ATTACHMENT 5

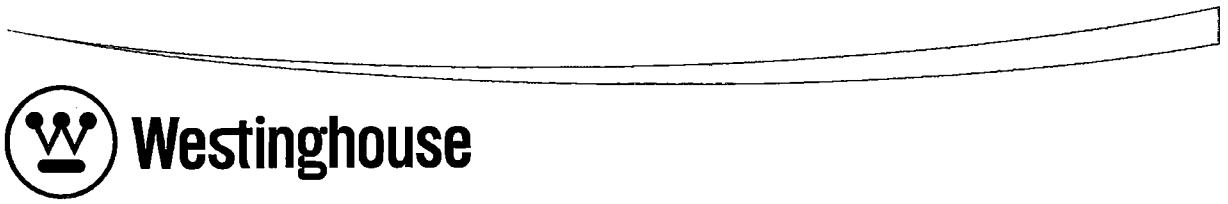
Westinghouse Electric Company

WCAP-18030-NP, Revision 1, *Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3* (Non-proprietary), dated October 2016

WCAP-18030-NP
Revision 1

October 2016

Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3



WCAP-18030-NP
Revision 1

Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3

Michael T. Wenner*
Core Engineering & Software Development

October 2016

Reviewer: Susan M. Nelson*
Core Engineering & Software Development

Approved: Douglas E. Sipes*, Manager
Core Engineering & Software Development

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

REVISION HISTORY

Revision	Description and Impact of the Change	Date
0-A	This is the draft version.	07/2015
0-B	This is the second draft version.	08/2015
0	Original Issue	09/2015
1-A	Draft Revision 1 Update to Address NRC Round 1 Requests for Additional Information	10/2016
1	Revision 1 Update to Address NRC Round 1 Requests for Additional Information	10/2016

TRADEMARK NOTICE

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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
ANP	Designation for fuel made by Siemens/Framatome
AoA	Area of Applicability
at%	Atom Percent
B&W	Babcock and Wilcox
BA	Burnable Absorber
BORAL	BORAL
CE	Combustion Engineering
C.E.A.	Commissariat à l'Energie Atomiqueo
CEA	Control Element Assembly
Decay time	Post-irradiation cooling time
EALF	Energy of Average Lethargy causing Fission
En	Enrichment
ENDF/B	Evaluated Nuclear Data File
FHE	Fuel Handling Equipment
FOSAR	Foreign Object Search and Retrieval
FRSB	Fuel Rod Storage Basket
GT	Guide Tube
GWd	Gigawatt-days
HTC	Haut Taux de Combustion
ID	Inner Dimension
IFBA	Integral Fuel Burnable Absorber
IFSR	Intermediate Fuel Storage Rack
I.P.S.N.	Institut de Protection et de Sûreté Nucléaire, now called Institut de Radioprotection et de Sûreté Nucléaire
ISG	Interim Staff Guidance
IT	Instrumentation Tube
k_{eff}	Effective neutron multiplication factor
KENO	SCALE Module KENO V.a
LEU	Low-Enriched Uranium
LUA	Lead Use Assembly
LWR	Light Water Reactor
MTU	Metric Ton Uranium
MWt	Megawatts-thermal
NFSR	New Fuel Storage Rack
NGF	Next Generation Fuel
NITAWL	NITAWL-III
NPM	Non-Parametric Margin
NRC	U.S. Nuclear Regulatory Commission
OD	Outer Dimension

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

ORNL	Oak Ridge National Lab
Palo Verde	Palo Verde Nuclear Generating Station
PNNL	Pacific Northwest National Laboratory
ppm	parts per million
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
SFP	Spent Fuel Pool
SNAP-IN	NETCO-SNAP-IN
SRSC	Service de Recherche en Sûreté Criticité, now called Service de Recherche en Neutronique et Sûreté Criticité
STD	Standard Fuel Assembly
TD	Theoretical Density
VAP	Value Added Pellet
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 Palo Verde Nuclear Generating Station (Palo Verde) spent fuel pools (SFPs), new fuel storage racks (NFSRs), interim fuel storage rack (IFSR), and fuel handling equipment (FHE). The report considers Palo Verde's past, current, and planned future operating history and fuel designs.

The main report details the SFP criticality safety analysis. Appendix A details the validation of the code used for eigenvalue calculations, and Appendix B details the analysis performed for the NFSR, IFSR, and FHE.

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). Additionally, some SFP storage arrays credit the presence of the neutron poison present in the NETCO-SNAP-IN® Neutron Absorber (SNAP-IN) insert. Finally, credit is taken for the presence of soluble boron in the SFP.

2.1 ACCEPTANCE CRITERIA

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

1. The effective neutron multiplication factor (k_{eff}) of all permissible storage arrangements at a soluble boron concentration of 0 parts per million (ppm) shall be less than 1.0 including 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 less than or equal to 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 be less than 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 SFP, compliance is demonstrated by establishing limits on the minimum allowable burnup as a function of enrichment and decay time for each fuel storage array. A conservative combination of best estimate and bounding values have been selected to model the fuel in this analysis to ensure that fuel represented by the proposed Palo Verde Technical Specifications is less reactive than the fuel modeled in this analysis. Therefore, burnup limits generated here will conservatively bound all fuel to be stored in the Palo Verde SFP. Input selection is discussed in Section 4 of this analysis.

The acceptability of the storage arrays developed in this analysis is ensured by controlling the assemblies that can be stored in each array. Assemblies are divided into Fuel Regions 1 through 6 based on assembly reactivity determined as a function of assembly average burnup, initial enrichment[1], and decay time. An assembly's fuel region determines which storage arrays are acceptable for the assembly in question. Fuel Region 1 defines the most reactive assemblies, i.e., representing a fresh 4.65 weight percent (wt%)

1. Initial enrichment is the maximum radial average ^{235}U enrichment of the central zone region of fuel, excluding axial cutbacks, 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.

^{235}U assembly and Fuel Region 6 defines the least reactive assemblies, i.e., representing assemblies that can be stored in Array F (see Table 2-1).

Table 2-1 Fuel Regions Ranked by Reactivity

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

Notes:
1. Fuel Regions are defined by assembly average burnup, initial enrichment and decay time as provided by Tables 6-3, 6-5, 6-7 and 6-9.
2. Fuel Regions are ranked in order of decreasing reactivity, e.g., Fuel Region 2 is less reactive than Fuel Region 1, etc.
3. Fuel Region 1 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U. No burnup is required.
4. Fuel Region 2 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U with at least 16.0 GWd/MTU of burnup.
5. Fuel Regions 3 through 6 are determined from the minimum burnup equation and coefficients provided in Tables 6-2, 6-4, 6-6 and 6-8.
6. Assembly storage is controlled through the storage arrays defined in Figure 5-1.
7. Each storage cell in an array can only be populated with assemblies of the Fuel Region defined in the array definition or a lower reactivity Fuel Region.

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 2) and its cross-section library based on Evaluated Nuclear Data File Version VI.3 (ENDF/B-VI.3), and (2) SCALE Version 6.1.2, as documented in ORNL/TM-2005/39, "SCALE: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation" (Reference 3), with the 238-group cross-section library based on ENDF/B-VII.

2.3.1 Two-Dimensional Transport Code PARAGON

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

PARAGON is the Westinghouse Electric Company LLC state-of-the-art 2-D lattice transport code. 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 for 174 isotopes, without any lumped fission products or pseudo cross-sections. PARAGON and its 70-group cross-section library are benchmarked, qualified, and licensed both as a standalone transport code and as a nuclear data source for a core simulator in a complete nuclear design code system for core design, safety, and operational calculations. The list of fuel isotopes modeled in PARAGON and subsequently modeled in the criticality analysis are given in Table 2-2.

Table 2-2 Isotopes Used in the Nuclear Criticality Safety Analysis a,c

[ ]

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

2.3.2 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. 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 SFP, as well as finite rack module and SFP representations to evaluate interfaces, soluble boron requirements, and postulated accident scenarios.

The SCALE package includes the control module CSAS5 and the following functional modules: BONAMI, CENTRM, PMC, and Scale Module KENO V.a (KENO). A brief description of each module follows:

- The BONAMI module utilizes Dancoff approximations to perform Bondarenko unresolved resonance self-shielding calculations. BONAMI solves problems in a one-dimensional multizone slab, cylindrical, or spherical geometry. Heterogeneous effects are accounted for through the use of one-dimensional (1-D) Dancoff factors evaluated for the geometry of the problem, defined as separate regions of fuel, cladding, and moderator.

- CENTRM computes continuous-energy neutron spectra in zero- or one-dimensional systems, by solving the Boltzmann Transport Equation using a combination of pointwise and multigroup nuclear data. One of the major functions of CENTRM is to determine problem-specific fluxes for processing resonance-shielded multigroup data. This is done by performing a CENTRM calculation for a simplified system model (e.g., a 1-D unit cell either isolated or in a lattice by reflecting the surfaces), and then utilizing the spectrum as a problem-dependent weight function for multigroup averaging.

- 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 the resolved resonance 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.

- The KENO module is a multigroup Monte Carlo criticality program used to calculate the k_{eff} of 3-D models. Flexible geometry features and the availability of various boundary condition prescriptions in KENO allow for accurate and detailed modeling of fuel assemblies in storage racks, either as infinite arrays or in actual SFP models. Anisotropic scattering is treated by using discrete scattering angles through the use of P_n Legendre polynomials. KENO uses problem-specific cross-section libraries, processed for resonance self-shielding and for the thermal characteristics of the problem.

The criticality sequence of SCALE 6.1.2 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. The details of the validation are found in Appendix A. The validation shows that SCALE 6.1.2 is an accurate tool for calculation of k_{eff} for SFP applications. The benchmark calculations use the same computer platform and cross-section libraries as 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 (HTC criticals).

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 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 library with the SCALE Version 6.1.2 CSAS5 module is documented in Appendix A.

3 PALO VERDE NUCLEAR GENERATING STATION

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

3.1 REACTOR DESCRIPTION

Palo Verde is a System 80+ Combustion Engineering (CE) pressurized water reactor (PWR) utilizing fuel with a 16x16 lattice. Palo Verde has operated at licensed powers from 3800 Megawatts-thermal (MWt) to 3990 MWt, using multiple fuel designs from several fuel vendors. All fuel assemblies used at Palo Verde incorporate a 16x16 square array of 236 fuel rods with four corner guide tubes (GTs) and one instrumentation tube (IT). The fuel rod cladding material is Zircaloy and its variants, such as ZIRLO® High Performance Fuel Cladding Material. Each fuel rod contains a column of enriched UO_2 fuel pellets. The pellets are pressed and sintered, and are dished on both ends.

Table 3-1 provides basic data on the type of reactor and the fuel types that comprise Palo Verde. Section 3.1.1 describes the different fuel designs outlined in Table 3-1, including outlining neutronically important physical characteristics of each fuel design. Section 3.1.2 describes the non-mechanical fuel features which further subdivide mechanically identical fuel designs. Additionally, the fuel management data for Palo Verde Units 1, 2, and 3 is provided in Table 3-3 through Table 3-6.

Table 3-1 Reactor General Specifications

Parameter	Value
Reactor type	CE System 80+
Reactor power (MWt)[1]	3800 – 4070
Fuel lattice	16x16
Fuel design 1	CE Standard Fuel (STD)
Fuel design 2	CE Value Added Pellet (VAP)
Fuel design 3	Framatome ANP (ANP)
Fuel design 4	CE Next Generation Fuel (NGF)

Note:
1. Palo Verde's current licensed operating power is 3990 MWt.

3.1.1 Fuel Designs

This section outlines the neutronically important mechanical features of the four (4) fuel designs outlined in Table 3-1. The fuel designs are shown in Table 3-2.

Table 3-2 Fuel Assembly Mechanical Specifications

Parameter	Value			
Assembly type	STD	VAP	ANP	NGF
Rod array size	16x16	16x16	16x16	16x16
Rod pitch, inch	0.506	0.506	0.506	0.506
Active fuel length, inch	150	150	150	150
Total number of fuel rods	236	236	236	236
Fuel cladding Outer Dimension (OD), in	0.382	0.382	0.382	0.374
Fuel cladding Inner Dimension (ID), in	0.332	0.332	0.332	0.329
Fuel cladding thickness, in	0.025	0.025	0.025	0.0225
Pellet diameter, in	0.325	0.3255	0.3255	0.3225
Number of GT/IT	4/1	4/1	4/1	4/1
GT/IT OD, in	0.980/0.980	0.980/0.980	1.023/0.980	0.980/0.980
GT/IT thickness, in	0.04/0.04	0.04/0.04	0.0615/0.04	0.04/0.04

3.1.2 Fuel Management History

This section discusses the non-mechanical fuel features which are important to criticality safety and how they impact the number of distinct fuel designs to be considered in this analysis. Specifically, the presence and enrichment of axial cutbacks are taken into account, as well as the presence and amount of Burnable Absorber (BA) being used such as Integral Fuel Burnable Absorber (IFBA) or Erbia. Table 3-3, Table 3-4, and Table 3-5 provide the fuel management history for Palo Verde Units 1, 2, and 3 respectively. Table 3-6 is based on fuel management studies which were performed to model future equilibrium cycles that include the use of IFBA. These tables provide information on the mechanical fuel design and the fuel features of the assemblies that are loaded fresh for each cycle.

Table 3-3 Palo Verde Unit 1 Fuel Management History

Cycle	Power (MWt)	Fuel Design	Cutback Type	Cutback Enrich.[1]	Cutback Length (in)	Type of BA	Max. # of BA Rods	Max. BA Loading
1	3800	STD	N/A	N/A	N/A	B_4C	16	0.02532 $g^{10}B/in$
2	3800	STD	N/A	N/A	N/A	B_4C	8	0.02000 $g^{10}B/in$
3	3800	STD	N/A	N/A	N/A	B_4C	16	0.02600 $g^{10}B/in$
4	3800	STD	N/A	N/A	N/A	B_4C	16	0.02800 $g^{10}B/in$
5	3800	VAP	N/A	N/A	N/A	B_4C	16	0.02400 $g^{10}B/in$
6	3800	VAP	Solid	Full	7	Er_2O_3	72	2.1 wt. %
7	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
8	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
9	3876	VAP	Solid	Full	7	Er_2O_3	72	2.1 wt. %
10	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
11	3876	VAP	Solid	Full	7	Er_2O_3	80	2.1 wt. %
12	3876	VAP	Solid	Full	7	Er_2O_3	84	2.1 wt. %
13	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
14	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
15	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
15[2]	3990	ANP	Solid	Full	7	Gd_2O_3	12	6 wt. %
16	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
17	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
18	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %

Note:
1. Full means the cutbacks are fully enriched.
2. This identifies the ANP lead use assemblies (LUAs) that have operated at Palo Verde.

Table 3-4 Palo Verde Unit 2 Fuel Management History

Cycle	Power (MWt)	Fuel Design	Cutback Type	Cutback Enrich.[1]	Cutback Length (in)	Type of BA	Max. # of BA Rods	Max. BA Loading
1	3800	STD	N/A	N/A	N/A	B_4C	16	0.02532 $g^{10}B/in$
2	3800	STD	N/A	N/A	N/A	B_4C	16	0.02200 $g^{10}B/in$
3	3800	STD	N/A	N/A	N/A	B_4C	16	0.02000 $g^{10}B/in$
4	3800	STD	N/A	N/A	N/A	B_4C	8	0.02400 $g^{10}B/in$
5	3800	VAP	N/A	N/A	N/A	B_4C	16	0.02400 $g^{10}B/in$
6	3800	VAP	Solid	Full	7	Er_2O_3	40	2.1 wt. %
7	3876	VAP	Solid	Full	7	Er_2O_3	80	2.1 wt. %
8	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
9	3876	VAP	Solid	Full	7	Er_2O_3	72	2.1 wt. %
10	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
11	3876	VAP	Solid	Full	7	Er_2O_3	80	2.1 wt. %
12	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
13	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
14	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
15	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
16	3990	VAP	Solid	Full	7	Er_2O_3	84	2.1 wt. %
17	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
18	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %

Note:
1. Full means the cutbacks are fully enriched.

Table 3-5 Palo Verde Unit 3 Fuel Management History

Cycle	Power (MWt)	Fuel Design	Cutback Type	Cutback Enrich.(1)	Cutback Length (in)	Type of BA	Max. # of BA Rods	Max. BA Loading
1	3800	STD	N/A	N/A	N/A	B_4C	16	0.02515 $g^{10}B/in$
2	3800	STD	N/A	N/A	N/A	B_4C	16	0.02600 $g^{10}B/in$
3	3800	STD	N/A	N/A	N/A	B_4C	16	0.02800 $g^{10}B/in$
4	3800	STD	N/A	N/A	N/A	B_4C	16	0.02000 $g^{10}B/in$
5	3800	VAP	Solid	Full	7	Er_2O_3	72	2.1 wt. %
6	3876	VAP	Solid	Full	7	Er_2O_3	64	2.1 wt. %
7	3876	VAP	Solid	Full	7	Er_2O_3	80	2.1 wt. %
8	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
9	3876	VAP	Solid	Full	7	Er_2O_3	72	2.1 wt. %
10	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
11	3876	VAP	Solid	Full	7	Er_2O_3	88	2.1 wt. %
12	3876	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
13	3876	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
14	3990	VAP	Solid	Full	7	Er_2O_3	84	2.1 wt. %
15	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
16	3990	VAP	Solid	Full	7	Er_2O_3	76	2.1 wt. %
16(2)	3990	NGF	Mixed(3)	Full	7	IFBA	80	2X Thickness(4)
17	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %
18	3990	VAP	Solid	Full	7	Er_2O_3	92	2.1 wt. %

Notes:
1. Full means the cutbacks are fully enriched.
2. This identifies the NGF lead use assemblies (LUAs) that have operated at Palo Verde.
3. The cutback region has a mixture of solid and annular blankets.
4. This equates to a loading of [ ]^{a,c}.

Table 3-6 Palo Verde Operation with IFBA and NGF

Cycle	Power (MWt)	Fuel Design	Cutback Type	Cutback Enrich.(1)	Cutback Length (in)	Type of BA	Max. # of BA Rods	Max. BA Loading
CEQ	4058	NGF	Annular	Full	6-8	IFBA	236	2X Thickness(2)

Notes:
1. Full refers to fully enriched.
2. This equates to a loading of [ ]^{a,c}.

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

Table 3-7 Fuel Design Parameters

Parameter	Fuel Design			
Fuel type	1	2	3	4
Fuel design	STD	VAP	ANP	NGF
Cutback type	N/A	Solid	Solid	Annular
Pin Pitch, in	0.506	0.506	0.506	0.506
Pellet OD, in	0.3250	0.3255	0.3255	0.3225
Clad OD, in	0.382	0.382	0.382	0.374
Clad ID, in	0.332	0.332	0.332	0.329
Center IT OD, in	0.980	0.980	0.980	0.980
Corner GT OD, in	0.980	0.980	1.023	0.980
Cutback enrichment	N/A	Full	Full	Full
BA type	B_4C	Er_2O_3	Gd_2O_3	IFBA
BA loading	16-Fingers	92 rods	12 rods	236 rods
BA loading	0.028 $g^{10}B/in$	2.1 wt. %	6 wt. %	[ ]^{a,c}

[

]^{a,c}

3.2 FUEL STORAGE DESCRIPTION

Palo Verde has three SFPs. The physical characteristics of these SFPs are described in this section. Each SFP contains a single rack design. Each SFP is surrounded by a concrete wall with a stainless steel liner. The three SFPs are identical in layout, and a representative diagram of the Unit 1, 2, and 3 SFPs is provided, see Figure 3-1. The presence of neither the SFP concrete wall nor liner is credited in this analysis. All storage arrays are conservatively assumed to be radially infinite.

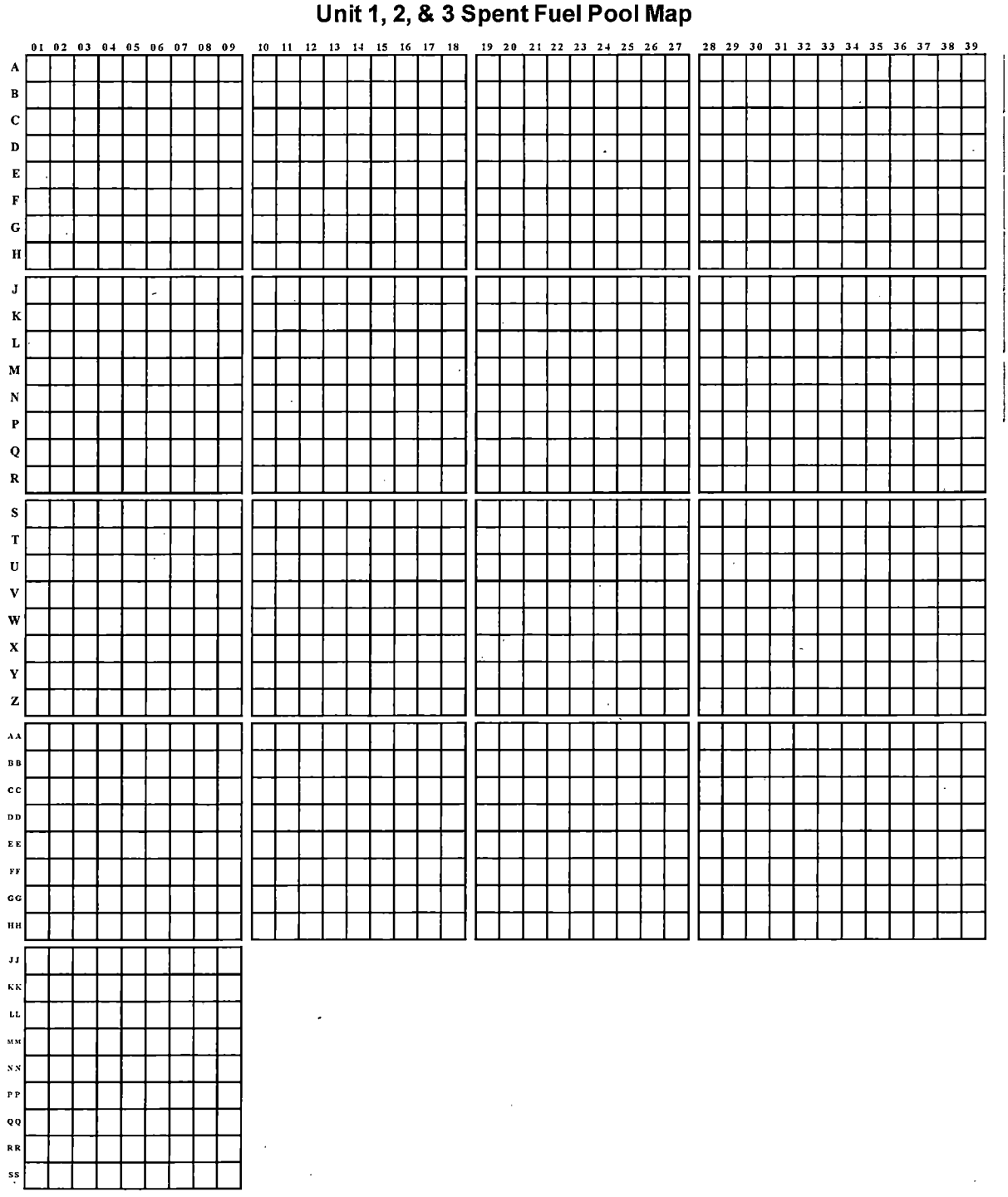


Figure 3-1. Representative Palo Verde SFP (Units 1, 2, and 3)

3.2.1 Spent Fuel Pool Description

Each SFP contains storage racks designed to maximize the number of storage cells available (minimize storage cell pitch). The racks were designed to be used in a checkerboard pattern of fresh and empty storage locations. The locations which were to contain fresh fuel had guiding inserts (L-inserts) made of stainless steel installed in them to help center the fuel assemblies within this space. The racks contain no fixed neutron absorber for reactivity suppression. The specifications for the racks are given in Table 3-8.

Table 3-8 Fuel Storage Rack Specifications

Parameter	Value	Tolerance
Rack cell pitch, in	9.515	+0.185/-0.005
Cell inner dimension, in	9.395	+0.18/-0.0
Cell wall thickness, in	0.12	± 0.005
L-Insert Length, in	172.25	± 0.063
L-Insert Thickness, in	0.175	± 0.005
L-Insert Material	304 Stainless Steel	N/A
L-Insert Offset from Cell Wall, in	0.6875	+0.060/-0.0
Min distance to SFP wall, in	11.0	± 1.125

3.2.2 Trashcan Dimensions and Tolerances

Palo Verde uses trashcans to store varius non-fissile material in the SFP. Examples of materials typically stored include discarded CEAs and In-core Instrumentation tubes, filters, and reconstitution materials. The specifications for the trashcans are summarized in Table 3-9.

Table 3-9 Trashcan Specifications

Parameter	Value	Tolerance
Trashcan ID, in	7.875	± 0.015625
Trashcan wall thickness , in	0.125	± 0.015625
Trashcan material	304 Stainless Steel	N/A
Trashcan contents	Non-fissile materials	N/A

3.2.3 NETCO SNAP-IN

This analysis assumes that NETCO SNAP-IN Neutron Absorber (SNAP-IN) inserts will be used in various configurations to suppress reactivity. The SNAP-IN is a chevron shaped rack insert covering the entire active length of the fuel which uses friction and compression forces to hold it within the fuel storage rack once installed. The SNAP-IN consists of an aluminum and B_4C core material clad in aluminum. The inserts provide neutron absorption capability to fuel storage racks with minimal impact on fuel movement operations. Once installed, SNAP-IN inserts remain in place, becoming an integral part of the spent fuel storage rack. The minimum specifications that the SNAP-INs must meet are listed in Table 3-10.

Table 3-10 SNAP-IN Insert Specifications

Parameter	Value	Tolerance
Neutron Absorber Material	Al-B_4C	N/A
Neutron Absorber Areal Density ($g^{10}B/cm^2$)	0.0156	Minimum
SNAP-IN Insert Length, in	> 150	N/A
SNAP-IN Total Thickness, in	0.125	[ ]^{a,c}
SNAP-IN Insert 'Wing-to-Wall' Gap, in	≤ 0.10	Maximum

4 DEPLETION ANALYSIS

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

]^{a,c}

4.1 DEPLETION MODELING SIMPLIFICATIONS & ASSUMPTIONS

[

]^{a,c}

4.2 FUEL DEPLETION PARAMETER SELECTION

4.2.1 Fuel Isotopic Generation

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

[

]^{a,c}

Based on Palo Verde fuel management, the fuel has isotopic number densities which are calculated at enrichments of 3.0, 4.0, and 5.0 wt% ^{235}U and decay times of 0, 5, 10, 15, and 20 years. [

]^{a,c}

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 4) provides discussion on the core operation parameters important to SFP criticality. This section outlines the parameters used in generating the fuel isotopics and why they are appropriate for use in this analysis. The operating conditions of the fuel are provided in Table 4-7. The table provides both the nominal values and the values assumed in the analysis.

4.2.2.1 Soluble Boron Concentration

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

Based on guidance from Reference 4, a constant cycle average soluble boron concentration (Equation 4-1) which assumes 19.9 at% ^{10}B may be modeled in place of a soluble boron letdown curve. To determine the maximum cycle average soluble boron concentration, fuel management strategies for Palo Verde have been reviewed. Table 4-1 provides the cycle average soluble boron concentration for Cycles 1 through 17 of operation for Units 1 and 3 and for Cycles 1 through 18 of operation for Unit 2.

[ ]^{a,c} Equation 4-1

[

]^{a,c}

Table 4-1 Cycle Average Soluble Boron Concentrations

Unit 1	Soluble Boron (ppm)	Unit 2	Soluble Boron (ppm)	Unit 3	Soluble Boron (ppm)
Cycle 1	410	Cycle 1	546	Cycle 1	571
Cycle 2	742	Cycle 2	879	Cycle 2	843
Cycle 3	796	Cycle 3	662	Cycle 3	647
Cycle 4	589	Cycle 4	628	Cycle 4	612
Cycle 5	731	Cycle 5	757	Cycle 5	732
Cycle 6	673	Cycle 6	559	Cycle 6	715
Cycle 7	825	Cycle 7	700	Cycle 7	871
Cycle 8	911	Cycle 8	875	Cycle 8	854
Cycle 9	832	Cycle 9	821	Cycle 9	821
Cycle 10	822	Cycle 10	695	Cycle 10	821
Cycle 11	820	Cycle 11	883	Cycle 11	814
Cycle 12	866	Cycle 12	761	Cycle 12	927
Cycle 13	**1143**	Cycle 13	936	Cycle 13	938
Cycle 14	959	Cycle 14	838	Cycle 14	875
Cycle 15	791	Cycle 15	856	Cycle 15	772
Cycle 16	767	Cycle 16	785	Cycle 16	746
Cycle 17	832	Cycle 17	809	Cycle 17	831
Cycle 18	N/A	Cycle 18	788	Cycle 18	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 fuel. Therefore, utilizing a higher fuel temperature is more conservative.

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

]^{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 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 4). However, assuming a bounding assembly average power (therefore high specific power) ensures high fuel temperature which is conservative throughout life. Interim Staff Guidance (ISG) DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools" (Reference 6) 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 4, the fuel temperature impact on reactivity is greater than the impact from specific power. This makes the selection of a high specific power, to maximize fuel temperature, appropriate. [

]^{a,c}

4.2.2.4 Maximum Average Assembly Power

[

]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 to calculate the burnup limits provided in Section 6.1.

As discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses" (Reference 7), as fuel is operated in the reactor, the center of each assembly generates more power than the ends. This leads to the burnup of each assembly varying along its length. Because the middle of each assembly generates most of the power, the burnup in the middle of the assembly is greater than the assembly average. At the same time, the ends of the assembly are less burned than the assembly average. When the burnup difference between the middle and end of an assembly is large enough, reactivity becomes driven by the end of the assembly rather than the middle, as the under depletion of the ends overcomes the reactivity loss due to the neutron leakage.

Without considering neutron leakage, a uniformly enriched assembly's reactivity will be driven by the lowest burnup section of the fuel. At the beginning of life, there is no axial burnup variation (fresh fuel) and thus the fuel is axially isoreactive unless neutron leakage is considered. When neutron leakage is considered, the axial location which has the least leakage will be most reactive, thus the center of the assembly drives reactivity. As the burnup of the assembly becomes more distributed, the impact of

neutron leakage decreases and the assembly reactivity is increasingly driven by the under-depleted ends of the fuel. Therefore, as a general rule, with increasing burnup the axial location driving assembly reactivity moves from the center towards the end of an assembly.

[

]^{a,c}

4.2.3.2 Axial Moderator Temperature Profile Selection

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

Selecting an appropriate moderator temperature profile is important as it impacts the moderator density and therefore the neutron spectrum during depletion as discussed in Reference 4. 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.

[

]^{a,c}

4.2.4 Burnable Absorber Usage

Burnable absorber usage at Palo Verde 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 used at Palo Verde include both discrete and integral BAs, and the history of BA usage is available in Table 3-3 through Table 3-6.

Palo Verde has previously used Solid Al_2O_3-B_4C rods in place of fuel rods. The discrete BA impacts assembly reactivity after discharge. These inserts contain boron, which is a strong thermal neutron absorber, hardening the neutron spectrum of the assembly where the BA rods reside. This phenomenon causes increased plutonium production in the assembly. However, these BA rods displace fissile material because they are used in place of the fuel rods of an assembly. The negative reactivity due to replacing fuel rods with BA rods offsets the spectral hardening due to the poison, as demonstrated in "Study of the Effect of Integral Integral Burnable Absorbers on PWR Burnup Credit" (Reference 8). The BA rod parameters are shown in Table 4-2.

Table 4-2 Discrete Burnable Absorber Specifications

Parameter	Value
BA material	Al_2O_3-B_4C
Maximum No. BA rods	16
BA Loading ($g^{10}B$/in)	0.028

In addition to the discrete absorber used at Palo Verde, the integral absorbers Erbia, IFBA, and Gadolinia have been used. As with the discrete absorber, the integral absorbers contain neutron poisons which cause increased plutonium production in the assembly, leading to the assembly being more reactive when discharged to the SFP. The integral burnable absorber parameters are shown in Table 4-3.

Table 4-3 Integral Burnable Absorber Specifications			
Parameter	**IFBA**	**Erbia**	**Gadolinia**
BA material	ZrB_2	Er_2O_3	Gd_2O_3
BA loading	[ ]^{a,c}	2.1 wt.%	6.0 wt.%
Max. No. rods	236	92	12
Max. BA length, in	138	136	136

4.2.5 Radial Enrichment Zoning Study

One of the characteristics of Combustion Engineering reactors is the large GTs and IT used in the fuel assembly design. Because of the increased moderation the fuel rods surrounding these large GTs and IT are exposed to, they operate at higher power than other fuel rods in the lattice. To control fuel rod power peaking, the Palo Verde fuel management strategy uses radial enrichment zoning. Individual assemblies may contain two or three different fuel rod enrichments which are used to control peaking factors. To determine the reactivity impact of operating with radial enrichment zoning instead of uniform radial zoning, the following study was performed.

[

]^{a,c}

Table 4-4 Core Operating Parameters for Depletion Analyses a,c

[

]a,c

a,c

Figure 4-1. Radial Enrichment Study Results

[

]a,c

4.3 FUEL DESIGN SELECTION

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

[

]a,c

Table 4-5 Fuel Design Parameters		
Parameter	**Value**	
Fuel Type	2	4
Fuel Design	VAP	NGF
Fuel Density, %	96.5	98.0
Cutback Type	Solid	Annular
Cutback Enrichment	Full	Full
BA Type	None	IFBA
BA Loading	None	236 rods
Note: 1. The values given are bounding, rather than nominal, values to conservatively account for each feature.		

[

]^{a,c}

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

Based on the results from Table 4-6, it is clear that the NGF design is limiting throughout life. Therefore, the NGF design will be used to develop the isotopics used in the spent fuel reactivity calculations.

4.4 FINAL DEPLETION PARAMETERS

This section outlines the parameters used in the final depletion calculations. The depletion parameters outlined 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 4-7 through Table 4-9.

Table 4-7 Parameters Used in Depletion Analysis

Parameter	Nominal Values	Depletion Analysis
Maximum cycle average soluble boron concentration, ppm	1143	[ ]^{a,c}
Rated thermal power, MWt	$\leq$ 3990	[ ]^{a,c}
Average assembly power, MWt	16.56	[ ]^{a,c}
Core outlet moderator temperature, °F	627.4	[ ]^{a,c}
Core inlet moderator temperature, °F	566.0	[ ]^{a,c}
Minimum RCS flow rate (Thermal Design Flow), gpm/coolant pump	105825	[ ]^{a,c}
Fuel designs	STD, VAP, ANP, NGF	NGF
Fuel assembly cutback	Fully Enriched Solid, Fully Enriched Annular	[ ]^{a,c}
Fuel density, % Theoretical Density	95.5	[ ]^{a,c}
BA	Solid B_4C; Erbia, Gadolinia, IFBA	[ ]^{a,c}
BA Lengths, in	136, 136, 136, 138	[ ]^{a,c}

Table 4-8 Fuel Assembly Mechanical Specifications	
Parameter	**Value**
Assembly type	NGF
Rod array size	16x16
Rod pitch, in	0.506
Active fuel length, in	150
Total number of fuel rods	236
Fuel cladding OD, in	0.374
Fuel cladding ID, in	0.329
Fuel cladding thickness, in	0.0225
Pellet diameter, in	0.3225
Number of GT/IT	4/1
Guide/instrument tube OD, in	0.980/0.980
Guide/instrument tube thickness, in	0.040/0.040

Table 4-9 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 at Palo Verde. The section also confirms continued safe SFP operation during both normal and accident conditions.

5.1 KENO MODELING 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 manufacturing tolerances, fuel depletion, eccentric positioning, and the allowable temperature range of the SFP. 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. [

]^{a,c}

[

]^{a,c}

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 and the fuel storage racks. These dimensions and tolerances are the basis for the KENO models used to determine the burnup requirements for each fuel storage array and to confirm the safe operation of the SFP under normal and accident conditions. This section also documents the mechanical design of the trashcans contained in Array B.

5.1.1.1 Fuel Assembly Dimensions and Tolerances

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

Table 5-1 Design Basis Fuel Assembly Design Specifications		
Parameter	**Value**	**Value (Analyzed)**
Assembly type	16x16 NGF	16x16 NGF
Rod array size	16x16	16x16
Rod pitch, in	0.506 [ ]^{a,c}	0.506 [ ]^{a,c}
Active fuel length, in	150	150
Fuel theoretical density, % TD	[ ]^{a,c}	[ ]^{a,c}
Maximum pellet enrichment, wt% ^{235}U	4.95	5.0
Maximum radial average enrichment, wt% ^{235}U	4.65	[ ]^{a,c}
Enrichment tolerance, wt% ^{235}U	[ ]^{a,c}	[ ]^{a,c}
Total number of fuel rods	236	236
Fuel cladding OD, in	0.374 [ ]^{a,c}	0.374 [ ]^{a,c}
Fuel cladding thickness, in	0.0225 [ ]^{a,c}	0.0225 [ ]^{a,c}
Pellet diameter, in	0.3225 [ ]^{a,c}	0.3225 [ ]^{a,c}
Number of GT/IT	4/1	4/1
GT/IT OD, in	0.980 [ ]^{a,c}	0.980 [ ]^{a,c}
GT/IT ID, in	0.900 [ ]^{a,c}	0.900 [ ]^{a,c}

5.1.1.2 Fuel Storage Cell Rack Dimensions and Tolerances

The storage racks used at Palo Verde are described in Section 3.2. The fuel storage cell characteristics, as they are modeled in the criticality analysis, are summarized in this section.

Table 5-2 Fuel Storage Rack Specifications		
Parameter	**Value**	**Value (Analyzed)**
Rack cell pitch, in	9.515 +0.185/-0.005	9.515 +0.185/-0.005
Cell inner dimension, in	9.395 + 0.18/-0.0	9.395 + 0.18/-0.0
Cell wall thickness, in	0.120 ± 0.005	0.120 ± 0.005
L-insert Height, in	172.25 ± 0.063	172.25 ± 0.063
L-insert Material	Stainless Steel 304	Stainless Steel 304
L-insert Offset from Cell Wall, in	0.6875 + 0.06 / -0.0	0.6875 + 0.06/-0.0
L-insert Thickness, in	0.175 ± 0.005	0.175 ± 0.005

5.1.1.3 Trashcan Dimensions and Tolerances

The trashcans used at Palo Verde are described in Section 3.2. The trashcan characteristics as they are modeled in the criticality analysis are summarized in Table 5-3.

Table 5-3 Trashcan Specifications

Parameter	Value	Value (Analyzed)
Trashcan ID, in	7.875 ± 0.015625	7.875 ± 0.015625
Trashcan wall thickness, in	0.125 ± 0.015625	0.125 ± 0.015625
Trashcan Material	Stainless Steel 304	Stainless Steel 304
Trashcan contents	See Note 1	[ ]^{a,c}

Note:
1. Only non-fissile materials are allowed to be stored in the trashcan.

5.1.1.4 NETCO-SNAP-IN® Neutron Absorber Insert Dimensions and Tolerances

The minimum requirements for the neutron absorber inserts used at Palo Verde are described in Section 3.2. The insert characteristics as they are modeled in the criticality analysis are summarized in Table 5-4.

Table 5-4 SNAP-IN Insert Specifications

Parameter	Minimum Requirements	Value (Analyzed)
Neutron Absorber Areal Density ($g^{10}B/cm^2$)	> 0.0156	0.0156
SNAP-IN insert Length, in(1)	> 150	150
SNAP-IN Total Thickness, in (5052 Aluminum Clad)	0.120	0.125
SNAP-IN Insert 'Wing-to-Wall' Gap, in	≤ 0.10	[ ]^{a,c}

Notes:
1. The inserts cover the entire active fuel region of the assembly.

5.1.2 The Impact of Structural Materials on Reactivity

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

Each of the fuel types which have been, or are planned to be, operated at the plant need to be considered. For Palo Verde, the determination of the bounding fuel assembly design for the analysis has been performed as outlined in Section 4.3. [

]^{a,c}

5.1.2.1 Composition of Structural Materials

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

]^{a,c}

5.1.2.2 Top and Bottom Nozzles

[

]^{a,c}

5.1.2.3 Grids and Sleeves

Westinghouse has performed a study to provide justification for neglecting grids and sleeves in criticality safety analyses. The study specifically included CE 16x16 fuel. The study determined the impact of grids and sleeves being present in an assembly both during core operation and storage in the SFP. The study was based on the [

]^{a,c} The study incorporated a variety of depletion parameters and several different [

]^{a,c}

5.2 BURNUP LIMIT GENERATION

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

Each assembly in the reactor core depletes under slightly different conditions and therefore can have a different reactivity at the same burnup. This is accounted for in the analysis by using a combination of depletion parameters that together produce a bounding isotopic inventory throughout life (see Section 4.4). Additionally, while fuel manufacturing is a very tightly controlled process, assemblies are not identical. Reactivity margin is added to the KENO reactivity calculations for the generation of burnup limits as discussed in Section 5.2.2 to account for manufacturing deviations.

5.2.1 Array Descriptions

Assembly storage is controlled through the storage arrays defined in this section. An array can only be populated by assemblies of the fuel region defined in the array definition or a lower reactivity fuel region. Fuel regions are defined by assembly burnup, initial enrichment, and decay time as provided by Table 6-2 through Table 6-9. Storage Arrays with empty cell locations have been additionally evaluated with cell blockers of an annular pip design which span the entire active fuel length. The desired cell blocker design is nominally a 3.50 cm ± 0.05 cm annular stainless steel pipe design with 0.300 ± 0.05 inch thick walls placed in the cell center. Analysis determine for cell blockers bounding the desired cell blocker design, up to 3.55 inches in outer diameter and an annulus cross sectional area of 2.105 in^2, there is no reactivity impact. The analysis considered cell centered and off center placement and all cases show no positive reactivity impact.

Table 5-5 provides an overview of the fuel region reactivity ranking.

Table 5-5 Fuel Regions Ranked by Reactivity

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

Notes:
1. Fuel Regions are defined by assembly average burnup, initial enrichment and decay time as provided by Tables 6-3, 6-5, 6-7 and 6-9.
2. Fuel Regions are ranked in order of decreasing reactivity, e.g., Fuel Region 2 is less reactive than Fuel Region 1, etc.
3. Fuel Region 1 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U. No burnup is required.
4. Fuel Region 2 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U with at least 16.0 GWd/MTU of burnup.
5. Fuel Regions 3 through 6 are determined from the minimum burnup equation and coefficients provided in Tables 6-2, 6-4, 6-6 and 6-8.
6. Assembly storage is controlled through the storage arrays defined in Figure 5-1.
7. Each storage cell in an array can only be populated with assemblies of the Fuel Region defined in the array definition or a lower reactivity Fuel Region.

Descriptions of the fuel storage arrays allowable for use at Palo Verde follow.

Array A Two Region 1 assemblies (1) checkerboarded with two empty cells (X). The Region 1 assemblies are in the cells with the L-inserts. No SNAP-IN inserts are credited.	**1**	**X**
	X	**1**
Array B Two Region 1 assemblies (1) checkerboarded with two cells containing trashcans (TC). The Region 1 assemblies are in the cells with the L-inserts, every cell without L-inserts must contain SNAP-INs.	**1**	**TC**
	TC	**1**
Array C Two Region 2 assemblies (2) checkerboarded with one Region 3 assembly (3) and one cell containing only water (X). The Region 2 assemblies are in the cells with the L-inserts. The Region 3 assembly is in a cell containing a SNAP-IN insert.	**2**	**X**
	3	**2**
Array D One Region 2 assembly (2) checkerboarded with three Region 4 assemblies (4). The Region 2 assembly and diagonally located Region 4 assembly are in the storage cells with the L-inserts. The two storage cells without L-inserts contain SNAP-IN inserts.	**2**	**4**
	4	**4**
Array E Four Region 5 assemblies (5). Two of the Region 5 assemblies are in the storage cells with the L-inserts and one Region 5 assembly is in a storage cell containing a SNAP-IN insert.	**5**	**5**
	5	**5**
Array F Four Region 6 assemblies (6). Two of the Region 6 assemblies are in the storage cells with the L-inserts. No SNAP-IN inserts are credited.	**6**	**6**
	6	**6**

Notes:

1. The shaded locations indicate cells which contain an L-insert.
2. An empty cell (X) contains only water in the active fuel region.
3. NETCO-SNAP-IN® inserts must be oriented in the same direction as L-inserts.
4. NETCO-SNAP-IN® inserts are only located in cells without L-inserts.
5. Any cell containing fuel or a trash can may instead be an empty (water-filled) cell in all storage arrays.
6. Any storage array location designated for a fuel assembly may be replaced with non-fissile material.

Figure 5-1. Allowable Storage Arrays

5.2.2 Target k_{eff} Calculation Description

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

]^{a,c}

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 and fission product uncertainties), or measurement related (burnup measurement uncertainty).

5.2.3.1 Bias & Uncertainty Descriptions

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

5.2.3.1.1 Manufacturing Tolerances

The reactivity effect of manufacturing tolerances is included in the criticality analysis. KENO is used to quantify these effects. [

]^{a,c}

[

]^{a,c}

a,c

Figure 5-2. [ ]a,c

a,c

Figure 5-3. [ ]a,c

[

]a,c

5.2.3.1.2 Burnup Measurement Uncertainty

[

]^{a,c}

5.2.3.1.3 Depletion Uncertainty

[

]^{a,c}

5.2.3.1.4 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. [

]^{a,c}

[

]a,c

5.2.3.1.5 Operational Uncertainty

The operational uncertainty of 0.002 Δk included in the rackup tables is an additional conservatism which is added to the conservatism inherent in the specific power histories from reactor operation as discussed in Section 4.2.2.3 and Reference 4.

5.2.3.1.6 Eccentric Fuel Assembly Positioning

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. [

]a,c

5.2.3.1.7 Other Uncertainties

An uncertainty in the predictive capability of SCALE 6.1.2 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.

5.2.3.1.8 SFP Temperature Bias

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

]^{a,c}

5.2.3.1.9 Volatile Isotopes Bias

Fission gases released during operation may not remain in the fuel pellet. As a result fission gas migrates to the gap region between the fuel and the cladding, and up into the plenum. As a result, a reduction in fission gases near the pellet occurs. This value is known to be less than 5% under normal operation as seen in Reference 15 and its supporting references. Reference 16 indicates that for cladding breaches under transportation scenarios, which is an accident scenario, that bounding values of up to 30% of fission gases and 0.02% of alkalis are released from the fuel rod. Reference 17, however, provides significantly more modern data for many of the isotopes for a more severe accident scenario of fuel failure during operation. [

]^{a,c}

5.2.3.1.10 Grid Growth Bias

[

]^{a,c}

[

]^{a,c} A fit of burnup versus reactivity impact was generated to conservatively quantify the impact of grid growth on the spent fuel pool criticality safety analysis. The results of the calculations were used to develop Equation 5-7 which provides a conservative quantification of the reactivity impact as a function of burnup in GWd/MTU for all storage arrays containing burnup. The reactivity impact due to grid growth is incorporated into the burnup coefficients documented in Section 6 for proposed use at Palo Verde Nuclear Generating Station.

[ ]^{a,c} Equation 5-7

5.2.3.1.11 Borated and Unborated Biases and Uncertainties

Palo Verde Technical Specifications require the SFP k_{eff} to be ≤ 0.95 under borated conditions accounting for all applicable biases and uncertainties. [

]^{a,c}

[

]^{a,c}

5.2.3.2 Storage Array Biases & Uncertainties Results

[

]^{a,c}

[

]^{a,c}

Table 5-6 Biases & Uncertainties for Array A a,c

Table 5-7 Biases & Uncertainties for Array B a,c

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

Table 5-9 Biases & Uncertainties for Array D

a,c

Table 5-10 Biases & Uncertainties for Array E a,c

Table 5-11 Biases & Uncertainties for Array F 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. At Palo Verde, every SFP has only a single storage rack design. Therefore the only interfaces that exist are those between arrays within the single storage rack design.

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

]^{a,c}

5.4 NORMAL CONDITIONS

This section discusses normal conditions within the SFP in addition to the steady-state storage of fresh and spent assemblies. During normal operation, the SFP has a soluble boron concentration of greater than 2150 ppm and a moderator temperature $\leq$ 180°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 placement of components in or near the intact fuel assemblies while normally stored in the storage racks. This also includes removal and reinsertion of these components into the fuel when stored in the rack positions using specifically designed tooling. Examples of these evolutions include: CEAs and GT inserts such as in-core instrumentation tubes.

The Type 1 normal conditions include insertion of components into fuel assemblies for storage in the SFP. The SFP as a single system is over moderated. A single fuel assembly however, is significantly under-moderated, and reducing the interstitial H/U ratio has a negative impact on the system k_{eff}. Calculations have been performed for Palo Verde which show that [

]^{a,c} While Palo Verde will not store fuel pins in guide tubes or instrument tubes, storage of depleted neutron sources in guide tubes is allowed since [

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

5.4.2 Type 2 Normal Conditions

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

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

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

Fuel assembly reconstitution is defined as either pulling damaged fuel rods out of an assembly and reinserting intact rods with less reactivity than the damaged rod or removing undamaged rods from a damaged assembly for insertion in a new assembly. In most cases, damaged rods will be replaced with stainless steel rods. Natural uranium rods may also be used. [

]^{a,c}

[[

]a,c

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

Fuel assembly cleaning, inspection, reconstitution, and sipping are bounded by this criticality analysis.
[

]a,c

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. Examples include failed fuel rod baskets, movable in-core detectors, and miscellaneous maintenance equipment. Any components that do not contain fissile materials can replace a fuel assembly of any fuel region in one of the approved storage configurations described in Section 5.2.1.

The fuel rod storage basket (FRSB) contains an array of fuel rods, used to store fuel rods and pellets which have been removed from a fuel assembly. Figure 5-4 is a drawing of the FRSB.

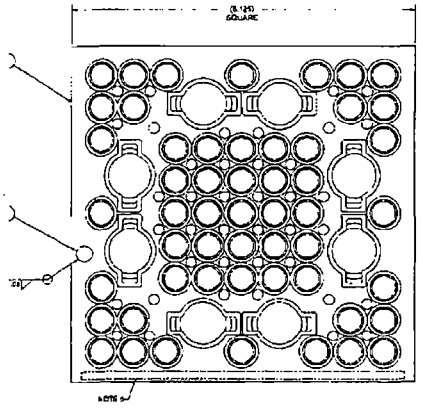


Figure 5-4. Fuel Rod Storage Basket

[

]a,c

5.4.4 Type 4 Normal Conditions

Type 4 normal conditions include temporary installation of non-fissile components on the rack periphery. Analyses of the storage arrays contained within this criticality analysis assume an infinite array of storage cells. This assumption bounds the installation of 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 FRSB, etc.) or credited as an empty cell.

Insertion of handling tools into the top of fuel assemblies or other components occurs frequently in the SFP environment. The insertion of handling tools into the top of an assembly is bounded by the storage of

inserts in fuel assemblies and therefore, from a criticality perspective, all fuel handling tools are acceptable for their intended purpose.

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.

5.5 SOLUBLE BORON CREDIT

Soluble boron credit for Normal Operating Conditions is evaluated for PVNGS. Each storage array is considered with [

]^{a,c} Additional pertinent details for modeling each storage array conservatively for the Normal Operating Conditions soluble boron determination are as follows:

- [

]^{a,c}

The minimum soluble boron concentration in the Palo Verde SFP to maintain $k_{eff} \leq 0.95$ for the limiting normal condition including biases, uncertainties, and administrative margin is 550 ppm. [

]^{a,c}

Table 5-12 [ ]^{a,c} a,c

550 ppm of soluble boron [

]^{a,c}

5.6 ACCIDENT DESCRIPTION

The following reactivity increasing accidents are considered in this analysis:

- Assembly misload
- SFP temperature greater than normal operating range (180°F)
- Dropped & misplaced fresh fuel assembly
- Seismic event
- Inadvertent removal of a SNAP-IN insert

5.6.1 Assembly Misload into the Storage Racks

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. This analysis addresses both multiple assemblies being misloaded in series into unacceptable storage locations and the misload of a single assembly into an unacceptable storage location.

5.6.1.1 Single Assembly Misload

[

]^{a,c} This accident requires 1200 ppm of boron (19.7 at% ^{10}B) to maintain $k_{eff} \leq 0.95$.

5.6.1.2 Multiple Assembly Misload

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

]^{a,c}

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

[

]

5.6.2 Spent Fuel Temperature Outside Operating Range

The SFP is to be operated at less than 180°F. However, under accident conditions this temperature could be higher. [

]^{a,c}

5.6.3 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. [

]^{a,c}

[

]a,c

5.6.4 Seismic Event

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

]a,c

5.6.5 Inadvertent Removal of a SNAP-IN Insert

With the incorporation of SNAP-IN inserts, a new potential accident event is created. When removing an assembly from a storage cell, it becomes possible for the SNAP-IN insert to be dragged out of the cell together with the assembly. The removal of the insert will cause a reactivity increase due to the loss of neutron absorbing material from the storage array.

[

]a,c

5.6.6 Accident Results

Table 5-14 [ ]a,c a,c

Inadvertent removal of a SNAP-IN insert from Array C (450 ppm)	0.93605
Note: 1. The k_{eff}'s provided do not include administrative margin.	

The results in Table 5-14 demonstrate that the Palo Verde SFPs will not exceed a k_{eff} of 0.95 at a 95 percent probability with 95 percent confidence. While the administrative margin is not included in the k_{eff}'s provided, the maximum credited soluble boron is [ ]^{a,c} from the minimum soluble boron technical specification limit.

5.7 RODDED OPERATION

Palo Verde, like the vast majority of nuclear power plants in the U.S., operates at hot full power conditions almost exclusively. While standard operation is performed unrodded, it is allowable to operate at hot full power with rods inserted to the power dependent insertion limits. Operating with CEAs inserted into the core impacts the assemblies in the rodded locations. The insertion of a CEA 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 due to the loss of moderator as water is displaced in the GTs when a CEA is inserted into the assembly; this will harden the neutron spectrum leading to increased plutonium production. The CEA will also preferentially absorb thermal neutrons, further hardening the neutron spectrum. In addition to the spectral hardening, the CEA 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 rod, 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 which experienced unrodded operation, with the same assembly average burnup.

While these items can increase reactivity, there are competing effects which reduce assembly reactivity due to rod insertion. When a CEA 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 CEA inserted in an assembly will shift power down, under depleting the top of the assembly while the CEA is present. Once the CEA 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 14) defines a significant amount of control rod insertion as more than 20 cm into the core.

Palo Verde has not operated at full power with control rods inserted a significant length, nor are there plans to begin operating in such a manner. Therefore, there is no significant burnup accrued during depletion with CEAs inserted in the active fuel height, and no need to account for these effects in burnup limits contained within this analysis.

While standard operation for Palo Verde is performed unrodded, there is potential to operate at reduced power levels for a short period of time with rods inserted to the power dependent insertion limits. Short term reduced power operation may be the result of plant equipment issues or economic considerations. Parameters will be checked with regards for Spent Fuel Pool Criticality Safety at PVNGS to make certain that PVNGS operates within the Criticality Analysis Area of Applicability for such operations.

6 ANALYSIS RESULTS & CONCLUSION

This section documents the results of the Palo Verde criticality safety analysis. Included in this section are the burnup requirements for the fuel storage arrays documented in this analysis. This section also contains the restrictions of this analysis. The Area of Applicability (AoA) of the validation suite is discussed in Appendix A.

6.1 BURNUP LIMITS 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 region defined in the array definition or a lower reactivity fuel region (see Table 6-1). Fuel regions are defined by assembly average burnup, initial enrichment[1], and decay time as provided by Table 6-2 through Table 6-9.

Table 6-1 Fuel Regions Ranked by Reactivity

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

Notes:
1. Fuel Regions are defined by assembly average burnup, initial enrichment and decay time as provided by Tables 6-3, 6-5, 6-7 and 6-9.
2. Fuel Regions are ranked in order of decreasing reactivity, e.g., Fuel Region 2 is less reactive than Fuel Region 1, etc.
3. Fuel Region 1 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U. No burnup is required.
4. Fuel Region 2 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U with at least 16.0 GWd/MTU of burnup.
5. Fuel Regions 3 through 6 are determined from the minimum burnup equation and coefficients provided in Tables 6-2, 6-4, 6-6 and 6-8.
6. Assembly storage is controlled through the storage arrays defined in Figure 5-1.
7. Each storage cell in an array can only be populated with assemblies of the Fuel Region defined in the array definition or a lower reactivity Fuel Region.

This analysis has provided burnup requirements at discrete decay times. However, it is acceptable to interpolate between these decay times to determine burnup limits at alternate decay times. Using linear

1. Initial Enrichment is the maximum radial average ^{235}U enrichment of the central zone region of fuel, excluding axial cutbacks, 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.

interpolation between two already analyzed decay times will give a conservative burnup requirement for the decay time in question. This is acceptable because isotopic decay is an exponential function which means assembly reactivity will decay faster than the calculations using linear interpolation would predict.

Table 6-2 Fuel Region 3: Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	-0.8100	6.5551	-2.9050	-21.0499
5	-0.9373	7.6381	-6.0246	-18.0299
10	-0.8706	6.8181	-3.1913	-21.0299
15	-0.7646	5.6311	0.7657	-25.1599
20	-0.7233	5.1651	2.3084	-26.7499

Notes:

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

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

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment between 2.50 wt% ^{235}U and 5.00 wt% ^{235}U may be used. Below 2.50 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20 years limits.

Table 6-3 Fuel Region 3: Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	2.50	**3.00**	**4.00**	**5.00**
0	0.00	7.36	20.37	27.05
5	0.00	7.33	20.09	25.63
10	0.00	7.25	19.57	24.63
15	0.00	7.17	19.06	23.86
20	0.00	7.13	18.83	23.50

Note:

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

Table 6-4 Fuel Region 4: Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.0333	-2.1141	27.4985	-41.8258
5	-0.2105	0.2472	19.7919	-34.2641
10	0.0542	-2.5298	28.0953	-41.7092
15	0.3010	-5.0718	35.6966	-48.5494
20	0.4829	-6.9436	41.3118	-53.6182

Notes:

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

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

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment between 1.75 wt% ^{235}U and 5.00 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 limits.

Table 6-5 Fuel Region 4: Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.75	**3.00**	**4.00**	**5.00**
0	0.00	22.54	36.47	46.97
5	0.00	21.65	35.38	44.55
10	0.00	21.27	33.66	42.29
15	0.00	21.02	32.35	40.76
20	0.00	20.86	31.43	39.70

Note:

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

Table 6-6 Fuel Region 5: Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.1586	-3.0177	28.7074	-39.8636
5	-0.2756	1.3433	14.5578	-26.4388
10	-0.2897	1.3218	14.6176	-26.4160
15	-0.0736	-0.9107	21.2118	-32.1887
20	0.1078	-2.7684	26.6911	-36.9873

Notes:

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

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

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment between 1.65 wt% ^{235}U and 5.00 wt% ^{235}U may be used. Below 1.65 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year limits.

Table 6-7 Fuel Region 5: Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.65	**3.00**	**4.00**	**5.00**
0	0.00	23.38	36.83	48.05
5	0.00	21.88	35.64	45.47
10	0.00	21.51	34.66	43.50
15	0.00	21.26	33.37	41.89
20	0.00	21.08	32.38	40.73

Note:

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

Table 6-8 Fuel Region 6: Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.4890	-6.7447	42.7619	-49.3143
5	0.5360	-6.9115	41.1003	-46.6977
10	0.4779	-6.1841	37.6389	-43.0309
15	0.4575	-5.8844	35.8656	-41.0274
20	0.3426	-4.7050	31.8126	-37.2800

Notes:

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

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

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

Table 6-9 Fuel Region 6: Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.45	**3.00**	**4.00**	**5.00**
0	0.00	31.47	45.11	57.00
5	0.00	28.87	41.42	53.01
10	0.00	27.13	39.16	50.29
15	0.00	25.96	37.56	48.37
20	0.00	25.06	36.61	46.97

Note:

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

6.2 ANALYSIS RESTRICTIONS

The purpose of this section is to summarize the restrictions of the fuel being stored at Palo Verde. One assembly pitch is defined as 1 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 pins missing, leaving a water hole, must be treated as fresh.

- Fuel assemblies which have had fuel pins replaced with either stainless steel or natural uranium pins may be stored as normal (by initial enrichment and burnup).

- Reconstituted fuel which contains fuel pins 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 to be empty must contain only water or a cell blocking device as described in Section 5.2.1.

- Any storage cells considered damaged (outside of their allowable tolerances) cannot be used to store fuel assemblies without further evaluation. These damaged cells may be used to store non-fuel assembly components such as failed fuel baskets or credited as empty cells in a storage array.

- SNAP-INs shall be oriented in the storage cells in the same direction as the L-inserts.

6.3 SOLUBLE BORON CREDIT

Soluble boron is credited in the Palo Verde SFP to keep $k_{eff} \leq 0.95$ under all normal and credible accident scenarios. Under normal conditions, this requires less than 550 ppm of soluble boron. Under accident conditions except for the multiple misload, 1200 ppm of soluble boron is required to ensure $k_{eff} \leq 0.95$, this leaves significant margin to the proposed Technical Specification value of 2150 ppm. The infinite 1oo4 multiple misload accident requires 1600 ppm of soluble boron.

7 REFERENCES

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

2. Westinghouse Document WCAP-16045-P-A, Rev. 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

3. "Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6.1, November June 2011.

4. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," U.S. Nuclear Regulatory Commission, February 2000.

5. Westinghouse Document WCAP-9522, "FIGHTH – A Simplified Calculation of Effective Temperatures in PWR Fuel Rods for Use in Nuclear Design," May 1979.

6. K.Wood, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," DSS-ISG-2010-001, Accession Number ML102220567, Nuclear Regulatory Commission, Rockville, MD, August 2010.

7. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," Oak Ridge National Laboratory, March 2003.

8. NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit,", Oak Ridge National Laboratory, March 2002.

9. EPRI Report 1022909, "Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty," Electric Power Research Institute, August 2011.

10. EPRI Report 1025203, "Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation," Electric Power Research Institute, April 2012.

11. L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," Nuclear Regulatory Commission, Rockville, MD, August 1998.

12. NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality (k_{eff}) Predictions," Oak Ridge National Laboratory, April 2012.

13. Westinghouse Document WCAP-16541-NP, Rev. 2, "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," June 2008.

14. C.E. Sanders, et al., "Parametric Study of the Effect of Control Rods for PWR Burnup Credit," NUREG/CR-6759, Oak Ridge National Laboratory, Oak Ridge, TN, February 2002.

15. "Transport and release properties of gaseous and volatile fission products in nuclear fuels - relevance and state of art of their characterizations in the laboratory," ISSN 0976-2108; No. 252, Bhabha Atomic Research Centre, Jan-Feb Bimonthly Newsletter, January 2005.

16. NUREG/CR-6487, "Containment Analysis for Type B Packages Used to Transport Various Contents," November 1996.

17. PNNL-18212 Rev 1 (ML112070118), "Update of the Gap Release Fractions for Non-Loca Events Utilizing the Revised ANS 5.4 Standard," June 2011.

APPENDIX A
VALIDATION OF SCALE 6.1.2

A.1 INTRODUCTION

This validation suite is intended to be used for fresh and spent fuel storage in the Palo Verde criticality safety analysis. Several types of calculations occur frequently with respect to analyzing fuel storage:

- Fresh fuel storage with absorbers credited; for example, a storage rack containing fresh fuel assemblies, either the storage rack or the fuel contains neutron absorbers which are credited for reactivity suppression.

- Fresh fuel storage without absorbers credited; for example, in a NFSR that does not have installed poisons and where the analysis does not credit absorbers in the fuel.

- A mixture of spent and new fuel with absorbers credited; for example, a spent fuel pool which takes credit for installed poisons, soluble boron, and/or burnable absorbers in the fuel.

- A mixture of spent and new fuel without absorbers credited; for example, a spent fuel pool which does not take credit for installed poisons, soluble boron, or burnable absorbers in the fuel.

In order to validate the SCALE Version 6.1.2 code system with the 238-group ENDF/B-VII library (referred to hereafter as SCALE) for the Palo Verde 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 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 Palo Verde Units 1, 2, and 3. It also covers the criticality analysis of all normal operations and postulated accidents in the spent fuel pool and fresh fuel storage. The validation is adequate to cover all present and anticipated (non-mixed-oxide) light water reactor (LWR) fuel designs at Palo Verde.

A.2 METHOD DISCUSSION

The validation methods recommended by 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 four subsets of experiments; fresh fuel with strong absorbers, fresh fuel without strong absorbers, fresh and burnt fuel with strong absorbers, and fresh and burnt fuel without strong absorbers. No critical experiments containing Gadolinia or Erbia were used because they will not be credited in the Palo Verde Criticality Safety Analysis either as fresh or residual absorbers.

[

]^{a,c}

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 4.5 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.
[

]^{a,c}

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,c}

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. [

]^{a,c}

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 where n is no larger than 50. Its only limitation is that it is applicable only to sample sizes between 3 and 50.

The null and alternative hypotheses are:

H_0: The sample comes from a normal distribution.

H_1: The underlying distribution is not normal.

The test statistics are:

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

where,

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

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

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=α.

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 applicable only to 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 statistics are:

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

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

[

]^{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 here. This approach involves determining a weighted mean that incorporates the uncertainty from both the measurement (σ_{exp}) and the calculation method (σ_{calc}). For the benchmark experiments chosen from References A2 and A5 through A7, the experimental uncertainties presented in References A2 and A5 through A7 are used. 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}):

$$k_{norm} = \frac{k_{calc}}{k_{exp}} \qquad \text{Equation A-5}$$

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

The Monte Carlo calculational uncertainty (σ_{calc}) and experimental uncertainties (σ_{exp}) are root-sum-squared to create a combined uncertainty for each experiment:

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

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 uncertainty for an "ith" k_{eff} is shown as σ_i and is taken to mean the combined calculational and experimental uncertainty, shown above as σ_t. The weighted equation variables for the single-sided lower tolerance limit are as follows:

Variance about the mean:

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

Average total uncertainty:

$$\bar{\sigma}^2 = \frac{n}{\Sigma\frac{1}{\sigma_i^2}} \qquad \text{Equation A-8}$$

The weighted mean k_{eff} value:

$$\overline{k}_{eff} = \frac{\sum \frac{1}{\sigma_i^2} k_{eff_i}}{\sum \frac{1}{\sigma_i^2}} \quad \text{Equation A-9}$$

The square root of the pooled variance:

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

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:

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

From Reference A1, 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:

$$K_L = \bar{k}_{eff} - U_{(n)} S_P \quad \text{Equation A-12}$$

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

$$K_L = 1 - U_{(n)} S_P \quad \text{Equation A-13}$$

Where, S_P is the pooled variance, and

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

US_P is then taken as the uncertainty to the untrended bias (methodology 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 determine a trend in the bias. In the following equations, "x" is the independent variable representing some parameter (e.g., enrichment). The variable "y" represents k_{eff}. Variables "a" and "b" are coefficients for the function where "b" is the slope and "a" is the intercept. 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.

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

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

$$b = \frac{1}{\Delta}\left[\sum\frac{1}{\sigma_i^2}\sum\frac{y_i x_i}{\sigma_i^2} - \sum\frac{x_i}{\sigma_i^2}\sum\frac{y_i}{\sigma_i^2}\right] \qquad \text{Equation A-14}$$

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

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:

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

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:

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

where,

SSE is the sum of the squares of the residuals:

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

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 should be used when it is more restrictive then the untrended value of the bias. In the area where untrended bias yields more restrictive value, the untrended bias and its associated uncertainty should be 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:

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

$K_{fit}(x)$ 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 $K_{fit}(x) > 1$:

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

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

γ = $\frac{1-p}{2}$

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

For a weighted analysis:

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

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

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

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

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

Within the equation for K_L:

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

And the uncertainty in the bias is:

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

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:

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

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 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. [ ]^{a,c} Thus, for a desired population fraction of 95% and k_{eff} [ ]^{a,c}, the percent confidence that a fraction of the population of n data points is above the lowest observed value is:

$$\beta = (1 - 0.95^n) \times 100\% \quad \text{Equation A-29}$$

Although 59 experiments would be required to reach a 95/95 tolerance limit 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:

$$Bias = \begin{matrix} k_{eff}^{min} - 1.0 & if\ k_{eff}^{min} < 1.0 \\ 0.0 & if\ k_{eff}^{min} \geq 1.0 \end{matrix} \qquad \text{Equation A-30}$$

And the uncertainty in the bias is:

$$Bias\ Uncertainty = uncertainty\ for\ k_{eff}^{min} \qquad \text{Equation A-31}$$

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.

A.3.1 [

]^{a,c}

[

]^{a,c}

A.3.2 [

]^{a,c}

[

]^{a,c}

[

]^{a,c}

A.3.3 [

]^{a,c}

[

]^{a,c}

[

]a,c

A.3.4 [

]a,c

[

]a,c

A.3.5 [

]^{a,c}

[

]^{a,c}

[

]^{a,c}

A.3.6 [

]^{a,c}

[

]^{a,c}

[

]a,c

A.3.7 [

]a,c

[

]a,c

A.3.8 [

]^{a,c}

[

]^{a,c}

A.3.9 [

]^{a,c}

[

]^{a,c}

[

]a,c

A.3.10 [

]a,c

[

]a,c

[

]a,c

A.3.11 [

]a,c

[

]a,c

A.3.12 [

]^{a,c}

[

]^{a,c}

A.3.13 [

]^{a,c}

[

]^{a,c}

A.3.14 [

]^{a,c}

[

]^{a,c}

[

]^{a,c}

A.3.15 [

]^{a,c}

[

]^{a,c}

[

]^{a,c}

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

A.3.16 [

]^{a,c}

[

]^{a,c}

A.3.17 Experiments from NUREG/CR-6361

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. However, Reference A3 contains several experiments that are considered important enough to be included. Table A-2 contains information from Table 2.1 of Reference A3 summarizing the cases modeled for benchmarking.

Table A-2 Summary of Benchmark Cases Chosen From NUREG/CR-6361 a,c

[

]a,c

A.3.18 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 is organized into three 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. [

]^{a,c}

[

]^{a,c}

A.4 RAW CALCULATION RESULTS

[

]^{a,c}

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

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

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

a,c

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

a,c

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

a,c

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

a,c

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

[ ]

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

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

a,c

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

a,c

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

a,c

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

Table A-6 [ ]^{a,c}
(cont.)

a,c

Table A-6 [ ]^{a,c} (cont.)

a,c

Table A-6 [ ]^{a,c}
(cont.)

a,c

Table A-6 [ ]^{a,c} (cont.)

a,c

Table A-6 (cont.) [ ]^{a,c}

a,c

Table A-7 [ ]^{a,c} a,c

Table A-7 [ ]^{a,c}
(cont.)

a,c

Table A-7 [ ]^{a,c} (cont.)

a,c

Table A-7 [ ]^{a,c}
(cont.)

a,c

Table A-7 [ ]^{a,c}
(cont.)

a,c

Table A-7 [ ]^{a,c}
(cont.)

a,c

A.5 DATA SET NORMALITY ASSESSMENT

A.5.1 [ ]^{a,c}

[ ]^{a,c}

Table A-8 [ ]^{a,c}

[ ]^{a,c}

[ ]^{a,c}

A.5.2 [ ]^{a,c}

[ ]^{a,c}

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

[ ]^{a,c}

[ ]^{a,c}

A.5.3 [ ]^{a,c}

[

]^{a,c}

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

[

]^{a,c}

A.5.4 [ ]^{a,c}

[

]^{a,c}

Table A-11 [ ]^{a,c} a,c

[

]^{a,c}

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.

A.6 TRENDING ANALYSIS

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

]^{a,c}

A.6.1 [ ]^{a,c}

[

]^{a,c}

a,c

Figure A-1. [ ]a,c

a,c

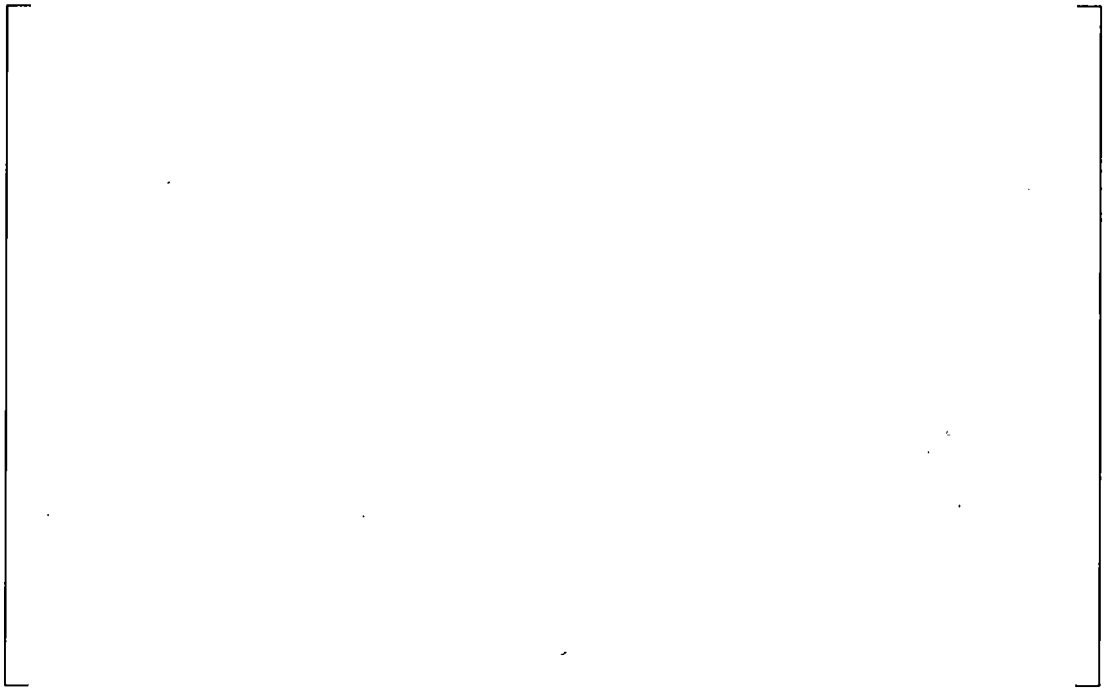


Figure A-2. [ ]a,c

a,c

Figure A-3. [**]a,c**

a,c

Figure A-4. [**]a,c**

[

]a,c

A.6.1.1 [ ]a,c

The values used to determine whether there is a statistically significant bias due to [ ]a,c are provided in Table A-12.

Table A-12 [ ]a,c a,c

[

]a,c

A.6.1.2 [ ]a,c

The values used to determine whether there is a statistically significant bias due to [ ]a,c are provided in Table A-13.

Table A-13 [ ][a,c] a,c

[

]a,c

A.6.1.3 [ ]^{a,c}

The values used to determine whether there is a statistically significant bias due to [ ]^{a,c} are provided in Table A-14.

Table A-14 [ ]^{a,c} a,c

[

]^{a,c}

A.6.1.4 [ ]^{a,c}

The values used to determine whether there is a statistically significant bias due to [ ]^{a,c} are provided in Table A-15.

Table A-15 [ ]^{a,c} a,c

[

]^{a,c}

A.6.1.5 [ ]^{a,c}

[

]^{a,c}

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

[

]^{a,c}

A.6.1.6 [ ]^{a,c}

[

]^{a,c}

Table A-17 [ ]^{a,c} a,c

[

[

]^{a,c}

[

]a,c

a,c

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

[ ]a,c

Table A-18 [ ][a,c]

a,c

[

]

[a,c]

Table A-19 [

]a,c

a,c

[

]a,c

Table A-20 [

][a,c] a,c

[

]a,c

A.6.1.7 [ ]^{a,c}

[

]^{a,c}

a,c

]^{a,c}

Figure A-6. [

[

]a,c

a,c

Figure A-7. [

]a,c

[ ]^{a,c}

Table A-21 [ ]^{a,c} a,c

[

]^{a,c}

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

[

]^{a,c}

A.6.1.8 [ ]^{a,c}

[

]^{a,c}

Table A-23 [

]^{a,c} a,c

Table A-24 [ ]^{a,c} a,c

A.6.2 [**]a,c**

[

]a,c

a,c

Figure A-8. [**]a,c**

a,c

Figure A-9. [ ][a,c]

a,c

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

a,c

Figure A-11. [ ]a,c

a,c

Figure A-12. [ ]a,c

a,c

Figure A-13. [ ]a,c

[

]a,c

Table A-25 [ ]^{a,c} a,c

[ ]^{a,c}

Table A-26 [**]**[a,c]**

a,c

A.6.3 [**]**[a,c]**

[

]a,c

a,c

Figure A-14. [**]**[a,c]**

a,c

Figure A-15. [ ]a,c

[

]a,c

a,c

Figure A-16. [

]a,c

a,c

Figure A-17. [

]a,c

[

]a,c

[

]a,c

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

[

]a,c

Table A-28 [

]^{a,c} a,c

[

]

A.6.4 [ ]^{a,c}

[

]^{a,c}

a,c

[ ]

Figure A-18. [ ][a,c]

a,c

[ ]

Figure A-19. [ ][a,c]

a,c

Figure A-20. [ ]a,c

a,c

Figure A-21. [ ]a,c

a,c

Figure A-22. [

]a,c

a,c

Figure A-23. [

]a,c

[

]a,c

Table A-29 [ ]a,c a,c

[

]a,c

Table A-30 [

]a,c a,c

A.7 AREA OF APPLICABILITY

As described in Reference A1, the AoA refers to the key physical parameter(s) that define a particular fissile configuration. This configuration can either be an actual system or a process. The AoA refers to the breadth of a physical parameter associated with a series of experiments. The AoA of this validation study is defined by the range of parameters in the validation suite and is summarized in Table A-31.

Table A-31 Area of Applicability a,c

A.8 VALIDATION SUMMARY

[

]a,c

Table A-32 Summary of Biases and Bias Uncertainties Determination a,c

[

]a,c

Table A-33 [ ][a,c] a,c

A.9 REFERENCES

A1. NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," Science Applications International Corporation, January 2001.

A2. "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC (95) 03, September 2013.

A3. NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," U.S. Nuclear Regulatory Commission, 1997.

A4. NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," U.S. Nuclear Regulatory Commission, September 2008.

A5. F.Fernex, "Programme HTC – Phase 1: Réseaux de crayons dans l'eau pure (Water-moderated and reflected simple arrays) Réévaluation des experiences," DSU/SEC/T/2005-33/D.R., Institut de Radioprotection et du Sûreté Nucléaire, 2008.

A6. F.Fernex, "Programme HTC – Phase 2: Réseaux simples en eau empoisonnée (bore et gadolinium) (Reflected simple arrays moderated by poisoned water with gadolinium or boron) Réévaluation des experiences," DSU/SEC/T/2005-38/D.R., Institut de Radioprotection et du Sûreté Nucléaire, May 2008.

A7. F.Fernex, "Programme HTC – Phase 3: Configurations "stockage en piscine" (Pool storage) Réévaluation des experiences," DSU/SEC/T/2005-37/D.R., Institut de Radioprotection et du Sûreté Nucléaire, May 2008.

A8. D. Lurie et al. "Applying Statistics," NUREG-1475, Revision 1, U.S. Nuclear Regulatory Commission, March 2011.

A9. R.I. Smith et al., "Clean Critical Experiment Benchmarks for Plutonium Recycle in LWR's," Volume I, EPRI NP-196, Electric Power Research Institute, April 1976.

A10. S.R. Bierman, "Criticality Experiments with Neutron Flux Traps Containing Voids," PNL-7167, Pacific Northwest Laboratory, April 1990.

A11. L.W. Newman, "Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," DOE/ET/34212-41 (BAW-1810), Babcock & Wilcox, April 1984.

A12. T.C. Engelder et al., "Spectral Shift Control Reactor Basic Physics Program, Critical Experiments on Lattices Moderated by D_2O-H_2O Mixtures," BAW-1231, Babcock & Wilcox Company, December 1961.

A13. T.C. Engelder et al., "Spectral Shift Control Reactor Basic Physics Program, Measurement and Analysis of Uniform Lattices of Slightly Enriched UO_2 Moderated by D_2O-H_2O Mixtures," BAW-1273, Babcock & Wilcox Company, November 1963.

APPENDIX B
ANALYSIS OF NEW AND INTERIM FUEL STORAGE RACKS AND FUEL TRANSFER EQUIPMENT

B.1 INTRODUCTION

The purpose of this appendix is to document the criticality safety analysis performed to support the operation of the Palo Verde NFSR, Intermediate Fuel Storage Rack (IFSR), New Fuel Elevator, and Fuel Upender and Transfer Machine. When discussing the New Fuel Elevator and Fuel Upender and Transfer Machine together, they will be referred to as the Fuel Handling Equipment (FHE). This appendix considers Palo Verde's past, current, and planned future fuel designs.

B.2 ACCEPTANCE CRITERIA

The existing NFSR, IFSR, and FHE are evaluated to confirm that each system maintains subcriticality while performing their designed purposes.

B.2.1 Acceptance Criteria

The objective of this criticality safety analysis is to ensure that the fuel storage operations are within the bounds of Code of Federal Regulations, Title 10, Part 50, Section 68, subsection b2 and b3, "Criticality Accident Requirements" (Reference B1) discussed here.

1. The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

2. If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

B.2.2 Design Approach

For the NFSR, IFSR, and FHE compliance is shown by demonstrating that the system k_{eff} does not exceed 0.95 at a 95 percent probability with 95 percent confidence. A conservative combination of best estimate and bounding values have been selected to model the fuel in this analysis to ensure that fuel represented by the proposed Palo Verde Technical Specifications is less reactive than the fuel modeled in this analysis.

B.2.3 Computer Codes

The analysis methodology employs SCALE Version 6.1.2, as documented in ORNL/TM-2005/39, "SCALE: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation" (Reference B2), with the 238-group cross-section library based on ENDF/B-VII. Section 2.3 describes the SCALE code and the 238 group library. Note that all analyses performed here use the "Fresh fuel without Absorber" validation suite.

B.3 PALO VERDE GENERATING STATION

This section describes the physical characteristics of Palo Verde that are important to criticality safety. Fuel designs are discussed in Section B.3.1 and the physical characteristics of the NFSR and FHE are discussed in Section B.3.2.

B.3.1 Fuel Description

All fuel assemblies used at Palo Verde incorporate a 16x16 square array of 236 fuel rods with four GTs and one IT. The fuel rod cladding material is Zircaloy and its variants such as ZIRLO High Performance Fuel Cladding Material. Each fuel rod contains a column of enriched UO_2 fuel pellets. The pellets are pressed and sintered, and are dished on both ends.

Table B-1 provides basic data on the fuel types that have been used at Palo Verde.

Table B-1 Fuel General Specifications

Parameter	Value
Fuel lattice	16x16
Fuel design 1	CE STD
Fuel design 2	CE VAP
Fuel design 3	ANP
Fuel design 4	CE NGF

B.3.1.1 Fuel Designs

This section outlines the neutronically important mechanical features of the four (4) fuel designs outlined in Table B-2.

Table B-2 Fuel Assembly Mechanical Specifications				
Parameter	**Value**			
Assembly type	STD	VAP	ANP	NGF
Rod array size	16x16	16x16	16x16	16x16
Rod pitch, inch	0.506	0.506	0.506	0.506
Active fuel length, inch	150	150	150	150
Total number of fuel rods	236	236	236	236
Fuel cladding OD, inch	0.382	0.382	0.382	0.374
Fuel cladding ID, inch	0.332	0.332	0.332	0.329
Fuel cladding thickness, inch	0.025	0.025	0.025	0.0225
Pellet diameter, inch	0.325	0.3255	0.3255	0.3225
Number of guide/instrument tubes	4/1	4/1	4/1	4/1
Guide/instrument tube OD, inch	0.980/0.980	0.980/0.980	1.023/0.980	0.980/0.980
Guide/instrument tube thickness, inch	0.04/0.04	0.04/0.04	0.0615/0.04	0.04/0.04

[

]^{a,c}

B.3.2 Fuel Storage and Handling Equipment Descriptions

The physical characteristics of the NFSRs, IFSRs, and FHE are described in this section. The three units use fuel storage and handling equipment of the same design and so all calculations performed are applicable to all three units.

B.3.2.1 New Fuel Storage Rack Description

The NFSR facility consists of two 15x3 assembly arrays that are divided by a 2-foot concrete wall. A total of 90 assemblies can be stored in this facility. The mechanical design of the NFSR is given in Table B-3.

Table B-3 New Fuel Storage Rack Specifications		
Parameter	**Value**	**Tolerance**
Rack cell pitch East-West, in	31.125	+0.125/-0.0
Rack cell pitch North-South, in	18.125	+0.125/-0.0
Cell ID, in	8.6875	± 0.0625
Cell wall thickness, in	0.12	Nominal
NFSR wall thickness, in	24	Nominal
NFSR wall material	Concrete	N/A

B.3.2.2 Intermediate Fuel Storage Racks

The IFSR is a four cavity fuel storage rack in a 1x4 array. It is designed as an intermediate storage location for fuel bundles during refueling. New fuel may be stored before being moved into the core, partially spent fuel may be moved out of the core and stored temporarily to provide spaces for fuel shuffling, or spent fuel may be stored before being sent to the spent fuel pool. The mechanical design of the IFSR is given in Table B-4.

Table B-4 Intermediate Fuel Storage Rack Specifications		
Parameter	**Value**	**Tolerance**
Rack cell pitch East-West, in	18.5625	+0.125/-0.0
Cell ID, in	8.6875	± 0.0625
Cell wall thickness, in	0.120	+0.0/-0.120

B.3.2.3 Fuel Upender and Transfer Machine

The Transfer Machine or Carriage conveys the fuel assemblies through the transfer tube. Two fuel assembly cavities are provided in the fuel carriage to reduce overall fuel handling time. After the refueling machine deposits a spent fuel bundle in the open cavity, it only has to move approximately one foot to pick up the new fuel assembly which was brought from the fuel building in the other cavity. The handling operation in the fuel building is similar. The dual cavity arrangement permits both fuel handling machines to travel fully loaded at all times. Fuel assemblies are placed on the transfer carriage in a vertical position, lowered to the horizontal position, moved through the fuel transfer tube on the transfer carriage, and then restored to the vertical position. Wheels support the carriage and allow it to roll on tracks within the transfer tube. The track sections at both ends of the transfer tube are mounted on the upending machines to permit the carriage to be properly positioned at the limits of its travel.

An upending machine is provided at each end of the transfer tube. Each machine consists of a structural support base from which is pivoted an upending straddle frame which engages the two-pocket fuel carrier. Hydraulic cylinders, attached to both the upending frame which engages the support base, rotate the fuel

carrier between the vertical and horizontal position as required by the transfer procedure. A third fuel assembly was modeled 5 inches from the transfer carriage to allow for the presence of an additional fuel assembly no closer than 5 inches from the carriage. The mechanical design of the machine is given in Table B-5.

Table B-5 Fuel Upender and Transfer Machine

Parameter	Value	Tolerance
Cavity pitch, in	14.25	+0.0/-1.25
Cell ID, in	9.00	± 0.060
Cell wall thickness, in	0.120	+0.0/-0.120

B.3.2.4 New Fuel Elevator

The New Fuel Elevator is utilized to lower new fuel from the operating floor to the bottom of the pool where it is grappled by the spent fuel handling tool. The elevator is powered by a cable winch and fuel is contained in a simple support structure whose wheels are captured in two rails. The mechanical design of the New Fuel Elevator is given in Table B-6.

Table B-6 New Fuel Elevator Specifications

Parameter	Value	Tolerance
Cell ID, in	9.0	± 0.060
Cell wall thickness, in	0.1875	+0.0/-0.1875
Separation from Racks, East-West, in	10.3125	Nominal
Separation from Racks, South, in	7.3125	Nominal
Elevator material	Stainless Steel 304	N/A

B.4 CRITICALITY ANALYSIS

This section describes the reactivity calculations and evaluations performed in accordance with the requirements of Reference B1at Palo Verde and confirming continued safe operation during both normal and accident conditions.

B.4.1 Keno Modeling Simplifications & Assumptions

As discussed in Section B.2.3, KENO is the criticality code used to support this analysis. KENO is used to determine the absolute reactivity of fresh fuel assemblies in the NFSR, IFSR, and FHE. [

]^{a,c}

[

]^{a,c}

B.4.2 Limiting Fuel Design Selection

There are four potentially limiting fuel designs that have been used at Palo Verde, shown in Table B-2. Of these fuel designs one, VAP, is currently in use on site. The other, NGF is planned for use in the future. Therefore both VAP and NGF fuel designs will be considered the two potentially limiting fuel designs. The STD and ANP fuel designs do not need to be addressed for these calculations because they are bounded by the VAP fuel design.

[

]^{a,c}

B.4.3 Treatment of Concrete

Concrete is a material which has a large variety of different potential compositions, all of which can be labeled as "concrete." The variety of potential concretes presents a challenge in determining what the correct concrete composition is to use in an analysis. Further complicating the problem is the fact that concrete compositions can change over time, potentially significantly affecting the concrete's impact on system reactivity. [

]^{a,c}

Table B-7 [ ]^{a,c} a,c

B.4.4 Target k_{eff} Determination

As discussed in Section B.2.1, this analysis provides confirmation that the Palo Verde NFSR, IFSR, and FHE meet the regulatory requirements. [

]^{a,c}

[ ]^{a,c}Equation B-1

[

]^{a,c}

B.4.5 Biases & Uncertainties Determination

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

]^{a,c}

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

B.4.5.1 Manufacturing Tolerances

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

]^{a,c}

[

]a,c

a,c

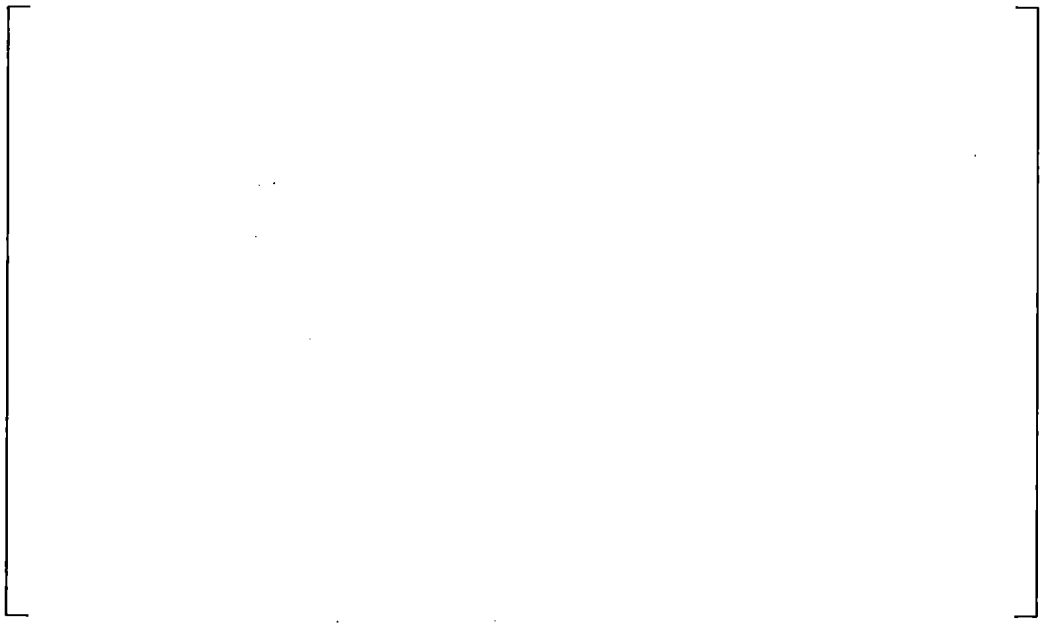


Figure B-1. [ ]a,c

a,c

Figure B-2. [ ]a,c

[

]a,c

B.4.5.2 Structural Material Presence

[

]a,c

B.4.5.3 Eccentric Fuel Assembly Positioning

[

]a,c

B.4.5.4 Other Uncertainties

An uncertainty in the predictive capability of SCALE 6.1.2 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.

B.4.5.5 SFP Temperature Bias

The Palo Verde NFSR, IFSR, and FHE do not have nominal temperatures; instead they operate within an allowable range. [

]^{a,c}

B.4.5.6 Planar Enrichment Bias

[

]^{a,c}

B.4.5.7 Borated and Unborated Biases and Uncertainties

[

]^{a,c}

B.4.6 New Fuel Storage Rack Analysis

The criticality safety analysis for the NFSR consists of determining the limiting fuel design under both the fully flooded and optimum moderation condition. Then biases and uncertainties for both fully flooded and optimum moderation conditions are calculated using the limiting fuel design. The mechanical parameters used to model the NFSR in KENO are given in Table B-8.

Table B-8 New Fuel Storage Rack Specifications

Parameter	Value (Tolerance)	Value Analyzed
Rack cell pitch East-West, in	31.125 +0.125/-0.0	30.89 (Min)
Rack cell pitch North-South, in	18.125 +0.125/-0.0	17.89 (Min)
Cell ID, in	8.6875 ± 0.0625	8.6875 ± 0.0625
Cell wall thickness, in	0.12	0.10 (Min)
NFSR wall thickness, in	24 (Nominal)	24
NFSR wall material	Concrete	[ ]^{a,c}

Table B-9 Limiting Fuel Design Study a,c

[

]a,c

Table B-10 Biases & Uncertainties for the New Fuel Storage Rack (5.0 wt% ^{235}U) a,c

As shown in Table B-10, the best estimate k_{eff} of the NFSR under both full density water and optimum moderation conditions is less than the target k_{eff}. This demonstrates that the NFSR complies with the requirements of Reference B1.

B.4.7 Intermediate Fuel Storage Racks

The criticality safety analysis for the IFSR consists of determining the target k_{eff} for the IFSR then confirming that the best estimate system k_{eff} (plus 2 σ) is below the target k_{eff} with the limiting fuel design. The analysis uses the NGF fuel design based on the comparison of NGF and VAP fuel performed in Sections 4.3 and B.4.6 which both demonstrate that the NGF fuel is more reactive than VAP fuel with full density water. The mechanical parameters used to model the NFSR in KENO are given in Table B-11.

Table B-11 Intermediate Fuel Storage Rack Specifications		
Parameter	**Value (Tolerance)**	**Value Analyzed**
Rack cell pitch East-West, in	18.5625 (+0.125/-0.0)	18.09 (Min)
Cell ID, in	8.6875 (±0.0625)	8.6875 (±0.063)
Cell wall thickness, in	0.120 (+0.0/-0.120)	0.120 (+0.0/-0.120)

Biases and uncertainties will be calculated using the NGF fuel design. The results of the biases and uncertainties calculation are provided in Table B-12.

Table B-12 Biases & Uncertainties for the Intermediate Fuel Storage Rack a,c

As shown in Table B-12, the best estimate k_{eff} of the IFSR is less than the target k_{eff}. This demonstrates that the NFSR complies with the requirements of Reference B1.

B.4.8 Fuel Upender and Transfer Machine

The criticality analysis for the Fuel Upender and Transfer Machine demonstrates that the machine can be used with fresh 4.65 wt% ^{235}U fuel without exceeding a k_{eff} of 0.95 at a 95 percent probability, 95 percent confidence interval. The design basis fuel is the NGF fuel design.

Table B-13 Biases & Uncertainties for the Fuel Upender and Transfer Machine a,c

As shown in Table B-14, the best estimate k_{eff} of the Fuel Upender and Transfer Machine is less than the target k_{eff}. This demonstrates that the NFSR complies with the requirements of Reference B1.

B.4.9 New Fuel Elevator

The criticality analysis for the New Fuel Elevator demonstrates that the machine can be used with fresh 5.0 wt% ^{235}U fuel without exceeding a k_{eff} of 0.95 at a 95 percent probability, 95 percent confidence interval. The design basis fuel is the NGF fuel design. The mechanical parameters and tolerances for the new fuel elevator are provided in Table B-14. [

]^{a,c}

Table B-14 New Fuel Elevator Specifications

Parameter	Value (Tolerance)	Value Analyzed
Cell ID, in	9.00 (±0.060)	9.00 ±0.060
Cell wall thickness, in	0.1875 (+0.0/-0.1875)	0.1875 +0.0/-0.1875
Separation from Racks, East-West, in	10.3125	10.3125
Separation from Racks, South, in	7.3125	7.3125
Elevator material	Stainless Steel 304	Stainless Steel 304

Biases and uncertainties will be calculated using the NGF fuel design. The results of the biases and uncertainties calculation are provided in Table B-15.

Table B-15 Biases & Uncertainties for the New Fuel Elevator a,c

As shown in Table B-15, the best estimate k_{eff} of the New Fuel Elevator is less than the target k_{eff}. This demonstrates that the NFSR complies with the requirements of Reference B1.

B.5 ACCIDENTS

The analysis of the NFSR, IFSR, and FHE has evaluated the potential impacts of accidents. The evaluation for each structure is as follows.

The normal condition of the NFSR is dry. The regulatory requirements are for the analysis of the two potential accident conditions, NFSR flooded with full density water or flooded with a hydrogenous material which provides optimum moderation.

[

]^{a,c}

B.6 ANALYSIS RESULTS & CONCLUSIONS

The analyses of the NFSR, IFSR, and FHE have demonstrated that these components can be operated in their design capacity without risk of exceeding the maximum reactivity imposed by regulation. The analyses support use of these components at up to 4.65 wt% ^{235}U for the IFSR and fuel upender and 5.0 wt% ^{235}U for the NFSR and new fuel elevator and with the STD, VAP, ANP, and NGF fuel designs discussed in Section B.3.1.1.

B.7 REFERENCES

B1. Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements."

B2. "Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6.1, November June 2011.