENO models generated for STD/RFA and OFA bounding criticality fuel designs were evaluated for different storage arrays. The enrichment, assembly average burnup, and decay time input were varied to determine appropriate storage limits based on resulting reactivity. Reactivity margin is added to the KENO reactivity calculations for the generation of burnup requirements as discussed in Section 5.2.2 to account for manufacturing deviations.

5.2.1 Array Descriptions and Fuel Categories

Assembly storage is controlled through the storage arrays defined in this section. For storage arrays A, B and C, separate requirements are determined by fuel type (STD/RFA as one type, OFA as the second type). Each storage array contains assemblies which are defined by a fuel category as given in Table 5-3. A fuel category is a ranking of assemblies by the maximum allowable reactivity of the individual assembly in a storage cell within each storage location. A lower fuel category is more reactive than a higher fuel category. Unique storage locations were determined for which to assign fuel categories using the desired storage patterns for use in the Farley Units 1 and 2 SFPs. Figure 5-2 shows the allowable storage arrays, including the fuel categories discussed in this section.

Reactivity as discussed in this paragraph pertains to the maximum allowable reactivity in a storage cell. Fuel Category 1 locations can contain the highest reactivity fuel assemblies, up to 5 wt% ^{235}U assemblies with no burnup, IFBA, or decay time credit required. Fuel Category 2 locations can contain fresh fuel assemblies with up to 5 wt% ^{235}U but are subject to IFBA requirements (thereby reducing the reactivity compared to Fuel Category 1 fuel assemblies). Additionally, Fuel Category 2 assemblies must have accumulated at least 10 GWd/MTU of burnup once they have been exposed to ensure peak reactivity is considered for IFBA burnout, with the exception of a Fuel Category 3 or 4 assembly (discussed in this section). Fuel Category 3 locations are storage cells defined within Storage Array C, an "all-cell" storage cell. Fuel Category 3 locations can contain fuel assemblies of up to 5 wt% ^{235}U and have minimum burnup and/or decay time requirements which determine acceptability for storage in Array C.

To determine that Fuel Category 3 assemblies are less reactive than Fuel Category 2 assemblies (which require IFBA credit), a comparison of the burnup requirements of Fuel Category 4 assemblies that are stored with the Fuel Category 2 assemblies (in Array B) is necessary. If the three Fuel Category 4 assemblies are individually less reactive than Fuel Category 3 assemblies, then Fuel Category 2 assemblies will be allowed a higher reactivity than Fuel Category 3 assemblies because the overall

storage array reactivities are the same (iso-reactive). Additionally, the 10 GWd/MTU required for (5 wt% ^{235}U fuel assemblies) Fuel Category 2 which have burnup is significantly less than the burnup requirements for 5 wt% ^{235}U Fuel Category 3 assemblies.

As can be seen by comparing the storage requirements of Fuel Category 4 assemblies and the storage requirements of Fuel Category 3 assemblies, the Fuel Category 3 assemblies require less burnup for the same enrichment, indicating the individual assemblies are allowed to be more reactive.

An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity fuel category (e.g., Fuel Category 3 assemblies can be stored in locations for Fuel Categories 1, 2, or 3, but cannot be stored in Fuel Category 4 locations). If lower reactivity fuel category requirements are met for an assembly, they need not meet the requirements of the fuel category cell for which they are stored). This is a unique occurrence because Fuel Category 2 requirements for exposed fuel were generated for the bounding case (10 GWd/MTU for 5 wt% ^{235}U assemblies) and applied for all burned fuel stored in Fuel Category 2 locations.

In addition to these defined fuel categories, Array D contains 11 fuel storage locations generically evaluated for storage within the Farley SFPs with STD fuel. This allows assemblies which meet the damaged fuel array storage requirements to be stored anywhere in the SFPs for which an array of this size is met. Array D assemblies were given the Fuel Category D label, indicating these storage cells are for damaged fuel. An example of Array D, which is currently employed at the Farley Unit 1 SFP, is shown in Figure 5-3 including the assembly ID used for storage. All 11 damaged assemblies meet the requirements of Array D.

Table 5-3 Fuel Categories Ranked by Reactivity	
Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	Low Reactivity

Notes:

1. Assembly storage is controlled through the storage arrays defined in Figure 5-2.
2. Fuel Categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
3. Each storage cell in an array can only be populated with assemblies of the fuel category defined in the array definition or a lower reactivity fuel category.
4. Fuel Category 1 contains fuel with an initial maximum enrichment up to 5 wt% ^{235}U. Neither burnup nor IFBA is required.
5. Fuel Category 2 contains fuel with an initial maximum enrichment up to 5 wt% ^{235}U. Storage of fresh fuel is determined from the minimum IFBA equation and coefficients provided in Table 6-1 for STD/RFA fuel and Table 6-3 for OFA fuel. Fuel Category 2 fuel which has been operated in the reactor requires at least 10.0 GWd/MTU of burnup with the exception of a Fuel Category 3 or 4 assembly.
6. Fuel Categories 3 and 4 are determined from the minimum burnup equation and coefficients provided in Table 6-7 and Table 6-11 for STD/RFA fuel, and in Table 6-9 and Table 6-13 for OFA fuel, respectively. Example burnup requirements at several initial enrichments and decay times are provided for Fuel Categories 3 and 4 in Table 6-8 and Table 6-12 for STD/RFA fuel, and Table 6-10 and Table 6-14 for OFA fuel, respectively.
7. Example IFBA requirements at several initial enrichments for IFBA thicknesses of 1.0X, 1.25X, and 1.50X are provided in Table 6-2 for STD/RFA fuel and in Table 6-4 for OFA fuel.

<table>
<tr><td colspan="5" rowspan="2">Array A

Two Category 1 assemblies with two empty storage locations. The Category 1 fuel assemblies must only be face adjacent to an empty storage location.</td><td>1</td><td>X</td></tr>
<tr><td>X</td><td>1</td></tr>
<tr><td colspan="5" rowspan="2">Array B

One Category 2 assembly with three Category 4 assemblies.</td><td>4</td><td>4</td></tr>
<tr><td>4</td><td>2</td></tr>
<tr><td colspan="5" rowspan="2">Array C

Four Category 3 assemblies.</td><td>3</td><td>3</td></tr>
<tr><td>3</td><td>3</td></tr>
<tr><td colspan="5" rowspan="5">Array D

Eleven Category D assemblies arranged in an array of four assemblies by three assemblies. One storage cell along the four storage cell wide side of the outside of the Array must remain empty. The storage array must have at least one row of empty cells between it and any other array (Array A, B, C, or D). A row of empty cells are not needed on any section of the configuration face adjacent to the SFP wall.</td><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td>X</td><td>D</td><td>X</td><td>D</td><td>D</td><td>X</td></tr>
<tr><td>X</td><td>D</td><td>D</td><td>D</td><td>D</td><td>X</td></tr>
<tr><td>X</td><td>D</td><td>D</td><td>D</td><td>D</td><td>X</td></tr>
<tr><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td colspan="7">Notes:
1. Any storage array location designated for a fuel assembly may be replaced with non-fissile material or an empty (water-filled) cell.
2. Empty locations designated with an X must remain completely empty.
3. Storage requirements are determined for different fuel types (RFA/STD and OFA) for fuel category 1 through 3. Only RFA/STD fuel is evaluated for storage as Category D.</td></tr>
</table>

Figure 5-2 Allowable Storage Arrays

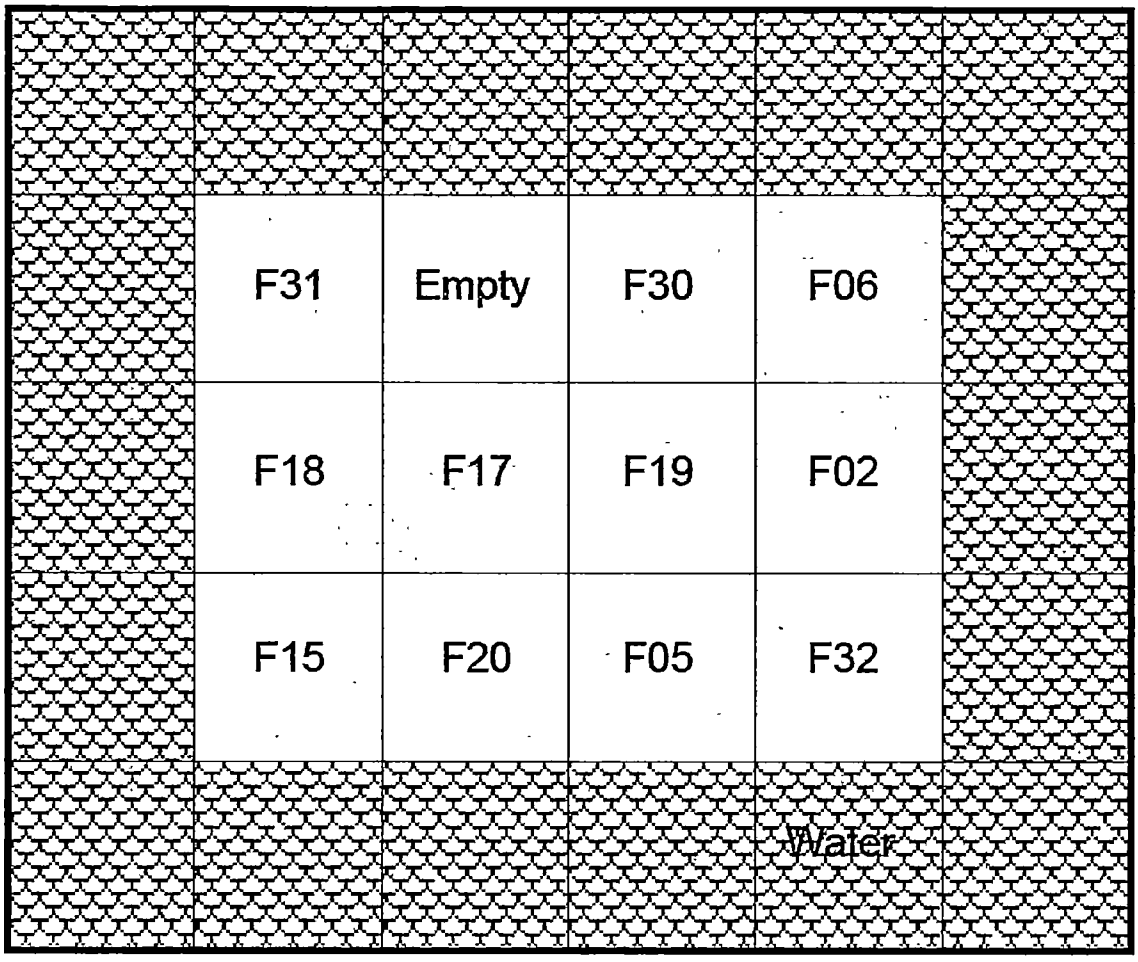

Figure 5-3 Currently Used Damaged Fuel Assembly Configuration (Farley Unit 1)

5.2.2 Target k_{eff} Calculation Description

As discussed in Section 2.1, this analysis provides burnup requirements such that the Farley Units 1 & 2 SFPs remain subcritical in unborated conditions. To ensure that the burnup requirements generated are appropriate, a target k_{eff} value is created for each array at different enrichments (maximum fresh, 3, 4 and 5 wt%, ^{235}U). The target k_{eff} value accounts for the reactivity effect of applicable biases and uncertainties and includes administrative margin to ensure safety as shown in Equation 5-1.

$$\text{Target } k_{eff} = Acceptance\ Criterion - Admin\ Margin - \sum(Biases\ \&\ Uncertainties) \quad \text{Equation 5-1}$$

where,

Acceptance Criterion = the maximum allowable k_{eff} for a storage array (see Section 2.1)

Admin Margin = the administrative margin (0.005 Δk) taken to provide additional certainty of safe operation

Σ(Biases & Uncertainties) = the amount of reactivity that accounts for biases and uncertainties in the reactivity calculation for each storage array

The sum of biases are simply additive while the sum of uncertainties are statistically added as the root sum square of the individual reactivity uncertainties.

5.2.3 Bias & Uncertainty 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 SFP temperature and code validation biases. Uncertainties account for allowable variations within the real model whether they are physical (manufacturing tolerances), analytical (depletion uncertainty and validation bias uncertainty), or measurement related (burnup measurement uncertainty). Biases have a greater impact due to their direct addition to the total sum of bias and uncertainty. Uncertainties are statistically added as the root sum square of the individual reactivity uncertainties.

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5.2.3.1 Bias & Uncertainty Descriptions including Manufacturing Tolerances

Reactivity biases and uncertainties as a result of manufacturing tolerances and other SFP characteristics are discussed in this section and the following subsections. KENO is used to quantify reactivity effects.

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5.2.3.1.1 Cladding Tolerance Reactivity Uncertainty

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5.2.3.1.2 Initial Fuel Enrichment Reactivity Uncertainty

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5.2.3.1.3 Guide Tube and Instrument Tube Reactivity Uncertainty

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5.2.3.1.4 Burnup Measurement Uncertainty

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5.2.3.1.5 Depletion Uncertainty

The depletion uncertainty takes into account the potential reactivity misprediction of the depletion code.
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5.2.3.1.6 Operational Uncertainty

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5.2.3.1.7 Flux Trap Gap Reactivity Uncertainty

The flux trap gap tolerance worth is not explicitly calculated; however the rack cell pitch tolerance cases explicitly include the effect of cell pitch and flux trap gap tolerance since the rack pitch would change the flux gap width and vice versa. The rack pitch tolerance was chosen as it is the larger change.

5.2.3.1.8 Borated Sheath Width Reactivity Uncertainty

Reference 14 indicates that for flux-trap rack designs, the uncertainty due to the manufacturing tolerance on the sheathing width is small but cannot generically be declared negligible. Despite not crediting the borated insert, sheath width tolerance reactivity uncertainty is determined.

5.2.3.1.9 Borated Insert Cavity Width Uncertainty

In addition to the borated sheath width reactivity uncertainty, the borated insert cavity width uncertainty is determined.

5.2.3.1.10 Other Uncertainties

An uncertainty in the predictive capability of Scale 6.2.3 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.

5.2.3.1.11 Assembly Envelope Expansion Bias

The assembly envelope expansion bias is comprised of [

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5.2.3.1.12 Fission Product and Minor Actinide Worth Bias

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. The validation of fission products, however, is more difficult because few critical experiments are available. Due to the limited availability of fission product benchmark data, a factor of uncertainty was considered in the criticality safety analysis.

NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality (k_{eff}) Predictions" (Reference 13) presents findings that show for minor actinide and fission product nuclides for which adequate critical experiment data are not available, calculations of k_{eff} uncertainty due to nuclear data uncertainties can be used to establish a bounding bias value which was approximately 1.5 percent of the worth of the minor actinides and fission products.
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5.2.3.1.13 Eccentric Fuel Assembly Positioning Bias

The fuel assemblies are assumed to be nominally located in the center of the storage rack cell; however, it is recognized that an assembly could in fact be located eccentrically within its storage cell. Reference 14 indicates that assembly eccentric positioning should be considered in racks without absorber panels. Racks in this analysis contain two absorber panels between each storage location, however, they are not credited in this analysis, so an eccentric positioning bias is determined.

To quantify the reactivity effects of eccentrically located fuel within a fuel storage cell, [

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Figure 5-4 [ ]^{a,c}

5.2.3.1.14 SFP Temperature Bias

The Farley Units 1 & 2 SFPs do not have a nominal temperature; instead it operates within an allowable range. [

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5.2.3.1.15 Borated and Unborated Biases and Uncertainties

Technical Specifications require each SFP to have k_{eff} to be < 0.95 under borated conditions accounting for all applicable biases and uncertainties. [

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5.2.3.2 Storage Array Biases & Uncertainties Results

Tables 5-4 through 5-10 give the calculated biases and uncertainties for Array A, B, and C, and D for STD and OFA fuel as well as the total sum of biases and uncertainties and administrative margin for determination of the listed Target k_{eff} values. Note that for Array B, the initial enrichments shown at the top of Table 5-6 for STD fuel and Table 5-7 for OFA fuel correspond to Fuel Category 4. [

]^{a,c} Fuel Category 2 assemblies are fresh fuel assemblies up to 5 wt%, with fresh IFBA requirements discussed in Section 6.1.

Table 5-4 Biases & Uncertainties for Array A with STD Fuel

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Table 5-5 Biases & Uncertainties for Array A with OFA Fuel

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Table 5-6 Biases & Uncertainties for Array B with STD Fuel a,c

Table 5-7 Biases & Uncertainties for Array B for OFA Fuel

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Table 5-8 Biases & Uncertainties for Array C with STD Fuel a,c

Table 5-9 Biases & Uncertainties for Array C with OFA Fuel

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Table 5-10 Biases & Uncertainties for Array D with STD Fuel a,c

5.3 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. In this analysis, only intra-region interfaces are evaluated since all racks are of the same design and no pool region interfaces are present. In addition to the intra-region interfaces, Array D assemblies are required to have a row of empty storage cells (or the pool wall) face adjacent to all sides of the storage array.

The only interface conditions that need to be addressed in this analysis are those between different fuel storage arrays. [

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

Additionally, Array D contains 11 fuel storage locations generically evaluated for storage within the Farley SFPs. One storage cell along the four storage cell wide side of the outside of the Array must remain empty i.e. water-filled. The storage array must have at least one row of empty cells between it and any other array (Array A, B, C, or D). A row of empty cells is not needed on any section of the configuration face adjacent to the SFP wall.

5.4 NORMAL CONDITIONS

This section discusses normal conditions within the SFPs which are in addition to the steady-state storage of fresh and spent fuel assemblies. During normal operation, the SFPs have a soluble boron concentration of greater than 2000 ppm and a moderator temperature ≤ 185°F. Beyond the storage of fuel assemblies, there are five major types of normal conditions covered in this analysis. These five conditions are explained in subsections 5.4.1 through 5.4.5.

5.4.1 Type 1 Normal Conditions

Type 1 conditions involve the placement of components in the guide tubes and/or instrument tube of 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 include control rods, neutron sources, guide tube probing, fuel assembly guide tube length measurement, and ultrasonic test equipment being placed in a testing location on top of a spent fuel storage rack.

The Type 1 normal conditions typically include the insertion of components into fuel assemblies for storage in the SFPs (e.g., depleted Pyrex). The SFPs as single systems are over moderated. A single fuel assembly however, is significantly undermoderated, and reducing the interstitial hydrogen to uranium ratio lowers the system k_{eff} as seen by the fact that all rod pitch uncertainty cases show that a reduction in rod pitch reduces reactivity. Additionally, calculations have been performed which show that [

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

5.4.2 Type 2 Normal Conditions

Type 2 conditions involve evolutions or transitional fuel assembly actions where the fuel assembly is removed from its normal storage rack location for a specific procedure and reinserted after the completion of the procedure. Examples of Type 2 conditions include fuel assembly visual inspection, reconstitution, cleaning and sipping. During the Type 2 assembly evolutions 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. Descriptions of each of these items are provided, along with the evaluation of the impact on this criticality safety analysis.

One cell pitch of separation is defined as one rack pitch distance away from all sides of the assembly (including both face adjacent and corner adjacent cells). Outside of a storage cell both pertain to fuel which has been removed to a location outside of the storage rack as well as to the areas of an assembly exposed above the rack due to partial insertion.

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 involves rod movement from and/or to an assembly. In most cases, damaged rods will be replaced with SS rods, but natural uranium rods may also be used. If a rod is replaced with either SS or a rod made of natural uranium the reactivity of the available fissile material of the single assembly will be decreased while capable moderation remains the same resulting in a reduction in reactivity. If the fuel rod in question is either replaced by a fuel rod from another assembly or the fuel rod is to be removed without replacement, adjustments must be made to the burnup storage requirement of the assembly being reconstituted.

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Fuel assembly sipping is defined as placing one fuel assembly in the sipping equipment. The fuel assembly is separated from all other stored fuel by at least one assembly pitch via the equipment design. While the sipping equipment can be placed within one assembly pitch adjacent to a storage rack loaded with fuel, the fuel assembly loaded into the sipping equipment must be 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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5.4.3 Type 3 Normal Conditions

Type 3 conditions involve insertion of components that are not intact fuel assemblies, into the fuel storage rack cells. For Farley Units 1 & 2 SFPs, these include a loose pellet canister as well as a failed fuel rod storage canister. Additionally, 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 5.2.1.

The fuel rod storage canister at Farley Units 1 & 2 is a rectangular lattice of storage tubes for failed fuel rods arranged in an 8 x 8 pattern. Not all rows contain 8 tubes as can be seen from the modeled schematic in Figure 5-6. Table 3-5 contains pertinent design information.

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Figure 5-6 Schematic View of Modeled Failed Fuel Rod Storage Canister

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The loose pellet transport canisters (LPTCs) are SS canisters designed to store up to 5000 loose fuel pellets. Design details are given in Table 3-6. Modeling of the LPTCs [

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Figure 5-7 [ ]a,c

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5.4.4 Type 4 Normal Conditions

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

5.4.5 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 or a fuel rod canister, etc.).

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.

Performance of Foreign Object Search and Retrieval (FOSAR) from fuel assemblies and/or storage cells must meet the following guidelines.

1. If a FOSAR is done on a storage cell, any fuel assembly residing in the storage cell must be removed before the action takes place.

2. For FOSAR done on a fuel assembly, if the operations do not occur in the active fuel region and do not require tooling to reside in the active fuel region, the FOSAR does not impact criticality and the assembly can remain in its storage cell.

3. If the FOSAR requires tooling to be present in the active fuel region, then the fuel assembly must be separated from other fuel assemblies by at least one assembly pitch.

The Farley SFPs have pump/filtration systems which sit on top of the fuel racks. These systems displace water above the assembly, which is conservative in unborated conditions. Borated models for accident analysis use unborated water above and below the active fuel before a reflective boundary is applied. In addition, the separation from the fuel rods due to the top nozzle and the fuel rod end plug is sufficient to prevent significant neutron interaction. Therefore, there is no restriction on the location of the filtration systems.

5.5 SOLUBLE BORON CREDIT

Section 2.1 contains k_{eff} requirements under both the assumption that the pool is flooded with pure water and that the pool contains soluble boron. This section outlines the calculations that were performed to demonstrate the soluble boron concentration necessary to meet the soluble boron requirements in Section 2.1. In reporting the soluble boron requirements, the atomic percent (at%) of ^{10}B in boron is conservatively assumed to be 19.4 to bound the potential variation in the isotopic concentration of boron within the SFPs.

5.5.1 Soluble Boron Requirements for Normal Conditions

Soluble boron credit for Normal Operating Conditions is evaluated for Farley Units 1 & 2. Additional pertinent details for modeling each storage array conservatively for the Normal Operating Conditions soluble boron determination are as follows. While additional models were evaluated with all fresh fuel, cases with the highest burnup are conservative as expected due to the reduction in boron worth with burnup due to spectral hardening. As a result, the following models are conservative for each storage array:

- Array A: Fuel assemblies were modeled as fresh 5 wt% ^{235}U fuel.
- Array B: Fuel is modeled as three 5 wt% initial enrichment fuel at 48 GWd/MTU and one fresh 5 wt% ^{235}U fuel assembly.
- Array C: Fuel is modeled as 5 wt% initial enrichment at 34 GWd/MTU.
- Array D: Fuel is modeled as 3 wt% ^{235}U initial enrichment at 4 GWd/MTU

To determine the maximum soluble boron concentration for normal conditions to meet a 95/95 k_{eff} of < 0.95 including biases and uncertainties, where 95/95 k_{eff} is defined as

$$95/95\ k_{eff} = \text{KENO}\ k_{eff} + 2\sigma_{k_{eff}} + \text{B\&U} + \text{Adm. margin}, \qquad \text{Equation 5-8}$$

where:

KENO k_{eff} = The simulated accident condition k_{eff}
$\sigma_{k_{eff}}$ = The simulated accident condition k_{eff} Monte Carlo simulation standard deviation
B&U = The total bias and uncertainty term[1]
Adm. Margin = Administrative margin.

The minimum soluble boron concentration to maintain k_{eff} < 0.95 for the limiting normal condition including biases, uncertainties, and administrative margin is 320 ppm, conservatively rounded up from value determined from linear interpolation (plus rounding) of Array B is 270 ppm. Results are given in Table 5-11 for STD/RFA fuel and in Table 5-12 for OFA fuel for all storage array.

[1] Biases and uncertainties are taken from the nominal storage condition for all storage arrays.

Table 5-11 [ ]a,c

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Table 5-12 [ ]a,c

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5.5.2 Soluble Boron Requirements for Accident Conditions

In addition to maintaining k_{eff} not to exceed 0.95 during normal operations, soluble boron is used to offset the potential reactivity insertion events in the SFPs. The following accidents are considered in this analysis:

- Assembly misload
- SFP temperature greater than normal operating range (> 185°F)
- Dropped & misplaced fresh fuel assembly
- Seismic event

5.5.2.1 Assembly Misload

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

5.5.2.1.1 Single Assembly Misload

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5.5.2.1.2 Multiple Assembly Misload

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

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Table 5-13 [ ]^{a,c}

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5.5.2.2 Spent Fuel Temperature Outside the Normal Operating Range

The J.M. Farley Units 1 and 2 SFPs are to be operated at less than 185°F. However, under accident conditions this temperature could be higher. [

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5.5.2.3 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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5.5.2.4 Seismic Event

In the event of an earthquake or similar seismic event, the SFPs 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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5.5.3 Soluble Boron Requirements

Table 5-14 data indicates that [

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Table 5-14 [ ]a,c

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5.6 RODDED OPERATION

While standard operation is performed unrodded, it is allowable to operate at hot full power with rods inserted to the power dependent insertion limits. Operating with control rods inserted into the core impacts the assemblies in the rodded locations. The insertion of a control rod into an assembly during operation has several effects.

The reactivity of an assembly experiencing rodded operation can increase relative to an assembly which does not experience rodded operation. The loss of moderator as water displaced in the GTs when a control rod is inserted into the assembly will harden the neutron spectrum increasing plutonium production. The control rod will also preferentially absorb thermal neutrons, further hardening the neutron spectrum. In addition to the spectral hardening, the control rod will lower the power in the area of the assembly where it is inserted. This will lower the burnup accumulated in the top of the assembly, increasing the end effect. These effects can all increase the reactivity of an assembly, making it possible for an assembly operated with rods inserted to be more reactive than an assembly of the same assembly average burnup which experienced unrodded operation.

While these items can increase reactivity, there are competing effects which reduce assembly reactivity due to rod insertion. When a control rod is inserted into an assembly, the power in that assembly will be reduced. This will reduce both the fuel and moderator temperatures. The reduction in fuel temperature will decrease Doppler broadening leading to less neutron capture by ^{238}U, thus lowering plutonium production. The reduction in moderator temperature will increase moderator density, increasing neutron moderation and therefore softening the neutron spectrum.

In addition to impacting the neutron spectrum, rodded operation can also affect the axial burnup profile of assemblies. Operation with a control rod inserted in an assembly will shift power down, under-depleting the top of the assembly while the control rod is present. Once the control rod has been withdrawn from the assembly, power preferentially moves to the under-depleted top of the assembly, and over time the axial burnup profile developed will return to a profile typical of unrodded operation. Therefore, time-in-life before final discharge of an assembly is an important factor in the impact of rodded operation on assembly reactivity.

NUREG/CR-6759, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit" (Reference 15) defines a significant amount of control rod insertion as more than 20 cm into the core.

Farley Units 1 and 2 have not operated at full power with control rods inserted a significant length into the core. Therefore, there is no significant burnup accrued during depletion with rods inserted in the active fuel height, and no need to account for these effects in burnup limits contained within this analysis. Any assemblies incurring significant rodded operation going forward must not credit the rodded burnup.

While typical operation for Farley Units 1 & 2 is performed unrodded, there is potential to operate at reduced power levels with rods inserted. Short term reduced power operation may be the result of plant equipment issues or economic considerations and has occurred at Farley Units 1 & 2. Any impact from short term operation at reduced power levels with rods inserted will be negligible.

6 ANALYSIS RESULTS & CONCLUSION

This section documents the final storage results of the Farley Units 1 & 2 Spent Fuel Pool 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 Area of Applicability of this analysis. The Area of Applicability (AoA) of the criticality code validation suite is discussed in Appendix A.

6.1 BURNUP AND IFBA REQUIREMENTS FOR STORAGE ARRAYS

Assembly storage is controlled through the storage arrays defined in Section 5.2.1. An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity fuel category (see Table 5-3). Fuel Category 1 does not require burnup or fresh IFBA for storage. Fuel Category 2 assembly storage requirements require that they either must have not been operated in the reactor and the IFBA loading must exceed the "minimum IFBA" (# rods per assembly) given by the IFBA requirements coefficients or have at least 10.0 GWd/MTU of exposure covering the peak reactivity of IFBA bearing assemblies with 5 wt% ^{235}U enrichment. (No IFBA requirements are needed beyond 10.0 GWd/MTU). Fresh IFBA requirements coefficients as well as sample IFBA requirements are given in Table 6-1 and Table 6-2 for STD/RFA fuel and in Table 6-3 and Table 6-4 for OFA fuel. Fuel categories D, 3, and 4 are defined by assembly average burnup, initial enrichment[1], and decay time with burnup requirement coefficients and sample evaluated burnup limits given in Table 6-5 through Table 6-14.

This analysis has provided burnup requirements at discrete decay times, measured in years. However, it is acceptable to interpolate between these decay times to determine burnup requirements 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. Linear interpolation based on actual decay time should be performed between calculated values of minimum burnup associated with tabulated decay times greater and less than the actual decay time. No extrapolation beyond 20 years is permitted. This is acceptable because isotopic decay is an exponential function which means assembly reactivity will decay faster than the calculations using linear interpolation would predict.

[1] Initial enrichment is the maximum nominal enrichment of the fuel, 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 cutbacks, the maximum initial enrichment value is to be used.

Table 6-1 Fuel Category 2: STD/RFA IFBA Fitting Coefficients			
	Fitting Coefficients		
IFBA Thickness	A1	A2	A3
1.00X	5.2750	8.3325	-79.9546
1.25X	3.7476	10.8046	-72.0974
1.50X	1.8593	19.8050	-81.5075

Notes:

1. For a fuel assembly to meet the requirements the assembly must either:
   a. Not have been operated in the reactor and the IFBA loading must exceed the "minimum IFBA" (# rods per assembly) given by the curve fit for the assembly "initial enrichment," or,
   b. Have at least 10.0 GWd/MTU of exposure.
2. The specific minimum IFBA required for each fuel assembly is calculated from the following equation:

$$\#\text{ of IFBA Rods} = A1 * En^2 + A2 * En + A3$$

3. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment greater than 3.2 wt% ^{235}U and less than or equal to 5 wt% ^{235}U may be used. The number of IFBA rods required must be rounded up to the next whole number. Below 3.2 wt% ^{235}U, IFBA is not required.

Table 6-2 Fuel Category 2: Example STD/RFA IFBA Requirements (# of IFBA Rods)					
IFBA Thickness	**Average Initial Enrichment, wt% ^{235}U**				
	3.2	**3.8**	**4.2**	**4.6**	**5**
1.00X	1	28	49	70	94
1.25X	1	24	40	57	76
1.50X	1	21	35	49	64

Note:
1. The values provided in this table are provided as an example. The requirements must be calculated using the coefficients in Table 6-1.

Table 6-3 Fuel Category 2: OFA IFBA Fitting Coefficients			
	Fitting Coefficients		
IFBA Thickness	A1	A2	A3
1.00X	6.2658	0.8890	-65.4949
1.25X	3.9144	9.3963	-68.9414
1.50X	1.5898	21.8436	-84.9630

Notes:

1. For a fuel assembly to meet the requirements the assembly must either:
   a. Not have been operated in the reactor and the IFBA loading must exceed the "minimum IFBA" (# rods per assembly) given by the curve fit for the assembly "initial enrichment," or,
   b. Have at least 10.0 GWd/MTU of exposure.
2. The specific minimum IFBA required for each fuel assembly is calculated from the following equation:

$$\#\text{ of IFBA Rods} = A1 * En^2 + A2 * En + A3$$

3. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment greater than 3.2 wt% ^{235}U and less than or equal to 5 wt% ^{235}U may be used. The number of IFBA rods required must be rounded up to the next whole number. Below 3.2 wt% ^{235}U, IFBA is not required.

Table 6-4 Fuel Category 2: Example OFA IFBA Requirements (# of IFBA Rods)					
IFBA Thickness	**Average Initial Enrichment, wt% ^{235}U**				
	3.2	**3.8**	**4.2**	**4.6**	**5**
1.00X	2	29	49	72	96
1.25X	2	24	40	58	76
1.50X	2	21	35	50	64

Note:
1. The values provided in this table are provided as an example. The requirements must be calculated using the coefficients in Table 6-3.

Table 6-5 Fuel Category D: STD/RFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0	0	9.6344	-24.5678
5	0	0	9.4528	-24.1047
10	0	0	9.3343	-23.8025
15	0	0	9.2508	-23.5896
20	0	0	9.1965	-23.4510

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 the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \qquad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 2.55 wt% ^{235}U and 3 wt% ^{235}U may be used. Below 2.55 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-6 Fuel Category D: Example STD/RFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Average Initial Enrichment, wt% ^{235}U	
	2.55	3
0	0	4.336
5	0	4.254
10	0	4.200
15	0	4.163
20	0	4.138

Note:

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

Table 6-7 Fuel Category 3: STD/RFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.2251	-2.5199	21.4065	-36.6115
5	0.3002	-3.4376	24.0978	-38.9002
10	0.1856	-2.3309	20.2704	-34.6503
15	0.0892	-1.3905	17.0683	-31.1550
20	0.0388	-0.9253	15.5082	-29.4500

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 the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \qquad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 2.15 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 2.15 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-8 Fuel Category 3: Example STD/RFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	2.15	**3**	**4**	**5**
0	0	11.007	23.103	35.561
5	0	10.560	21.702	33.174
10	0	10.194	21.015	31.629
15	0	9.944	20.579	30.574
20	0	9.795	20.261	29.809

Note:

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

Table 6-9 Fuel Category 3: OFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.1692	-1.8852	18.5219	-32.7830
5	0.0191	-0.4154	13.4482	-27.1777
10	-0.0705	0.4300	10.5987	-24.0722
15	-0.1420	1.1146	8.2825	-21.5440
20	-0.1959	1.6375	6.5093	-19.6130

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 the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \qquad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 2.15 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 2.15 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-10 Fuel Category 3: Example OFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	2.15	3	4	5
0	0	10.384	21.970	33.847
5	0	9.944	21.191	32.066
10	0	9.690	20.691	30.859
15	0	9.501	20.332	29.984
20	0	9.363	20.087	29.384

Note:

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

Table 6-11 Fuel Category 4: STD/RFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	-0.6112	4.6655	6.7127	-21.8911
5	-0.3326	2.0713	12.8468	-26.1880
10	-0.1305	0.0505	18.3242	-30.7080
15	0.1360	-2.6856	26.5239	-38.3300
20	0.2321	-3.7177	29.5977	-41.1200

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 the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \qquad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 1.7 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 1.7 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-12 Fuel Category 4: Example STD/RFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.7	**3**	**4**	**5**
0	0	23.734	40.477	51.916
5	0	21.836	37.008	48.240
10	0	21.183	34.918	45.812
15	0	20.735	33.344	44.140
20	0	20.480	32.371	42.890

Note:

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

Table 6-13 Fuel Category 4: OFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.4957	-6.0715	37.2851	-49.1282
5	0.7476	-8.7581	45.3241	-56.5172
10	0.9041	-10.4334	50.3246	-61.0800
15	1.0799	-12.2326	55.7508	-66.1820
20	1.2541	-13.9154	60.5977	-70.5720

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 the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \qquad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 1.75 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 1.75 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-14 Fuel Category 4: Example OFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.75	3	4	5
0	0	21.401	34.600	47.456
5	0	20.806	32.495	44.606
10	0	20.401	31.027	42.668
15	0	20.138	30.017	41.691
20	0	19.845	29.273	41.277

Note:

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

6.2 ANALYSIS AREA OF APPLICABILITY

This section details the area of applicability of the analysis concerning assembly characteristics and associated fuel management, including a summary of the data which needs to be confirmed to assure that the results presented here remain valid. Additionally, restrictions are given for other normal SFP conditions. Farley Units 1 & 2 have operated with the STD and OFA fuel designs. [

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Table 6-15 Criticality [ ]a,c

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Additional restrictions for fuel storage are given here. One assembly pitch is defined as one cell in any direction, including both face adjacent and corner adjacent cells.

- Fuel assembly evolutions (fuel cleaning, inspection, reconstitution, and sipping) 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.

- Fuel assemblies stored with one or more rods missing, leaving a water hole, need to be stored as fresh fuel.

- Fuel assemblies which have had fuel rods replaced with SS, natural uranium, or zirconium alloy rods may be stored as normal (by initial enrichment and burnup).

- Reconstituted fuel which contains fuel rods from other fuel assemblies will be controlled as follows:

    1. The fuel assembly enrichment will be assumed to be the higher of the inserted rod or reconstituted fuel assembly's initial enrichment; and
    2. The fuel assembly burnup will be assumed to be the lower of the reconstituted rod or reconstituted fuel assembly's burnup.

- 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.

- An inspection can occur within the 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.

- 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 in a storage array.

6.3 SOLUBLE BORON CREDIT

Soluble boron is credited in the Farley Units 1 & 2 SFPs to keep $k_{eff} < 0.95$ under all normal and credible accident scenarios. Under normal conditions, this requires less than 320 ppm of soluble boron. Under accident conditions the most limiting accident is the multiple misload accident requiring 1710 ppm of soluble boron.

7 REFERENCES

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17. ML19189A111, "Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report 3002010613, 'Benchmarks for Qualifying Fuel Reactivity Depletion Uncertainty—Revision 1' and Topical Report 3002010614, 'Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation—Revision 1,'" U.S. Nuclear Regulatory Commission, 2019.

APPENDIX A
VALIDATION OF SCALE 6.2.3

A.1 INTRODUCTION

This validation suite is intended to be used for fresh and spent fuel storage in the Farley Units 1 & 2 Spent Fuel Pool Criticality Safety Analysis. [

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In order to validate the Scale Version 6.2.3 code system with the 238-group ENDF/B-VII library (referred to hereafter as Scale) for the Farley Units 1 & 2 Spent Fuel Pool criticality safety analysis, guidance from the NRC publication "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (Reference A1) was used and, as recommended in Reference A1, the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (Reference A2), has been used as the primary source of critical benchmarks for the validation effort. References A3 through A7 were also used as additional sources of critical benchmarks.

Section 3 in each of the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (Reference A2) individual evaluations provides benchmark material compositions as number densities which were reviewed and used for modeling experiments.

Per Reference A1, the following are important parameters when defining the area of applicability of a benchmark suite: fissile isotope, enrichment of the fissile isotope, fuel density, fuel chemical form, type of neutron moderators and reflectors, range of moderator to fissile isotope, neutron absorbers, and physical configurations. Therefore, these were the parameters considered when choosing which critical experiments to include in this validation suite.

This validation suite is designed to cover fresh and spent fuel storage for Farley Units 1 & 2. It also covers the criticality analysis of all normal operations and postulated accidents in the SFPs and fresh fuel storage. The validation is adequate to cover all present and anticipated (non-mixed-oxide) light water reactor (LWR) fuel designs at Farley.

A.2 METHOD DISCUSSION

The validation methods recommended in Reference A1 are the basis of this validation of Scale for nuclear criticality safety calculations. The code methodology bias and the uncertainty associated with the bias will be used in combination with other biases and uncertainties, as well as additional subcritical margin to ensure the regulatory requirements are met. Statistical analysis is performed to determine whether trends exist in the bias for three subsets of experiments; fresh fuel with strong absorbers, fresh fuel without strong absorbers, and fresh and burnt fuel with strong absorbers. No critical experiments containing Gadolinia, IFBA or Erbia were used because they will not be credited in the Farley Criticality Safety Analysis either as fresh or residual absorbers.

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According to NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data" (Reference A4), the HTC experiments are a series of experiments performed with mixed oxide rods designed to have U and Pu isotopic compositions equal to that of UO_2 PWR fuel with initial enrichment of 4.50 wt% ^{235}U at 37,500 MWd/MTU burnup. No fission products are included in the compositions. The HTC experiments are included to ensure the validation suite covers spent fuel as well.

Normality testing for the data subsets is performed as outlined in References A1 and A8 using the Shapiro-Wilk test for data sets with a sample size of 50 or less and the D'Agostino normality test for the data sets with a sample size of more than 50. For the cases which fail the normality tests, the non-parametric statistical treatment recommended in Reference A1 is used.

A.2.1 Test for Normality (Goodness-of-Fit Test)

As stated in Reference A1, the statistical evaluation performed must be appropriate for the distribution of the data. A goodness-of-fit test is a procedure designed to examine whether a sample has come from a postulated distribution. Among the methods for testing goodness-of-fit, some are superior to others in their sensitivity to different types of departures from the hypothesized distribution. Some of the tests are quite general in that they can apply to just about any distribution, while other tests are more specific, such as tests that apply only to the normal distribution. [

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A.2.1.1 Shapiro-Wilk Test for Normality

References A1 and A8 discuss the Shapiro-Wilk test for normality (W-test). The W-test is applicable when neither the population mean (μ) nor the population standard deviation (σ) is specified. The W-test is considered an omnibus test for normality because of its superiority to other procedures over a wide range of problems and conditions that depend on an assumption of normality. The W-test is superior to the chi-square test (used by USLSTATS from Scale package) in many situations. This analysis thus uses the W-

test as recommended in Reference A8 for sample sizes between 3 and 50, the range over which Table T-6b provides the critical value $w_q(n)$.

The null and alternative hypotheses are:

H_0: The sample comes from a normal distribution.

H_1: The underlying distribution is not normal.

The W-test statistic is:

$$W = \frac{B^2}{(n-1)S^2} \qquad \text{Equation A-1}$$

where,

n is the number of experiments in the group,
S^2 is the dataset variance, and

$$B = \sum_{i=1}^{k} a_i \left(y_{(n-i+1)} - y_{(i)}\right) \qquad \text{Equation A-2}$$

where,

k = $n/2$ if n is even or $(n-1)/2$ if n is odd
a_i = i coefficients obtained from Table T-6a of NUREG-1475, "Applying Statistics" (Reference A8) associated with sample size n

$\{y_{(1)}, y_{(2)}, \ldots, y_{(n)}\}$ is the normalized k_{eff} of each experiment arranged in ascending order

The null hypothesis H_0 of normality is rejected at the α level of significance if the calculated value of W is less than the critical value $w_q(n)$ obtained from Table T-6b of Reference A8. Note that in this table, the quantile $q=\alpha$.

A.2.1.2 D'Agostino Test for Normality

Reference A8 discusses the D'Agostino test for normality (D' test). Like the W-test, the D' test is also applicable when neither μ nor σ is specified. Like the W-test, the D' test is also considered an omnibus test for normality because of its superiority to other procedures over a wide range of problems and conditions that depend on an assumption of normality. The D' test complements the W-test, which is used for samples no larger than 50, and can be used for any sample size greater than 50.

The null and alternative hypotheses for D' test are:

H_0: The sample comes from a normal distribution.

H_1: The underlying distribution is not normal.

The test statistic is:

$$D' = \frac{T}{\sqrt{S^2(n-1)}} \qquad \text{Equation A-3}$$

where,

n is the number of experiments
S^2 is the dataset variance

$$T = \sum_{i=1}^{n}\left(i - \frac{(n+1)}{2}\right) y_{(i)} \qquad \text{Equation A-4}$$

$\{y_{(1)}, y_{(2)}, \ldots, y_{(n)}\}$ is the normalized k_{eff} of each experiment arranged in ascending order

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]^{a,c} The D' test involves a comparison of the calculated D' value with two quintiles from Table T-14 of Reference A8. The test is two-sided and requires two critical values that bound a noncritical region. For each combination of n and α, the critical values are found in Table T-14 under the row that corresponds to n and the columns for $q_{\alpha/2}(n)$ and $q_{1-\alpha/2}(n)$. If the calculated D' is not between these two values, the null hypothesis is rejected.

If the null hypothesis is rejected, a non-parametric treatment may be applied. If the null hypothesis is not rejected, then a technique such as a one-sided tolerance limit described in Reference A1 can be used to determine the appropriate bias and bias uncertainty.

A.2.2 Determination of Bias and Bias Uncertainty

The statistical analysis presented in Section 2.4 of Reference A1 is followed for all datasets that passed the appropriate test for Normality. This approach involves determining a weighted mean that incorporates the uncertainties from both the measurement (σ_{exp}) and the calculation method (σ_{calc}). The benchmark experiments chosen from References A2, A6, and A7, use the experimental uncertainties presented in References A2, A6 and A7, respectively. Experimental uncertainty is not presented for the experiments contained in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages" (Reference A3), so the average value of experimental uncertainties of similar experiments documented in Reference A2 is used. This is consistent with the recommendation in Reference A1 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 calculated k_{eff} (k_{calc}) is divided by the k_{eff} evaluated in the experiment (k_{exp}):

$$k_{normal} = \frac{k_{calc}}{k_{exp}}$$ Equation A-5

The normalized k_{eff} (k_{normal})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 calculational uncertainty (σ_{calc}) and experimental uncertainties (σ_{exp}) are root-sum-squared to create a combined uncertainty (σ_t) for each experiment:

$$\sigma_t = \sqrt{\sigma_{calc}^2 + \sigma_{exp}^2}$$ Equation A-6

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 "ith" member of that set is shown with a subscript "*i*." Henceforth, unless otherwise specified, the combined uncertainty σ_t for an "ith" k_{eff} is shown as σ_i. The weighted equation variables for the single-sided lower tolerance limit are as follows:

Variance about the mean:

$$s^2 = \frac{\left(\frac{1}{n-1}\right)\sum\frac{1}{\sigma_t^2}\left(k_{eff_i} - \bar{k}_{eff}\right)^2}{\frac{1}{n}\sum\frac{1}{\sigma_t^2}}$$ Equation A-7

Average total uncertainty:

$$\bar{\sigma}^2 = \frac{n}{\sum\frac{1}{\sigma_t^2}}$$ Equation A-8

The weighted mean k_{eff} value:

$$\bar{k}_{eff} = \frac{\sum\frac{1}{\sigma_t^2}k_{eff_i}}{\sum\frac{1}{\sigma_t^2}}$$ Equation A-9

The square root of the pooled variance:

$$S_P = \sqrt{s^2 + \bar{\sigma}^2}$$ Equation A-10

where,

s^2 = variance about the mean
n = number of critical experiments used in the validation
$\bar{\sigma}$ = average total uncertainty

Bias is determined by the relation:

$$Bias = \begin{matrix} \bar{k}_{eff} - 1.0 & if\ \bar{k}_{eff} < 1.0 \\ 0.0 & if\ \bar{k}_{eff} \geq 1.0 \end{matrix} \qquad \text{Equation A-11}$$

Reference A1 states that when a relationship between a calculated k_{eff} and an independent variable cannot be determined (no trend exists), a one-sided lower tolerance limit should be used. This method provides a single lower limit above which a defined fraction of the true population of k_{eff} is expected to lie, with a prescribed confidence and within the area of applicability. Use of this method requires the experimental results to have a normal statistical distribution. Lower tolerance limits, at a minimum, should be calculated with a 95% confidence that 95% of the data lies above K_L. The equation for the one-sided lower tolerance band from Reference A1 is:

$$K_L = \bar{k}_{eff} - US_P \qquad \text{Equation A-12}$$

Or, if $\bar{k}_{eff} \geq 1$,

$$K_L = 1 - US_P \qquad \text{Equation A-13}$$

Where, S_P is the pooled variance,

U is the one sided lower tolerance factor (found in Table T-11b of Reference A8 where n is the number of experiments contained in the data set).

US_P is then taken as the uncertainty to the untrended bias (untrended bias uncertainty).

A.2.3 Identify Trends in the Data

Trends are determined using 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 evaluate whether there is a trend in the bias. In the following equations, "x" is the independent variable representing the parameter of interest (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. The function $Y(x)$ represents $K_{fit}(x)$.

Per Reference A1, the equations used to produce a weighted fit of a straight line to the data are given in this section.

$$Y(x) = a + bx \qquad \text{Equation A-14}$$

where,

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

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

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

Once the data has been fit to a line, a determination as to the "goodness of fit" must be made. Per Reference A1, two steps should be employed when determining the goodness of fit. The first step is to plot the data against the independent variable which allows for a visual evaluation of the effectiveness of the regression fit.

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 A1, the linear correlation coefficient is one standard method used to numerically measure the linear association between the random variables x and y.

The sample correlation coefficient between x and y (linear-correlation coefficient) is a quantitative measure of the degree to which a linear association exists between two variables. For weighted data, the linear correlation coefficient is:

$$r = \frac{\sum \frac{1}{\sigma_i^2} (x_i - \bar{x})(y_i - \bar{y})}{\sqrt{\sum \frac{1}{\sigma_i^2} (x_i - \bar{x})^2} \sqrt{\sum \frac{1}{\sigma_i^2} (y_i - \bar{y})^2}} \qquad \text{Equation A-15}$$

where,

The weighted mean for the independent parameter is:

$$\bar{x} = \frac{\sum \frac{1}{\sigma_i^2} x_i}{\sum \frac{1}{\sigma_i^2}} \qquad \text{Equation A-16}$$

The weighted mean for the dependent parameter ($\bar{y}$) is $\bar{k}_{eff}$.

The value of r^2 is the coefficient of determination. It can be interpreted as the percentage of variance of one variable that is predictable from the other variable. The closer r^2 approaches the value of 1, the better the fit of the data to the linear equation. Note that the value of a sample correlation coefficient r shows only the extent to which x and y are linearly associated. It does not by itself imply that any sort of causal relationship exists between x and y.

In addition to the linear correlation coefficient, the Student's t test is used to determine if the trend in the linear fit of the data is statistically significant. A trend is statistically significant when the slope of the linear regression fit (b) is equal to some specified value (b_0). For the purposes of this validation suite, the null hypothesis, H_0. $b_0 = 0$ is that no statistically significant trend exists (slope is zero) with an alternative hypothesis of H_1: $b_0 \neq 0$, at a significance level of $\alpha = 0.05$.

In order to determine if the null hypothesis is supported, t_{score} is calculated and compared to the Student's t distribution ($t_{\alpha/2,n-2}$). The t_{score} for the slope of a regression line is given by:

$$t_{score} = \frac{(b - b_0)\sqrt{n-2}}{\sqrt{\frac{SSE}{\sum(x_i - \bar{x})^2}}}$$ Equation A-17

where,

SSE is the sum of the squares of the residuals:

$$SSE = \sum[k_{eff_i} - (a + bx_i)]^2$$ Equation A-18

The null hypothesis is rejected if $|t_{score}| > t_{\alpha/2,\, n-2}$.

When H_0 is rejected and a statistically significant trend is determined, the trended value of a bias and its associated uncertainty are used when it is more restrictive than the untrended value of the bias. In the area where untrended bias yields more restrictive value, the untrended bias and its associated uncertainty are used.

Per Reference A1, when a relationship between a calculated k_{eff} and an independent variable can be determined (the trend exists), a one-sided lower tolerance band may be used. This conservative method 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 from Reference A1 is:

$$K_L(x) = K_{fit}(x) - S_{P_{fit}}\left\{\sqrt{2F_a^{(2,n-2)}\left[\frac{1}{n} + \frac{(x-\bar{x})^2}{\sum(x_i-\bar{x})^2}\right]} + z_{2P-1}\sqrt{\frac{(n-2)}{\chi^2_{1-\gamma,n-2}}}\right\}$$ Equation A-19

$K_{fit}(x)$ is the function derived in the trend analysis described above. Because a positive bias may not be conservative, the following equation must be used for all values of x where $K_{fit}(x) > 1$:

$$K_L(x) = 1 - S_{P_{fit}}\left\{\sqrt{2F_a^{(2,n-2)}\left[\frac{1}{n} + \frac{(x-\bar{x})^2}{\sum(x_i-\bar{x})^2}\right]} + z_{2P-1}\sqrt{\frac{(n-2)}{\chi^2_{1-\gamma,n-2}}}\right\}$$ Equation A-20

where,

p = The desired confidence level (0.95)

$F_a^{(2,n-2)}$ = 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

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

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

For a weighted analysis:

$$\sum(x_i - \bar{x})^2 = \frac{\sum \frac{1}{\sigma_i^2}(x_i-\bar{x})^2}{\frac{1}{n}\sum \frac{1}{\sigma_i^2}} \quad \text{Equation A-21}$$

$$\bar{x} = \frac{\sum \frac{1}{\sigma_i^2}x_i}{\sum \frac{1}{\sigma_i^2}} \quad \text{Equation A-22}$$

$$S_{P_{flt}} = \sqrt{s_{flt}^2 + \bar{\sigma}^2} \quad \text{Equation A-23}$$

$$\bar{\sigma}^2 = \frac{n}{\sum \frac{1}{\sigma_i^2}} \quad \text{Equation A-24}$$

$$s_{flt}^2 = \frac{\left(\frac{1}{n-2}\right)\sum\left\{\frac{1}{\sigma_i^2}\left[k_{eff_i} - K_{flt}(x_i)\right]^2\right\}}{\frac{1}{n}\sum \frac{1}{\sigma_i^2}} \quad \text{Equation A-25}$$

Within the equation for K_L:

$$Bias(x) = \begin{matrix} K_{flt} - 1.0 & if\ K_{flt} < 1.0 \\ 0.0 & if\ K_{flt} \geq 1.0 \end{matrix} \quad \text{Equation A-26}$$

And the uncertainty in the bias is:

$$95/95\ Bias\ Uncertainty(x) = S_{P_{flt}}\left\{\sqrt{2F_a^{(2,n-2)}\left[\frac{1}{n} + \frac{(x-\bar{x})^2}{\sum(x_i-\bar{x})^2}\right]} + z_{2P-1}\sqrt{\frac{(n-2)}{\chi^2_{1-\gamma,n-2}}}\right\} \quad \text{Equation A-27}$$

When H_0 is rejected and a statistically significant trend is determined, the trended value of a bias and its associated uncertainty should be used while it is more restrictive than the untrended value of the bias. In the area where an untrended bias yields a more restrictive value, the untrended bias and its associated uncertainty shall be used.

A.2.4 Non-Parametric Treatment

If the data fails the test for normality, a non-parametric treatment of the data will be necessary. Per Reference A1, the determination of K_L, the lower limit of the 95/95 tolerance interval is as follows:

$$K_L = k_{eff}^{min} - uncertainty\ for\ k_{eff}^{min} - NPM \qquad \text{Equation A-28}$$

where,

k_{eff}^{min} is the minimum (smallest) normalized k_{eff} in a dataset,

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

NPM is the non-parametric margin, which is added to account for the small sample size.

The non-parametric treatment outlined in Reference A1 uses the order statistics to represent the characteristics of a dataset after it has been ranked (ordered) from the smallest observed k_{eff} (k_{eff}^{min}) to the largest observed k_{eff} (k_{eff}^{max}). The following equation is the general equation that determines the percent confidence that a fraction of the population is above the lowest observed value:

$$\beta = \left[1 - \sum_{j=0}^{m-1} \left(\frac{n!}{j!(n-j)!}\right)(1-q)^{j} q^{n-j}\right] \text{ x } 100\% \qquad \text{Equation A-29}$$

where:

q is the desired population fraction (normally 0.95)

n is the number of data values in one data set

m is the rank order indexing from the smallest sample value to the largest (m = 1 for the smallest sample value; m = 2 for the second smallest sample value, etc.).

The smallest observed k_{eff} has the rank order index 1 and the largest observed k_{eff} has the rank order index equal to the number of observations. Thus, for a desired population fraction of 95% and k_{eff}^{min} (rank order index 1), the percent confidence that a fraction of the population of n data points is above the lowest observed value is:

$$\beta = (1 - 0.95^{n}) \times 100\% \qquad \text{Equation A-30}$$

Similarly, for a desired population fraction (q) of 95% and the 2nd lowest k_{eff} (rank order index m=2), the percent confidence that a fraction of the population of n data points is above the second lowest observed value is:

$$\beta = \left[1 - \left[0.95^{n} + \frac{n!}{(n-1)!} * (1 - 0.95) * 0.95^{n-1}\right]\right] \text{ x } 100\% \qquad \text{Equation A-31}$$

Although 59 experiments would be required to reach a 95/95 tolerance limit for rank order 1 as stated in Reference A1, the recommended non-parametric margin (NPM) correction is 0.0 for confidence values greater than 90 percent, as also indicated in Table 2.2 of Reference A1.

Within the equation for K_L:

$$Bias = \begin{matrix} k_{eff}^{min} - 1.0 & if\ k_{eff}^{min} < 1.0 \\ 0.0 & if\ k_{eff}^{min} \geq 1.0 \end{matrix} \quad \text{Equation A-32}$$

And the uncertainty in the bias is:

$$Bias\ Uncertainty = u\sqrt{{\sigma_{calc}}^2 + {\sigma_{exp}}^2} \quad \text{Equation A-33}$$

where,

${\sigma_{calc}}^2$ is the Monte Carlo uncertainty from the selected rank order case
${\sigma_{exp}}^2$ is the experimental uncertainty from the selected rank order case

A rank order of 2 may be used in determination of the percent of confidence that a fraction of the population is above the lowest observed value if the sample size is greater than 93. This effectively means that the second lowest k_{eff} is used for the determination of the Bias and Bias uncertainty. Employing the aforementioned methods will produce a particularly conservative bias and bias uncertainty and negate the need for any further trending analysis for a non-parametric data set.

A.3 DESCRIPTION OF CRITICAL EXPERIMENTS

Many studied series of the critical experiments allow using a simplified model with some zones homogenized or omitted. Only the complete model provided in Section 3.0 of each evaluated series of experiments is used for k_{eff} determination.

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A.3.2 [

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A.3.3 [

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A.3.4 [

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A.3.5 [

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A.3.6 [

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A.3.7 [

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A.3.8 [

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A.3.9 [

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A.3.10 [

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A.3.11 [

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Table A-1 Benchmark Values of k_{eff} and Respective Uncertainties a,c

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NUREG/CR-6361 (Reference A3) is intended as a guide for performing criticality benchmark calculations for LWR fuel applications. It documents 180 critical experiments and includes recommendations for selecting suitable experiments and determining the calculational bias and bias uncertainty. When selecting experiments, preference is given to Reference A2 because it is more current than Reference A3. [

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

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A.3.17 HTC Experiments

The HTC experiments are a series of experiments performed with mixed oxide rods designed to have a U and Pu isotopic composition representative to that of U(4.5%)O_2 PWR fuel with 37,500 MWd/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 was taken from two phases:

- Phase 1 – Water-Moderated and Reflected Simple Arrays (Reference A5)

- Phase 2 – Reflected Simple Arrays Moderated by Water Poisoned with Gadolinium or Boron (Reference A6)
- Phase 3 – Pool Storage (Reference A7)

Reference A4 is an ORNL evaluation of the HTC experiments. [

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A.4 RAW CALCULATION RESULTS

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Definitions of k_{calc}, k_{exp}, k_{normal} and their associated uncertainties are explained in Section A.2.

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Table A-3 [ ]^{a,c} a,c

Table A-3 [ ]^{a,c} (cont.)

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Table A-3 [ ]a,c **(cont.)** a,c

Table A-3 [ ]^{a,c} (cont.) a,c

Table A-3 [ ]^{a,c} (cont.) a,c

Table A-4 [**]**[a,c]** n,c

Table A-4 [ ]^{a,c} (cont.) a,c

Table A-4 [ ]^{a,c} (cont.) a,c

Table A-4 [ ]^{a,c} (cont.)

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

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Table A-5 [ ]^{a,c} (cont.) a,c

Table A-5 [ ]^{a,c} (cont.) a,c

Table A-5 [ ]a,c (cont.) a,c

Table A-5 [ ]a,c (cont.) a,c

Table A-6 [

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Table A-6 [ ]a,c

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Table A-6 [ ]a,c

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Table A-6 [ ]a,c

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

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A.5 DATA SET NORMALITY ASSESSMENT

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Table A-7 [ ]a,c a,c

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A.5.2 [ ]a,c

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Table A-8 [ ]a,c a,c

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A.5.3 [ ]a,c

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Table A-9 [ ]a,c

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Note that the quintiles, $D'_q(n)$, of the distribution of the D' statistic in Reference A8 are provided only for even n. For the odd n, linear interpolation is used between adjacent values.

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Table A-10 [ ]a,c

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A.6 TRENDING ANALYSIS

Of the four main datasets evaluated, trending analysis will only be performed on the [

]a,c The regression fits and goodness of fit tests described in Section A.2.1 are applied [

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A.6.1 [ ]a,c

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Table A-11 [

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Statistical Data	Parameter

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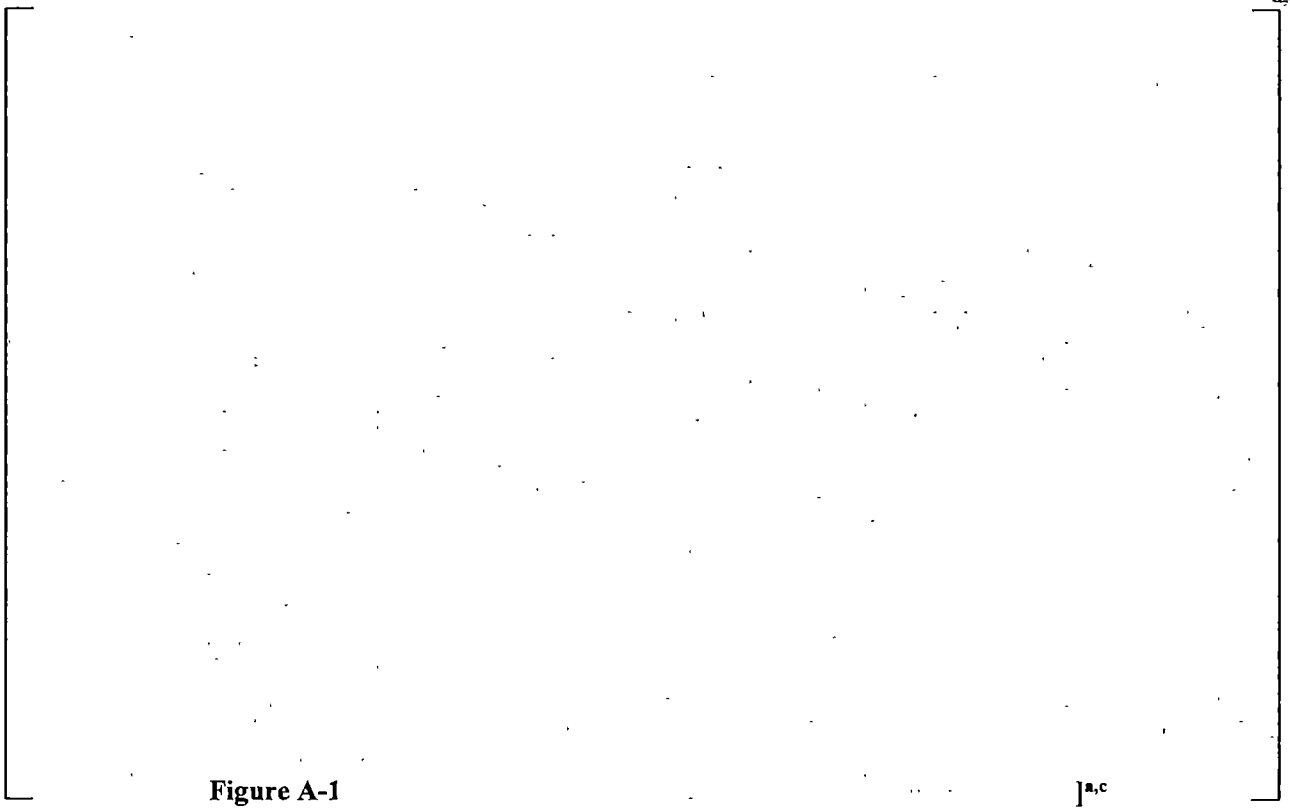

Figure A-1]a,c

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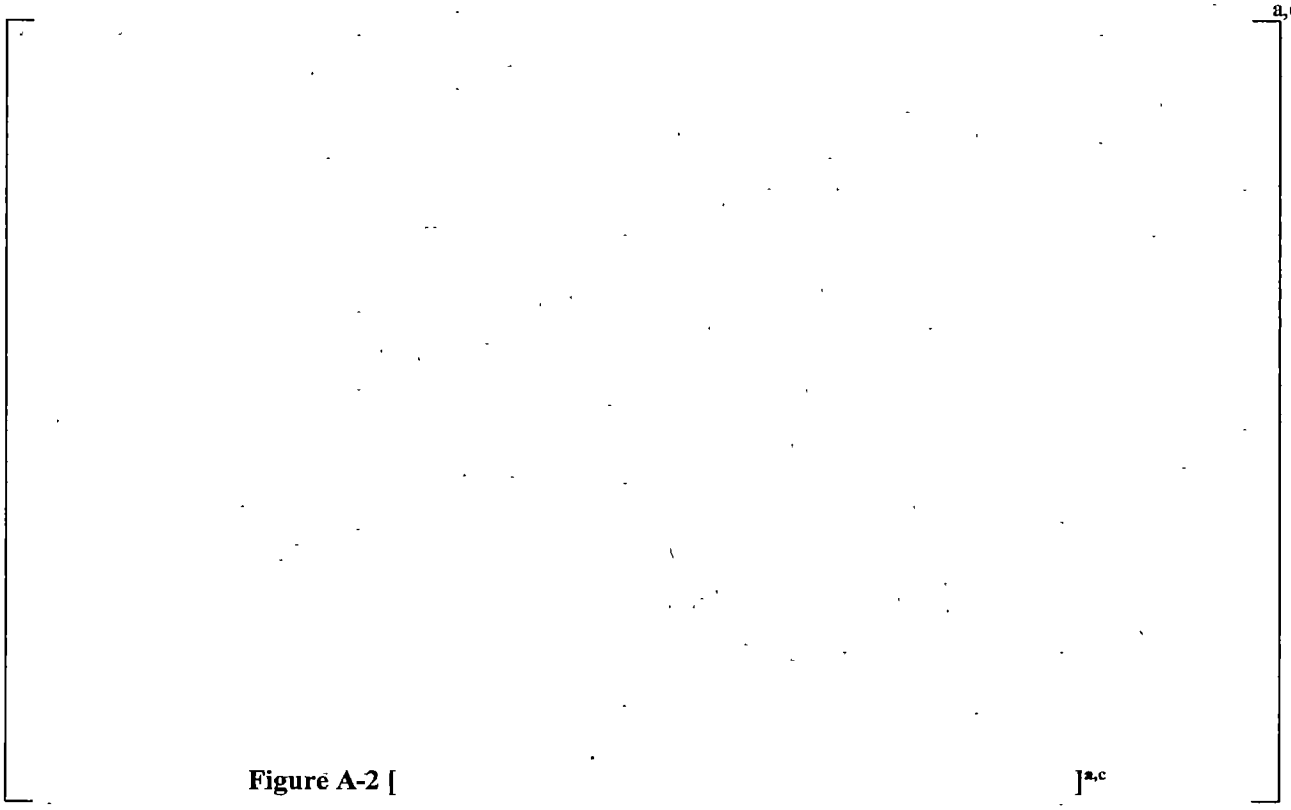

Figure A-2 [ ]a,c

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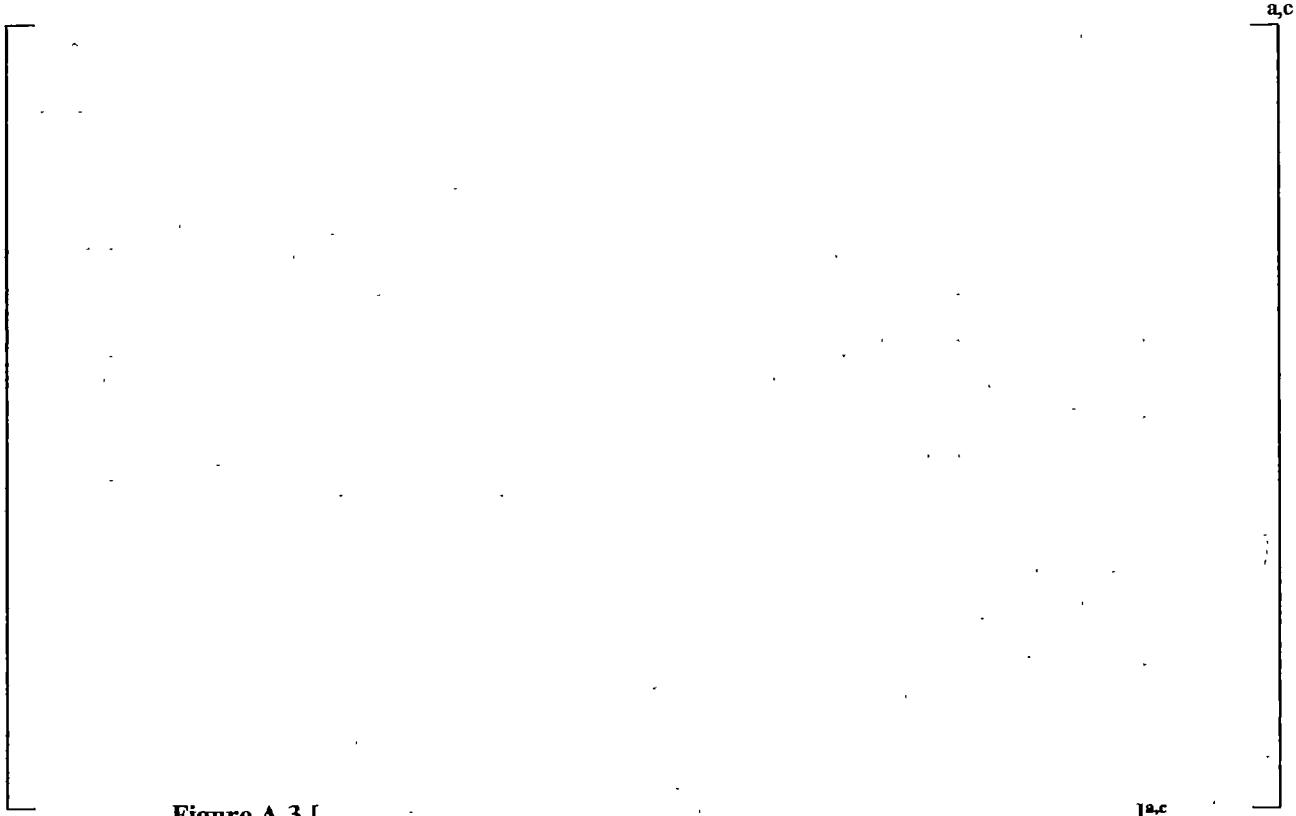

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Figure A-3 [ ]a,c

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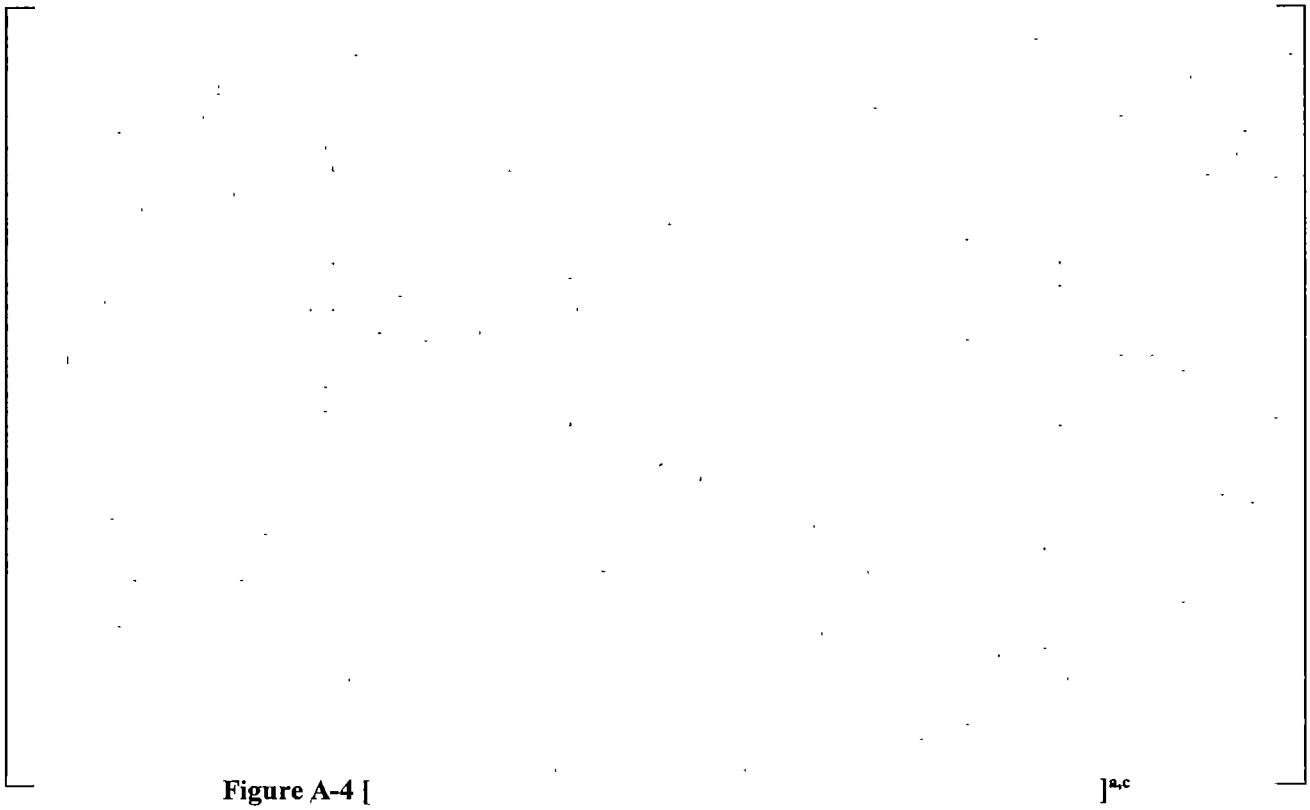

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Figure A-4 [ ]a,c

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A.6.1.1 [ ]a,c

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Table A-12 [ ]a,c

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Table A-13 [

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Table A-14 [ ]a,c

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Figure A-5 [ ]a,c

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Table A-15 [ ]a,c

Parameter	
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A.6.2 [ ]a,c

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Table A-16 [ ]a,c

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A.7 AREA OF APPLICABILITY

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

Table A-17 [ ]a,c

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A.8 VALIDATION SUMMARY

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Table A-18 Summary of Biases and Bias Uncertainties Determination a,c

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with the trended bias uncertainty determined as

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A.9 REFERENCES

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