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Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation – Revision 1

2018 TECHNICAL REPORT

Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation – Revision 1

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NOTE

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Abstract

To conservatively apply burnup credit in spent fuel criticality analyses, code validation for both fresh and used fuel is required. Fresh fuel validation is typically performed by modeling experiments from the International Handbook of Evaluated Criticality Safety Benchmark Experiments. Burnup credit methodologies for criticality safety analyses include determination of the isotopic inventory of the discharged fuel in order to take credit for the reduced reactivity of the fuel due to the burnup of fissile material and the accumulation of fission products (FPs). One important aspect of the burnup credit is the uncertainty associated with the reactivity worth of the isotopic inventory in spent fuel. Historically, engineering judgment has been used for determining the depletion uncertainty for spent fuel pool (SFP) burnup credit calculations. EPRI recently developed a series of experimental benchmarks that can be used to quantify reactivity decrement biases and uncertainties of other code systems used in spent-fuel pool (SFP) and cask criticality analyses. Final 95/95 tolerance limits for the measured reactivity decrements were shown to be accurate within 3% of the fuel assembly depletion reactivity. In this report, Pressurized water reactor (PWR) burnup credit validation is demonstrated using the EPRI benchmarks for quantifying fuel reactivity decrements.

Keywords
Burnup credit
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PRIMARY AUDIENCE: Nuclear criticality safety analysts at nuclear power plants and regulators

SECONDARY AUDIENCE: Nuclear criticality safety analysts at research organizations and vendors

KEY RESEARCH QUESTION

The companion report (3002010613, Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty-Rev 1) answered the question of "What are the magnitude of the bias and uncertainty in analytical computations of depleted fuel reactivity for spent fuel criticality analysis that arises from fuel burnup during nuclear plant operation?" This report focuses on how to use the 11 EPRI benchmarks for spent fuel depletion and criticality analysis.

RESEARCH OVERVIEW

The measured depletion reactivity decrements for eleven Pressurized Water Reactor (PWR) benchmark lattices are tabulated and provided to spent fuel criticality analysts so that they can quantify the biases and uncertainties of their specific lattice depletion and criticality computational tools. As an example, the computations are performed using SCALE modules, TRITON for depletion and KENO for criticality, to provide guidance to the user.

KEY FINDINGS

- 95/95 tolerance limits for the measured reactivity decrements are shown to be less than 3% of the fuel assembly depletion reactivity.
- The computational results, based on SCALE, demonstrated that the agreement between measured and predicted reactivity decrements is better when ENDF/B-VII libraries are used.
- It is recommended to use ENDF/B-VII as cross section library for depletion and criticality analysis to avoid excessive conservatism.

WHY THIS MATTERS

Quantification of reactivity decrement uncertainties and biases are an essential part of the criticality analysis for spent fuel. In this report and companion report, an approach for quantification of bias and uncertainty is presented.

HOW TO APPLY RESULTS

In this report, comparison to EPRI benchmarks using SCALE computer code is presented. Additionally, to guide future users, steps for utilizing EPRI benchmarks are described.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- This report, along with companion report (3002010613, Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty-Rev. 1) is aimed for criticality analysts that are working on burnup credit.
- In addition to the nuclear criticality analysts at nuclear power plant, this report is useful for criticality analysts at research organizations. NEI 12-16, Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants, provides additional resources for complete analysis.

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Acronyms

ENDF	Evaluated Nuclear Data File
EPRI	Electric Power Research Institute
IFBA	Integral Fuel Burnable Absorber
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
SFP	Spent Fuel Pool
WABA	Wet Annular Burnable Absorber

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Section 1: Introduction

Criticality analysis of spent fuel pools and casks is performed using Monte Carlo or multi-group transport calculations. The criticality calculations for spent fuel have two fundamental sources of uncertainty: 1) the nuclide inventory of the fuel assemblies, and 2) neutron cross-section data. For fresh fuel, these uncertainties can be quantified directly by making comparisons to experimental benchmarks that have been extensively validated. Such analyses provide quantification of computational uncertainties as a function of fuel assembly design, fuel pin enrichments, storage cell geometries, coolant temperature, coolant boron concentration, etc. Such uncertainties include contributions from: the nuclide inventory of the fuel assemblies, basic neutron cross-section data, and analytical methods.

However, for depleted fuel assemblies, there is currently not sufficient critical experiments and given how expensive such a program would be, it is not clear this data will be available in the near future. In order to compensate for the lack of critical experiments with spent fuel, historically, engineering judgment has been used for determining the depletion uncertainty for spent fuel pool (SFP) burnup credit calculations. The Kopp memorandum (NRC Memorandum, 1998) considers it an acceptable assumption to use an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest, in the absence of any other determination of the depletion uncertainty [1]. The guidance in the "Kopp Memo" provided regulatory clarity and stability for many years. Starting in 2005, regulatory staff positions on acceptable criticality analysis methods evolved through interactions with licensees, and the basis for the guidance in the Kopp Memo was portrayed as being insufficient in light of operational/licensing changes being sought by utilities. In 2010, the NRC requested applicants to supply quantification and/or justification for the 5% uncertainty assumption.

To address this issue, EPRI sponsored a project for quantifying fuel depletion uncertainties by using operational reactor data and corresponding reactor analysis tools. As part of routine PWR operation, in-core flux map data are usually taken every 30 days. This flux map data can be used to quantify the accuracy of computed assembly "power distributions." This data is routinely used to determine 95/95 confidence intervals on predicted assembly and pin power distributions needed for NRC licensing of core designs. The analytical methods employed in this project use the same measured flux map data and core analysis tools to deduce errors in assembly reactivities at each flux map and to determine the assembly reactivity decrement bias and uncertainty.

The methods and resulting benchmarks from this study were published in 2011 in an EPRI report [2], which provides a set of benchmarks that are to be used for validation of burnup credit. The previous approach recommended using constant values for bias and uncertainty of the reactivity decrement [4, 5] independent of the burnup value. Recently, this report has been updated to include a more rigorous statistical approach for uncertainty and bias quantification [3] and subsequently, recommended using burnup dependent values for bias and uncertainty.

In this report, an example analysis and application of the benchmarks in burnup credit are presented. The analysis is performed using SCALE 6.1 [6] with ENDF/B-V and ENDF/B-VII cross section libraries. Depletion calculations are performed using TRITON and criticality calculations are performed using the KENO module.

Section 2: Review of the EPRI Depletion Benchmarks

PWR depletion benchmarks were developed by EPRI using a large set of power distribution measurements to determine reactivity biases. For this purpose, 680 flux maps from 44 cycles of PWR operation at 4 different plants were used to infer the depletion reactivity [2, 3]. The predicted reactivity of the fuel assemblies was adjusted to find the best match between the predicted and measured power distribution. With these adjustments, bias as a function of burnup was quantified for the CASMO code. It was demonstrated that these biases were independent of fuel type, enrichment, soluble boron concentration, etc. Experimental reactivity decrement benchmarks were then constructed by adding CASMO biases to the CASMO calculated reactivity decrements.

The depletion reactivity has been used to create 11 benchmark cases for 10, 20, 30, 40, 50, and 60 GWd/MTU and 3 cooling times 100 h, 5 years, and 15 years. All of these benchmark cases can be analyzed with the depletion code to be used in the criticality analysis to establish a bias for the depletion reactivity. The uncertainty in the benchmarks is used as the depletion reactivity uncertainty. These biases and uncertainties cover both the isotopic content uncertainty and the worth uncertainty associated with depletion. They account for all the changes from the initial fresh fuel condition to burnup up to 60 GWd/MTU.

2.1 EPRI Benchmark Specifications

The benchmarks are a 2D infinite lattice of 17 x 17 PWR fuel assemblies with a typical in-core assembly pitch. The nominal depletion conditions for the benchmarks include 900 ppm constant soluble boron concentration at 580 K moderator and 900 K fuel temperatures. The nominal pool conditions are 300 K water temperature and 0 ppm soluble boron. A summary of the description of the 11 EPRI benchmarks are presented in Table 2-1. In this table:

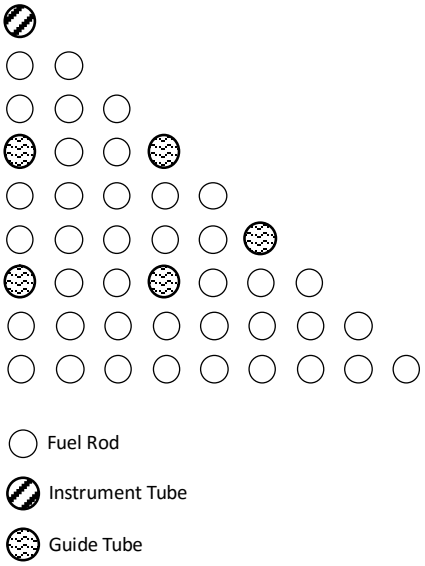
- Cases 1-3 provide depletions corresponding to three different enrichments: 3.25, 5.0, and 4.25 wt% U-235.
- The 4.25 wt% U-235 enrichment (Case 3) is then used as the base case for the following cases. Cases 4-8 each perturb a single parameter in the depletion: smaller pin diameter, 20 Wet Annular Burnable Absorber (WABA) rods, 104 Integral Fuel Burnable Absorber (IFBA) rods, 20 WABA rods with 104 IFBA rods, and higher soluble boron.

- Cases 9 and 10 use the depletions from Case #3 (4.25 wt% U-235), but perturb only the final conditions (rack).
- Finally, Case 11 provides depletions using 1.5 times the nominal specific power.

Table 2-1
EPRI Benchmark Specifications

1	3.25% Enrichment
2	5.00% Enrichment
3	4.25% Enrichment
4	off-nominal pin diameter depletion
5	20 WABA depletion
6	104 IFBA depletion
7	104 IFBA plus 20 WABA depletion
8	high boron depletion = 1500 ppm
9	Nominal Case Branch to SFP Hot Isothermal Temperatures = 150°F
10	Nominal Case Branched to SFP High Boron Concentration = 1500 ppm
11	high power depletion (power, coolant/fuel temp)

The benchmark geometry and nominal parameters are illustrated in Figure 2-1. In order to aid with the analysis, the benchmark specifications for all cases are reproduced in Appendix A of this report.



Physical Description

Number of pins along side	17
Pin pitch	1.2598 cm
Inter-assembly spacing	21.5036 cm
Fuel pellet OR	0.4096 cm
Clad IR	0.4180 cm
Clad OR	0.4750 cm
Guide/instrument tube IR	0.5610 cm
Guide/instrument tube OR	0.6120 cm

Structural Material Description

Material (Zr-4)Density	6.55 (g/cm3)
Temp., unheated	580K
Temp., heated	$0.12*T_{fuel}+0.88*T_{coolant}$
Nuclide	**Number Density**
Zr-4	4.32444E+22

Coolant Description, Depletion (Nominal)

Boron Concentration	900 ppm
Temperature	580 K
Nuclide	**Number Density**
H	4.75756E+22
O	2.37894E+22
B	3.56773E+19

Coolant Description, Cold

Boron Concentration	0 ppm
Temperature	293 K
Nuclide	**Number Density**
H	6.67431E+22
O	3.33738E+22

Figure 2-1
Description of the Nominal Assembly for the EPRI Benchmarks

2.2 Measured Reactivity Decrements and Uncertainty of Measurements for EPRI Benchmarks

The measured reactivity decrements for the 11 benchmarks for 100 hour, 5 year, and 15 year are presented in Table 2-2, Table 2-3, and Table 2-4, respectively. These values are reproduced from Reference 3 and included in this report to aid with the analysis. The measured reactivity decrement biases and tolerance limits, expressed in units of pcm and as a percentage of the absolute value of reactivity decrement, are presented in Table 2-5 [3].

It should be noted that Table 2-5 contains both biases and uncertainties. Since the experimental biases for EPRI benchmarks are already added to the measured reactivity decrement values, tabulated in Tables 2-2 to 2-4, the application of EPRI benchmarks requires inclusion of only uncertainties in the final analysis.

Table 2-2
Measured Reactivity Decrements for EPRI Benchmarks – 100-Hour Cooling

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.1324	0.2334	0.3208	0.3960	0.4570	0.5032
2	0.1141	0.2016	0.2803	0.3549	0.4254	0.4897
3	0.1218	0.2152	0.2987	0.3762	0.4461	0.5059
4	0.1202	0.2171	0.3072	0.3935	0.4731	0.5415
5	0.2040	0.2330	0.2995	0.3721	0.4388	0.4962
6	0.1731	0.2210	0.2965	0.3730	0.4434	0.5039
7	0.2519	0.2413	0.2978	0.3690	0.4359	0.4940
8	0.1211	0.2124	0.2929	0.3666	0.4326	0.4890
9	0.1232	0.2166	0.2995	0.3760	0.4448	0.5035
10	0.0962	0.1779	0.2527	0.3221	0.3842	0.4365
11	0.1230	0.2144	0.2942	0.3668	0.4315	0.4868

Table 2-3
Measured Reactivity Decrements for EPRI Benchmarks – 5-Year Cooling

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.1365	0.2466	0.3444	0.4288	0.4967	0.5475
2	0.1158	0.2081	0.2940	0.3765	0.4545	0.5252
3	0.1242	0.2240	0.3161	0.4022	0.4797	0.5455
4	0.1227	0.2258	0.3247	0.4201	0.5079	0.5827
5	0.2064	0.2419	0.3168	0.3978	0.4719	0.5351
6	0.1755	0.2299	0.3137	0.3988	0.4767	0.5433
7	0.2542	0.2502	0.3151	0.3945	0.4688	0.5326
8	0.1236	0.2213	0.3103	0.3926	0.4661	0.5284
9	0.1256	0.2252	0.3165	0.4013	0.4775	0.5420
10	0.0981	0.1853	0.2672	0.3434	0.4112	0.4677
11	0.1263	0.2240	0.3122	0.3932	0.4652	0.5262

Table 2-4
Measured Reactivity Decrements for EPRI Benchmarks – 15-Year Cooling

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.1417	0.2650	0.3765	0.4724	0.5487	0.6051
2	0.1179	0.2179	0.3137	0.4062	0.4934	0.5720
3	0.1272	0.2367	0.3402	0.4373	0.5242	0.5972
4	0.1255	0.2380	0.3485	0.4555	0.5532	0.6355
5	0.2097	0.2550	0.3412	0.4329	0.5161	0.5863
6	0.1787	0.2427	0.3379	0.4338	0.5210	0.5947
7	0.2576	0.2634	0.3395	0.4295	0.5127	0.5836
8	0.1268	0.2343	0.3348	0.4280	0.5108	0.5803
9	0.1286	0.2379	0.3405	0.4361	0.5214	0.5930
10	0.1009	0.1966	0.2882	0.3733	0.4484	0.5104
11	0.1295	0.2372	0.3370	0.4288	0.5100	0.5782

Table 2-5
Measured Reactivity Decrement Biases and Tolerance Limits Expressed as Percentage of Absolute Value of Depletion Reactivity Decrement

Burnup (GWd/MTU)	10	20	30	40	50	60
Bias(pcm)	66	101	106	80	22	-64
95/95 Tolerance Limit(pcm)	348	537	654	752	831	888
CASMO-5 Bias (% of depletion)	0.58	0.50	0.38	0.23	0.05	-0.13
95/95 Tolerance Limit (% of depletion)	3.05	2.66	2.33	2.12	1.95	1.81

2.3 Applicability of EPRI Benchmarks for Range of Fuel Types

The measured data for the EPRI benchmarks is based on a limited number of fuel assembly types; Westinghouse and AREVA 17x17 fuel. This section investigates the applicability of this data to other fuel types. The fuel design parameters for major PWR fuel types are tabulated in Table 2-6. The burnup averaged Energy of the Average Lethargy Causing Fission (EALF) for various fuel types are listed in Table 2-7. In these tables, the fuel types for which benchmarks are developed (W17x17 Std and W 17x17 OFA) are specified with bold font to enable easier distinction.

As evident from these tables, all PWR fuel designs are very similar and the key parameter that determines the difference between fuel assemblies is the amount of moderation. Using a spectral index provides a useful metric for comparing the benchmarks to other fuel designs. Since all the materials (UO_2, Zircaloy, and water) are the same for different fuel types, including the ones from which the benchmarks are created, if a trend existed it can be attributed to the relative proportion of these materials which is measured by the spectral index (i.e., EALF). Table 2-8 shows the average depletion EALF for all the benchmark cases. As can be seen from Table 2-8, there is a significant range of depletion spectra.

Table 2-6
Fuel Design Dimensions and Spectra

Fuel Design	Pellet OD cm (in.)	Clad OD cm (in.)	Pitch cm (in.)	Water to Pellet Volume Ratio
W 14X14 Std	0.930 (0.366)	1.072 (0.422)	1.412 (0.556)	1.61
W 14X14 OFA	0.874 (0.344)	1.016 (0.400)	1.412 (0.556)	1.97
W 15X15	0.930 (0.366)	1.072 (0.422)	1.430 (0.563)	1.68
W 16X16	0.820 (0.323)	0.950 (0.374)	1.232 (0.485)	1.53
W 17X17 Std	**0.820 (0.323)**	**0.950 (0.374)**	**1.260 (0.496)**	**1.67**
W 17x17 OFA	**0.785 (0.309)**	**0.914 (0.360)**	**1.260 (0.496)**	**1.93**
B&W 15X15	0.937 (0.369)	1.092 (0.430)	1.443 (0.568)	1.66
B&W 17X17	0.820 (0.323)	0.963 (0.379)	1.275 (0.502)	1.70
CE 14X14	0.958 (0.377)	1.118 (0.440)	1.473 (0.580)	1.66
CE 16x16	0.826 (0.325)	0.970 (0.382)	1.285 (0.506)	1.70

Table 2-7
Burnup Averaged Energy of the Average Lethargy Causing Fission (EALF) For Various Fuel Types (4.25 wt% U-235 Case 3 conditions)

Fuel Type	Burnup (GWd/MTU)					
	10	20	30	40	50	60
W 14x14	0.849	0.913	0.965	1.011	1.055	1.100
W 14x14 OFA	0.591	0.625	0.652	0.678	0.704	0.734
W 15x15	0.782	0.837	0.882	0.921	0.960	1.000
W 16x16	0.909	0.986	1.050	1.105	1.157	1.209
W 17x17	**0.781**	**0.839**	**0.887**	**0.929**	**0.970**	**1.011**
W 17x17 OFA	**0.608**	**0.646**	**0.678**	**0.706**	**0.734**	**0.765**
B&W 15x15	0.782	0.837	0.882	0.921	0.960	1.000
B&W 17x17	0.745	0.798	0.842	0.880	0.918	0.956
CE 14x14	0.757	0.807	0.848	0.884	0.919	0.957
CE 16x16	0.737	0.789	0.830	0.867	0.903	0.941

Table 2-8
Burnup Averaged Energy of the Average Lethargy Causing Fission (EALF) from the Depletion Analysis of the Benchmark Cases

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.664	0.734	0.796	0.854	0.910	0.965
2	0.891	0.943	0.984	1.017	1.048	1.079
3	0.781	0.839	0.887	0.929	0.970	1.011
4	0.608	0.646	0.678	0.706	0.734	0.765
5	1.084	1.081	1.086	1.103	1.129	1.160
6	1.006	0.987	0.994	1.013	1.038	1.069
7	1.381	1.276	1.226	1.212	1.217	1.235
8	0.876	0.943	1.001	1.052	1.100	1.148
9	0.877	0.951	1.013	1.066	1.116	1.166
10	0.664	0.734	0.796	0.854	0.910	0.965
11	0.891	0.943	0.984	1.017	1.048	1.079

Since the benchmark experiments include geometries, materials, configurations, and energy spectra that are comparable to the fuel assembly and storage rack, even though they are not exact replicas of geometries being analyzed, they provide a good indication of the ability of the code and cross-sections utilized to accurately determine the k_{eff} of the system modelled. Therefore, the comparison

of the EALF between different fuel types across the expected burnup range provides a similar comparison showing the applicability of the depletion benchmarks to other PWR fuel designs not explicitly contained in the depletion benchmarks. Subsequently, it can be concluded that the depletion benchmarks are applicable to the range of PWR fuel types containing UO_2 fuel and Zirconium based cladding.

However, it is recommended that the applicant compare the burnup averaged EALF from their depletion analysis to the benchmarks and if it is outside the range of the benchmarks in a non-conservative direction, add an additional bias determined by extrapolation of the bias as a function of EALF.

Section 3: Comparison of Measured Versus Predicted Reactivity Decrements Using SCALE

Computations were performed using SCALE 6.1.2 with the 238-group ENDF/B-VII cross section library [12]. Computations were also performed using 44 group ENDF/B-V library for comparison purposes. Analyses consist of two steps: depletion analysis and criticality calculation. The depletion model uses SCALE's TRITON t5-depl sequence. The criticality computations were performed using KENO V.

The reactivity values were computed for the 11 EPRI Benchmarks at the cooling times of 100 hrs, 5 yrs, and 15 yrs. Then, the depletion reactivity decrement values for Cases 1-4 and 9-11 are computed as:

$$\Delta k_{depletion} = k_{fresh} - k_{depleted} \qquad \text{(Equation 3-1)}$$

For cases 5 to 8, the depleted reactivity values are compared to the Case 3 fresh fuel reactivity to ensure consistency with the EPRI Benchmark Report [3]. These values are then compared to the values inferred from measured values, denoted as Δk_{EPRI} and tabulated in Tables 2.2 to Table 2.4 for each cooling time. The analysis utilizes the benchmarks as "measured reactivity decrements" and the reactivity decrement uncertainty (i.e., benchmark uncertainty) was reported in Table 2-5. Therefore, any difference between the predictions ($\Delta k_{depletion}$) and the benchmarks (Δk_{EPRI}) is conservatively treated as a bias. Therefore:

$$\text{Bias} = \Delta k_{depletion} - \Delta k_{EPRI} \qquad \text{(Equation 3-2)}$$

If the calculated decrement is smaller than the measured EPRI decrement (i.e., negative difference), the analysis is considered conservative.

After computing bias values for each case and cooling time, the uncertainties and bias in the measured values will also be incorporated (See Section 4).

3.1 Depletion Reactivity Decrement Values Using ENDF/B-VII Cross Section Libraries

In this section, the depletion reactivity decrement values using SCALE with the ENDF/B-VII cross section library are presented.

The difference between calculated and measured reactivity decrement ($\Delta k_{depletion}$ - Δk_{EPRI}) for 100 hr cooling time using SCALE with ENDF/B-VII cross section library is presented in Table 3-1.

Table 3-1
Difference Between Calculated and Measured Reactivity Decrements for EPRI Benchmarks with 100-Hour Cooling Using ENDF/B-VII Cross Section Library

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.0001	-0.0003	-0.0007	-0.0019	-0.0030	-0.0052
2	0.0009	0.0010	0.0006	0.0002	-0.0011	-0.0022
3	0.0010	0.0007	0.0004	-0.0008	-0.0016	-0.0035
4	0.0007	0.0003	-0.0001	-0.0014	-0.0027	-0.0046
5	0.0010	0.0014	0.0010	-0.0002	-0.0018	-0.0029
6	0.0021	0.0015	0.0011	-0.0006	-0.0024	-0.0044
7	0.0020	0.0021	0.0013	-0.0002	-0.0017	-0.0041
8	0.0008	0.0009	0.0004	-0.0005	-0.0017	-0.0034
9	0.0002	0.0003	-0.0002	-0.0008	-0.0020	-0.0034
10	0.0000	-0.0005	-0.0013	-0.0023	-0.0039	-0.0056
11	0.0006	0.0007	0.0001	-0.0006	-0.0019	-0.0033

The difference between calculated and measured reactivity decrement ($\Delta k_{depletion}$ - Δk_{EPRI}) for 5 yr cooling time using SCALE with ENDF/B-VII cross section library is presented in Table 3-2.

Table 3-2
Difference Between Calculated and Measured Reactivity Decrements for EPRI Benchmarks with 5 yr Cooling Using ENDF/B-VII Cross Section Library

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.0007	0.0002	-0.0008	-0.0013	-0.0028	-0.0047
2	0.0011	0.0017	0.0014	0.0007	-0.0006	-0.0021
3	0.0013	0.0010	0.0007	-0.0003	-0.0014	-0.0031
4	0.0012	0.0010	-0.0002	-0.0008	-0.0023	-0.0039
5	0.0014	0.0020	0.0011	0.0003	-0.0012	-0.0029
6	0.0022	0.0018	0.0014	0.0002	-0.0017	-0.0039
7	0.0021	0.0030	0.0022	0.0008	-0.0011	-0.0031
8	0.0012	0.0011	0.0004	0.0003	-0.0013	-0.0024
9	0.0005	0.0008	0.0005	-0.0005	-0.0012	-0.0027
10	0.0001	-0.0001	-0.0010	-0.0019	-0.0034	-0.0052
11	0.0011	0.0007	0.0006	-0.0001	-0.0012	-0.0024

The difference between calculated and measured reactivity decrement ($\Delta k_{depletion}$ - Δk_{EPRI}) for 15 yr cooling time using SCALE with ENDF/B-VII cross section library is presented in Table 3-3.

Table 3-3
Difference Between Calculated and Measured Reactivity Decrements for EPRI Benchmarks with 15 yr Cooling Using ENDF/B-VII Cross Section Library

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.0009	0.0007	-0.0008	-0.0016	-0.0026	-0.0045
2	0.0016	0.0018	0.0015	0.0008	-0.0004	-0.0018
3	0.0014	0.0015	0.0005	-0.0001	-0.0015	-0.0030
4	0.0013	0.0010	0.0000	-0.0010	-0.0025	-0.0041
5	0.0022	0.0019	0.0014	0.0006	-0.0010	-0.0026
6	0.0031	0.0026	0.0019	0.0004	-0.0014	-0.0033
7	0.0027	0.0033	0.0024	0.0009	-0.0005	-0.0029
8	0.0021	0.0014	0.0008	0.0000	-0.0014	-0.0029
9	0.0010	0.0010	0.0005	-0.0005	-0.0014	-0.0028
10	0.0009	0.0001	-0.0008	-0.0022	-0.0034	-0.0050
11	0.0014	0.0016	0.0008	-0.0001	-0.0010	-0.0026

3.2 Depletion Reactivity Decrement Values Using ENDF/B-V Cross Section Libraries

In this section, the depletion reactivity decrement values using SCALE with the ENDF/B-V cross section library are presented.

The difference between calculated and measured reactivity decrement ($\Delta k_{depletion}$ - Δk_{EPRI}) for 100 hr cooling time using SCALE with ENDF/B-V cross section library is presented in Table 3-4.

Table 3-4
Difference Between Calculated and Measured Reactivity Decrements for EPRI Benchmarks with 100-Hour Cooling Using ENDF/B-V Cross Section Library

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.0000	-0.0015	-0.0032	-0.0064	-0.0097	-0.0126
2	0.0013	0.0015	0.0005	-0.0013	-0.0041	-0.0074
3	0.0005	0.0001	-0.0007	-0.0035	-0.0062	-0.0101
4	0.0003	-0.0003	-0.0021	-0.0047	-0.0080	-0.0119
5	-0.0012	-0.0008	-0.0020	-0.0043	-0.0068	-0.0088
6	0.0032	0.0005	-0.0013	-0.0047	-0.0075	-0.0097
7	0.0019	0.0014	-0.0011	-0.0042	-0.0071	-0.0097
8	0.0010	0.0005	-0.0005	-0.0030	-0.0059	-0.0090
9	0.0003	0.0001	-0.0012	-0.0037	-0.0066	-0.0098
10	-0.0009	-0.0020	-0.0040	-0.0066	-0.0100	-0.0135
11	0.0008	0.0015	0.0005	-0.0013	-0.0040	-0.0068

The difference between calculated and measured reactivity decrement ($\Delta k_{depletion}$ - Δk_{EPRI}) for 5 yr cooling time using SCALE with ENDF/B-V cross section library is presented in Table 3-5.

Table 3-5
Difference Between Calculated and Measured Reactivity Decrements for EPRI Benchmarks with 5 yr Cooling Using ENDF/B-V Cross Section Library

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	0.0004	-0.0020	-0.0055	-0.0097	-0.0137	-0.0172
2	0.0011	0.0009	-0.0007	-0.0038	-0.0074	-0.0114
3	0.0011	-0.0002	-0.0023	-0.0061	-0.0103	-0.0145
4	0.0004	-0.0007	-0.0037	-0.0072	-0.0117	-0.0160
5	0.0005	0.0009	-0.0011	-0.0043	-0.0084	-0.0110
6	0.0051	0.0022	-0.0008	-0.0046	-0.0085	-0.0119
7	0.0034	0.0031	0.0000	-0.0043	-0.0082	-0.0116
8	0.0013	0.0000	-0.0020	-0.0053	-0.0092	-0.0133
9	0.0005	-0.0005	-0.0026	-0.0059	-0.0100	-0.0139
10	-0.0005	-0.0019	-0.0043	-0.0081	-0.0122	-0.0157
11	0.0015	0.0009	-0.0007	-0.0034	-0.0068	-0.0108

The difference between calculated and measured reactivity decrement ($\Delta k_{depletion}$ - Δk_{EPRI}) for 15 yr cooling time using SCALE with ENDF/B-V cross section library is presented in Table 3-6.

Table 3-6
Difference Between Calculated and Measured Reactivity Decrements for EPRI Benchmarks with 15 yr Cooling Using ENDF/B-V Cross Section Library

Case	Burnup (GWd/MTU)					
	10	20	30	40	50	60
1	-0.0002	-0.0030	-0.0078	-0.0125	-0.0173	-0.0210
2	0.0015	0.0005	-0.0019	-0.0054	-0.0100	-0.0151
3	0.0011	-0.0010	-0.0039	-0.0083	-0.0135	-0.0182
4	0.0004	-0.0016	-0.0053	-0.0099	-0.0154	-0.0199
5	0.0009	0.0004	-0.0026	-0.0066	-0.0111	-0.0142
6	0.0050	0.0021	-0.0021	-0.0067	-0.0115	-0.0149
7	0.0036	0.0027	-0.0016	-0.0062	-0.0107	-0.0145
8	0.0009	-0.0005	-0.0033	-0.0079	-0.0125	-0.0172
9	0.0007	-0.0011	-0.0040	-0.0082	-0.0128	-0.0174
10	-0.0005	-0.0024	-0.0057	-0.0096	-0.0139	-0.0178
11	0.0016	0.0003	-0.0024	-0.0061	-0.0099	-0.0141

Section 4: Application of EPRI Depletion Benchmarks in Criticality Analyses

The EPRI Depletion Benchmark report (Section 9.4 of Reference 3) provides details on how to apply the experimental reactivity decrements in a spent fuel criticality analysis. Additional detail is provided below to supplement this guidance for how the end-user uses the results from their code validation of the lattice benchmark experiments to their analysis.

The 95/95 tolerance limit of the depletion reactivity decrement (i.e., uncertainty), in units of pcm and a percentage of the depletion reactivity worth, are presented in Table 4-1. As evident from the table, the uncertainty increases as a function of the burnup. Compared to Table 2-5, presented in Section 2.2, this table only includes experimental uncertainties since experimental biases are already included in the values listed in Table 2-2 to 2-4.

Table 4-1
Depletion Reactivity Worth Uncertainty

Burnup (GWd/MTU)	**10**	**20**	**30**	**40**	**50**	**60**
95/95 Tolerance Limit (pcm)	348	537	654	752	831	888
% of Depletion Worth	3.05	2.66	2.33	2.12	1.95	1.81

The steps for validation are described in the EPRI Depletion Benchmark Report [3], with additional guidance for the determination of biases and uncertainties for use in the end-users criticality analysis:

1. Lattice depletions are performed with the user's lattice depletion tool to 60 GWd/MTU at the precise physical conditions specified in the benchmarks.
2. Decay calculations for each cooling interval of interest (e.g., 100-hours, 5-years, and 15-years) are performed with the user's lattice depletion tool from each depletion branch (10, 20, 30, 40, 50, 60 GWd/MTU) of Step 1.
3. Fuel number densities at each depletion/cooling branch from Step 2 are transferred to the user's criticality model of the lattice, and cold k-infinities

are computed for each combination of lattice/burnup/cooling time and lattice conditions.

Note: A criticality analysis methodology may make modeling approximations which involve averaging of fuel pin number densities. In such cases, the averaging must also be performed at this Step of the Benchmark analysis.

4. Construct reactivity decrement tables as a function of lattice type, burnup, and cooling interval from the computed k-infinities.

5. Calculate the difference between the applicants calculated reactivity decrement and measured reactivity decrements (calculated minus measured) for the 11 benchmarks. The experimental reactivity decrement tables are presented in Section 2.2 (Tables 2.2 to 2.4). Using this approach, construct biases for the user's methodology/tools as a function of lattice type, burnup, and cooling interval, per the guidance in Section 3. Then, determine **the maximum positive difference to be applied as an additional bias**, defined as the Application Code Bias.

   Include the Applicant Depletion Code Bias in the overall calculation of maximum k_{eff}.

6. Reactivity decrement uncertainties are applied for each reactivity decrement used in the criticality analysis.

   Evaluate the EPRI depletion uncertainty, from Table 4-1, to be statistically combined with other uncertainties for inclusion in the overall calculation of the maximum k_{eff}.

7. Biases and uncertainties are combined with biases and uncertainties arising from other portions of the SFP/cask criticality analysis.

Section 5: Summary and Conclusions

An important aspect of the burnup credit is the uncertainty associated with the reactivity worth of the isotopic inventory in spent fuel. This uncertainty depends on depletion analysis and the reactivity worth of isotopes not found in the fresh fuel critical experiments. Historically, the 5% decrement approach for depletion uncertainty was used, as set forth in the Kopp memo, to address this gap. EPRI developed a set of lattice benchmarks covering a wide range of enrichments, burnups, cooling times, and burnable absorbers, that can be used to assess the accuracy of the criticality analyst's depletion and criticality code system.

The EPRI benchmarks indicate that the 5% decrement approach is conservative for determining depletion uncertainty in SFP criticality calculations with burnup credit when a user's code-specific bias is small, and demonstrated that for PWRs, additional margins can be applied provided a user's depletion code is properly benchmarked. The EPRI benchmarks were described in detail in Reference 3.

In this companion report, steps for applying the benchmarks are described. For this purpose, computations were performed using SCALE 6.1. Depletion computations were performed using t5-depl sequence of TRITON and criticality calculations were performed using KENO V.

To evaluate the impact of cross section libraries used in the analysis, computations were performed using both ENDF/B-VII and ENDF/B-V cross section libraries. As evident from the results presented in Section 4, the agreement between measured benchmark reactivity decrement values and computed reactivity values using ENDF/B-VII libraries are significantly better, compared to the estimations using ENDF/B-V libraries. Consequently, it is recommended to perform the analysis, both depletion and criticality, using ENDF/B-VII libraries.

Section 6: References

1. L. Kopp, NRC memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998 (ADAMS Accession No. ML003728001).
2. *Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty.* EPRI, Palo Alto, CA: 2011. 1022909.
3. *Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty – Revision 1.* EPRI, Palo Alto, CA: 2017. 3002010613.
4. Lancaster, D. B., K. S. Smith, and A. J. Machiels. "*Depletion Reactivity Benchmarks—II: Utilization in PWR Spent-Fuel Pool Criticality Analysis*," Nuclear Technology, 2014.
5. Kucukboyaci, V., "*EPRI depletion benchmark calculations using PARAGON*," Annals of Nuclear Energy, 2015.
6. Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, ORNL/TM-2005/39, Version 6.1, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

Appendix A: Reactivity Benchmark Specifications

Eleven experimental benchmarks, based on simplifications of publically-available data for the Westinghouse RFA and OFA assemblies, are described here. They cover a range of

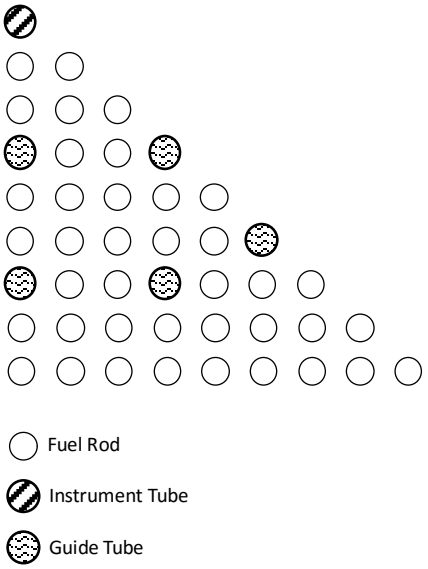
- enrichments
- burnable absorber ladings
- boron concentrations
- fuel and coolant temperatures
- decay times

For each case in Table A-1, a complete geometrical and material description follows.

Note that all lattices are depleted with a power density of 104.5 W/cc (38.1 W/gm heavy metal) – except for case 11 which is depleted at 156.75 W/cc (150% of nominal power density). **All temperatures are in K and nuclide number densities are expressed in atoms/cc**. For lattices that do not explicitly include the "Structural Material Description and the "Coolant Description," the nominal lattice values are applicable.

Table A-1
Benchmark Lattice Cases

1	3.25% Enrichment
2	5.00% Enrichment
3	4.25% Enrichment
4	off-nominal pin diameter depletion
5	20 WABA depletion
6	104 IFBA depletion
7	104 IFBA plus 20 WABA depletion
8	high boron depletion = 1500 ppm
9	branch to hot rack (150°F coolant/fuel) = 338.7K
10	branch to high rack boron = 1500 ppm
11	high power depletion (power, coolant/fuel temp)



Physical Description

Number of pins along side	17
Pin pitch	1.2598 cm
Inter-assembly spacing	21.5036 cm
Fuel pellet OR	0.4096 cm
Clad IR	0.4180 cm
Clad OR	0.4750 cm
Guide/instrument tube IR	0.5610 cm
Guide/instrument tube OR	0.6120 cm

Structural Material Description

Material (Zr-4)Density	6.55 (g/cm3)
Temp., unheated	580K
Temp., heated	$0.12*T_{fuel}+0.88*T_{coolant}$
Nuclide	**Number Density**
Zr-4	4.32444E+22

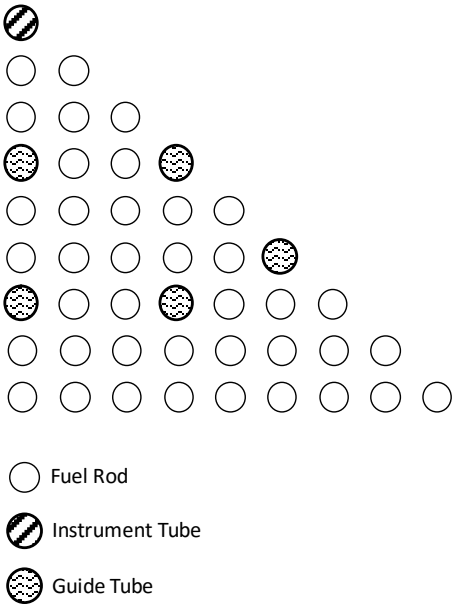
Coolant Description, Depletion (Nominal)

Boron Concentration	900 ppm
Temperature	580 K
Nuclide	**Number Density**
H	4.75756E+22
O	2.37894E+22
B	3.56773E+19

Coolant Description, Cold

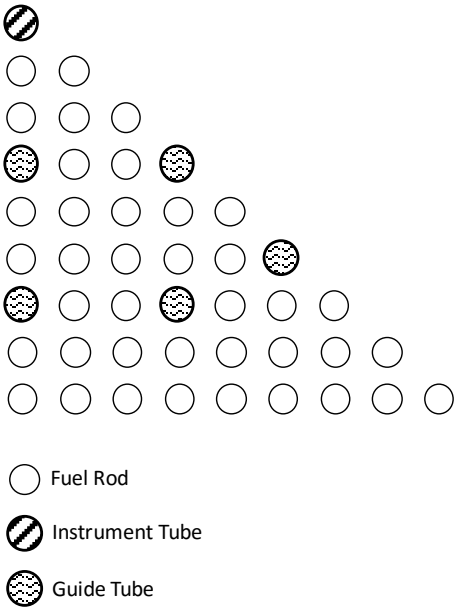
Boron Concentration	0 ppm
Temperature	293 K
Nuclide	**Number Density**
H	6.67431E+22
O	3.33738E+22

Figure A-1
Nominal Fuel Assembly



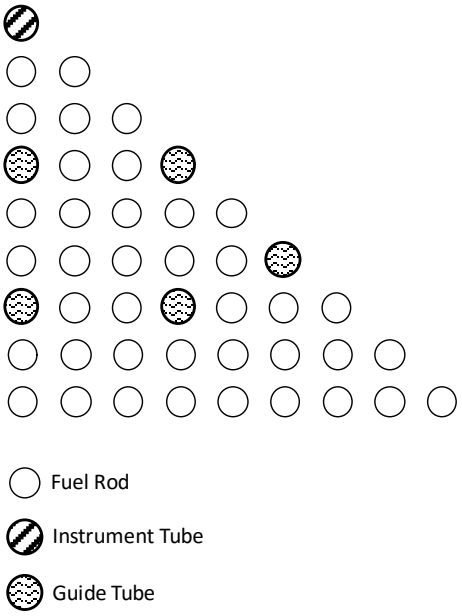
Fuel Material Description	
Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	7.59010E+20
U-234	6.09917E+18
U-238	2.23037E+22
O	4.61377E+22

Figure A-2
CASE 1: 3.25% Enriched - No Burnable Absorbers



Fuel Material Description	
Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	1.16768E+21
U-234	9.38308E+18
U-238	2.18964E+22
O	4.61469E+22

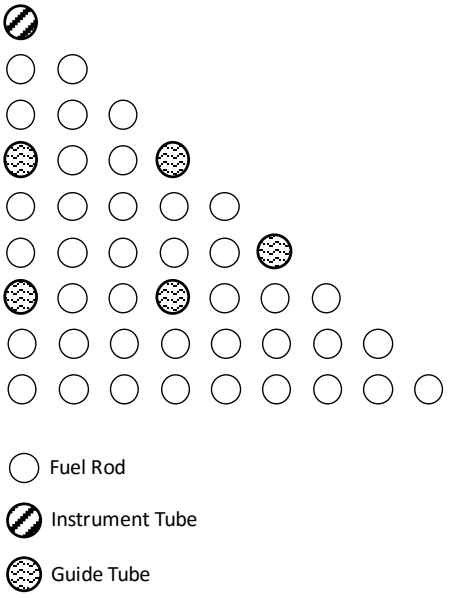
Figure A-3
CASE 2: 5.00% Enriched - No Burnable Absorbers



Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Figure A-4
CASE 3: 4.25% Enriched - No Burnable Absorbers



Physical Description

Number of pins along side	17
Pin pitch	1.2598 cm
Inter-assembly spacing	21.5036 cm
Fuel pellet OR	0.3922 cm
Clad IR	0.4000 cm
Clad OR	0.4572 cm
Guide/instrument tube IR	0.5610 cm
Guide/instrument tube OR	0.6120 cm

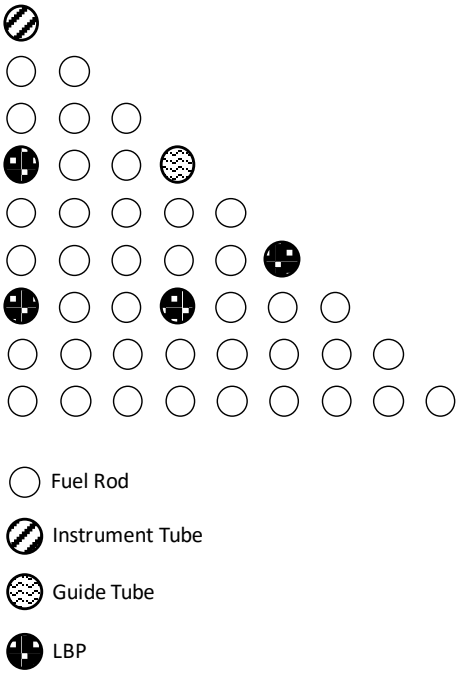
Structural Material Description

Material (Zr-4)Density	6.55 (g/cm3)
Temp., unheated	580K
Temp., heated	$0.12*T_{fuel}+0.88*T_{coolant}$
Nuclide	**Number Density**
Zr-4	4.32444E+22

Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Figure A-5
CASE 4: Small Fuel Pin



Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Lumped Burnable Poison (WABA)

Annular clad IR	0.2860 cm
Annular clad OR	0.3390 cm
Active region IR	0.3530 cm
Active region OR	0.4040 cm
Inner clad IR	0.4180 cm
Inner clad OR	0.4840 cm
Outer clad IR	0.5610 cm
Outer clad OR	0.6120 cm

Active Region

Material Density	3.65 (g/cm3)
Boron Loading	6.03 mg/cm B-10
Nuclide	**Number Density**
C	1.40923E+21
O	6.23784E+22
Al	4.15904E+22
B-10	2.99030E+21

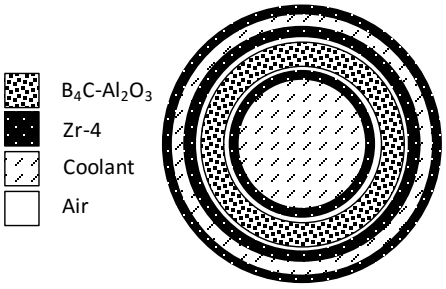
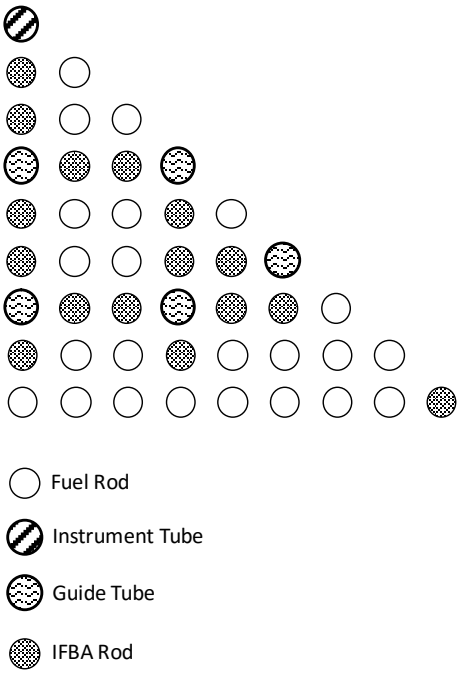
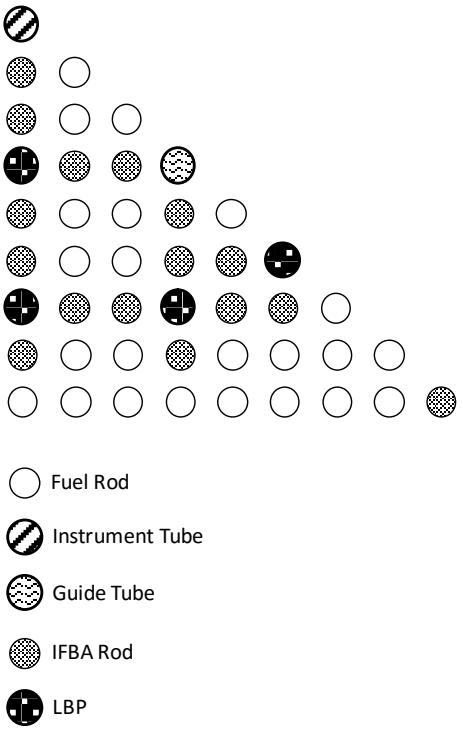


Figure A-6
CASE 5: Lumped Burnable Poison (WABA) Pins



Fuel Material Description	
Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
IFBA Description	
Material Density	6.100 (g/cm3)
Coating Density	0.925 mg/cm B-10
Coating Thickness	0.01 mm
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22
Zr-4	3.22187E+22
B-10	2.15913E+22

Figure A-7
CASE 6: 104 Integral Fuel Burnable Absorbers (IFBA) Pins



Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K

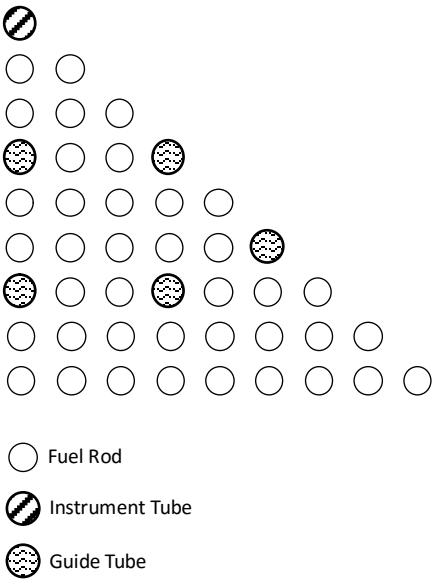
IFBA Description

Material Density	6.100 (g/cm3)
Coating Density	0.925 mg/cm B-10
Coating Thickness	0.01 mm
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22
Zr-4	3.22187E+22
B-10	2.15913E+22

Lumped Burnable Poison (WABA)

Material Density	3.65 (g/cm3)
Boron Loading	6.03 mg/cm B-10
Nuclide	**Number Density**
C	1.40923E+21
O	6.23784E+22
Al	4.15904E+22
B-10	2.99030E+21

Figure A-8
CASE 7: 104 IFBA and 20 WABA Pins



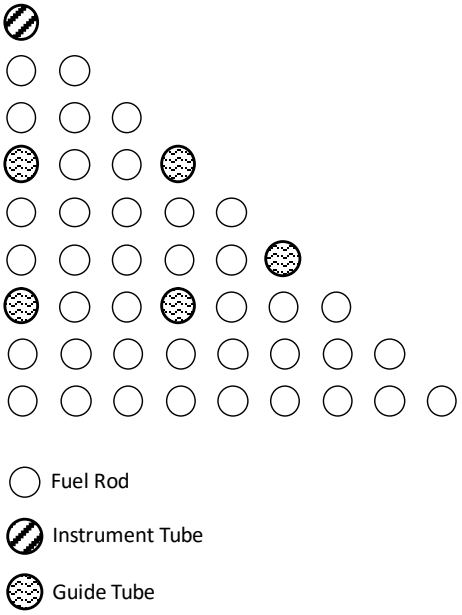
Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Coolant Description, Depletion

Boron Concentration	1500 ppm
Temperature	580 K
Nuclide	**Number Density**
H	4.75756E+22
O	2.37894E+22
B	5.94621E+19

Figure A-9
CASE 8: High Boron Depletion



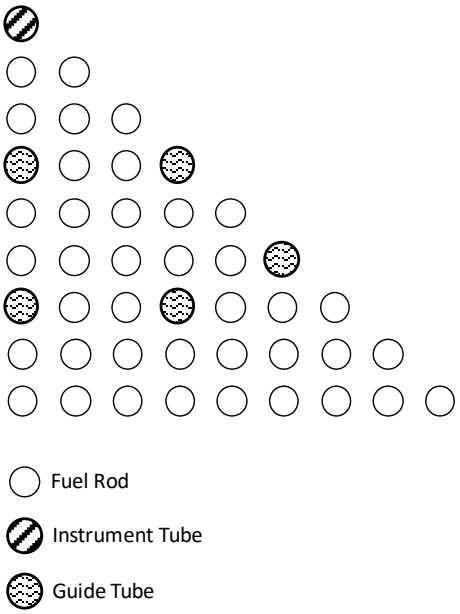
Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Coolant Description, Cold

Boron Concentration	0 ppm
Temperature	338.7 K
Nuclide	**Number Density**
H	6.55262E+22
O	3.27653E+22

Figure A-10
CASE 9: Nominal Case – Branch to Hot Rack Conditions



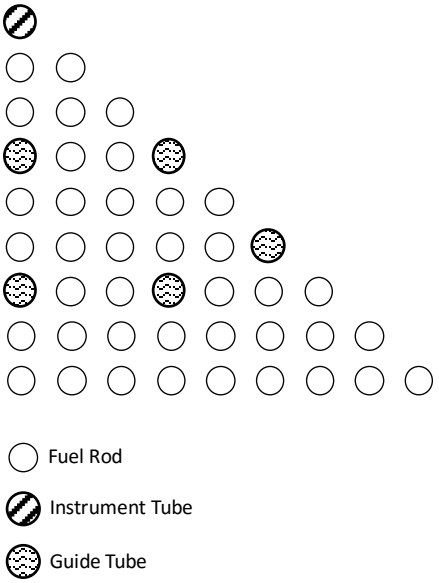
Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Coolant Description, Cold

Boron Concentration	1500 ppm
Temperature	293 K
Nuclide	**Number Density**
H	6.67431E+22
O	3.33738E+22
B	8.34184E+19

Figure A-11
CASE 10: Nominal Case – Branch to High Rack Boron



Fuel Material Description

Material Density	10.340 (g/cm3)
Fuel Temperature	1072.5 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Coolant Description, Depletion

Boron Concentration	900 ppm
Temperature	592.5 K
Nuclide	**Number Density**
H	4.55525E+22
O	2.27778E+22
B	3.41601E+19

Figure A-12
CASE 11: High Power (150% of Nominal) Depletion

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