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Burnup Credit — Contribution to the Analysis of the Yankee Rowe Radiochemical Assays

2011 TECHNICAL REPORT

Burnup Credit – Contribution to the Analysis of the Yankee Rowe Radiochemical Assays

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Report Summary

This report presents a methodology for validation of the isotopic contents of spent light water reactor fuel for actinide-only burnup credit with additional high-quality radiochemistry assay (RCA) data obtained from the Yankee Rowe pressurized water reactor. The additional Yankee Rowe RCA data were not included in previous isotopic validation studies for burnup credit due to the difficulty of accurately modeling the complex Yankee Rowe fuel assembly design using the SAS2H one-dimensional sequence of the earlier SCALE code system. The advent of the two-dimensional TRITON sequence of the SCALE 6.0 code system provided the ability to accurately model the complex Yankee Rowe fuel assembly.

Background

Burnup credit refers to taking credit for the burnup of nuclear fuel in the performance of criticality safety analyses. "Actinide-only" burnup credit refers to a methodology that considers only the two major actinides present in spent fuel: uranium and plutonium. "Fission-product" burnup credit considers a number of fission products and minor actinides. "Full" burnup credit refers to a combination of actinide-only and fission-product burnup credits.

Progress has been made in recent years in the validation of the isotopic compositions and cross sections for the major actinides, using a combination of experimental data and computer-based analyses. In particular, reactivity validation using the French HTC data [NUREG/CR-6979] has solidified the already strong validation of cross sections relevant to actinide-only burnup credit.

Experimental data necessary for validation of the isotopic compositions have been somewhat limited in the past to 50–60 different RCA samples that have been considered to be of acceptable quality. Additional data are being sought from a variety of programs pursued by various laboratories. A source of additional high-quality actinide-only RCAs was the Yankee Rowe reactor. The RCAs from Yankee Rowe included a wide range of flux conditions including asymptotic flux (that is, flat flux) conditions inside the inner portion of the assembly, perturbed conditions adjacent to the control follower blades, and intermediate conditions in the vicinity of the control follower blades but not adjacent to them.

Objectives

- To improve isotopic validation for actinide-only burnup credit, either alone or as a component of full burnup credit

- To provide a larger RCA sample set to improve confidence in the bias and uncertainty associated with isotopic contents

Approach

The TRITON sequence of the SCALE 6.0 code system was used to compute the isotopic contents of the traditional RCA sample set that consisted of RCA data from nine reactors: Calvert Cliffs, H.B. Robinson, Mihama, Obrigheim, Takahama, Trino Vercelles, Turkey Point, Bugey, and Gravelines. Eight Yankee Rowe RCA samples usually included in the RCA dataset used for SAS2H analyses were supplemented by 71 additional samples obtained from regions of stronger neutron flux gradients that required accurate two-dimensional modeling. These 71 Yankee Rowe data points were not appropriate for calculations by the earlier SCALE code system because of the one-dimensional limitations of the SAS2H sequence.

Results

A validation analysis of 79 Yankee Rowe RCAs for the major actinides of actinide-only burnup credit was performed. The bias and uncertainties for actinide-only burnup credit isotopic contents were determined and compared to results using SAS2H. The TRITON isotopic calculations were in much closer agreement than typical SAS2H results. Interestingly, the TRITON predictions were as accurate in the perturbed flux zone as near the control blade followers, indicating that the TRITON sequence was capable of resolving the flux variations for the asymptotic, intermediate, and perturbed flux zones. The bias and uncertainty of the Yankee Rowe dataset were found to be smaller than the bias and uncertainty of the combined nine-reactor dataset. The Yankee Rowe dataset thus contributes to the improvement of the overall isotopic validation dataset. The Yankee Rowe RCA dataset also included 60 ^{137}Cs measurements and 19 measurements of five RCA samples for ^{242}Cm and ^{244}Cm. These measurements were compared with the isotopic contents calculated by TRITON. Differences between measurements and calculated values were within the expected ranges.

Keywords

Bias
Burnup credit
Chemical assays
Isotopic validation
Uncertainty

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Section 1: Introduction

"Burnup credit" refers to taking credit for the burnup of nuclear fuel in the performance of criticality safety analyses. Historically, criticality safety analyses for transport of spent nuclear fuel have assumed the fuel to be unirradiated (i.e., "fresh" fuel). Considerable improvement in cask capacity can be realized if credit for depletion of the fresh fuel can be credited ("burnup credit"). The methodology used to perform criticality analyses using burnup credit, and the validation of the calculations, has not been fully resolved primarily due to validation issues. Recent progress in validation of Actinide-Only cross sections would be complemented by similar progress in validation of Actinide-Only isotopic contents through the addition of radiochemistry assays (RCA) data from the Yankee Rowe reactor fuel.

1.1 Burnup Credit Terminology

"Actinide-Only" burnup credit refers to calculations employing only uranium (^{234}U, ^{235}U, ^{236}U, and ^{238}U) and plutonium (^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) isotopes. Some burnup credit evaluations also include ^{238}Pu, which was not included in this isotopic validation report. "Full" burnup credit refers to a combination of the uranium and plutonium isotopes evaluated in "Actinide-Only" burnup credit, plus a number of fission products and minor actinides. "Fission Product" or "FP" burnup credit is used to refer to a number of fission product isotopes and minor actinide (e.g., neptunium and americium) isotopes.

1.2 Analysis Methodology

The overall methodology used to validate TRITON/MCNP5 for the actinide-only criticality analysis of spent nuclear fuel is illustrated in Fig. 1-1. The analytical methods employed for this evaluation are the TRITON depletion control module [2] of the SCALE 6.0 code system [1] and the Monte Carlo criticality code MCNP5 Version 1.40. [3] The TRITON system, a part of the SCALE code system, can perform two-dimensional fuel depletion and decay calculations.

All TRITON calculations used the 44-group or 238-group neutron cross section library structure and CENTRM to generate isotopic inventories.

Radiochemistry assays (RCA) from spent fuel samples irradiated in nine different pressurized water reactors were selected from References [4], [5], [6], [7], [8], [9] and [10]. Based upon fuel assembly design, power history, and operating data for

these specific pressurized water reactors (PWRs), computational representations were developed for use with TRITON and MCNP5 for the selected spent fuel samples.

Based on operating history only, the TRITON computer code sequence was used to perform a two-dimensional fuel depletion analysis to predict the isotopic concentrations in the sampled fuel pellets. The isotopic concentrations predicted by the depletion module were then used as material input to MCNP5 to generate k_∞ reactivity values; these operating-history-derived reactivity values are denoted as $\mathbf{k_{Calc}}$ in this report. These values were then compared with MCNP5 k_∞ reactivity values derived by using the measured concentrations (RCAs); these RCA-derived reactivity values are denoted as $\mathbf{k_{RCA}}$ in this report. The difference between $\mathbf{k_{Calc}}$ and $\mathbf{k_{RCA}}$ is denoted as $\mathbf{\Delta k_\infty}$. The process is illustrated in Fig. 1-1. This methodology determines an "integral" reactivity isotopic bias and uncertainty rather than an isotope specific bias and uncertainty. This approach complies with the intent of the Burnup Credit Standard, ANSI/ANS-8.27. [11]

TRITON Fuel Depletion

The actinide-only burnup credit calculations were performed with isotopes of uranium and plutonium. Minor actinides and fission products were not included in these calculations. Isotopes of uranium included U^{234}, U^{235}, U^{236}, and U^{238}. In some cases, U^{234} was not provided in the RCA data and was thus also omitted from the calculated isotopic data. Isotopes of plutonium included Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, and Pu^{242}. In some cases, Pu^{238} was not provided in the RCA and was thus also omitted from the calculated isotopic data. Am^{241}, Am^{242m}, Am^{243}, and Np^{237} may be included in actinide-only burnup credit calculations, but were not included in this work because they were not included in the RCAs.

TRITON computer models were developed to represent the two-dimensional cross section of the fuel assembly at the height in the assembly occupied by the sampled fuel pellet(s). TRITON is able to model the material of each fuel rod individually, but at a cost in terms of computer resources and run time. Instead, fuel rods occupying positions in the fuel assembly that have approximately equal neutron flux conditions were represented by one material. The fuel rod containing the sampled fuel pellet is always represented as a separate material. This approach takes advantage of symmetry in the burning of the fuel assembly without compromising the accuracy of the calculated isotopic contents.

MCNP5 Reactivity Calculation

MCNP5 is a general purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues (k_∞) for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells with the ability to describe complex three-dimensional (3-D) geometric structures including spent fuel assemblies and transport/storage containers. The Monte Carlo method is used to duplicate theoretically a statistical process. The individual probabilistic events that comprise a process are simulated sequentially.

The probability distributions governing these events are statistically sampled to describe the total phenomenon.

The MCNP geometry models for each fuel assembly represent the fuel rods, water rod positions (guide tubes and instrument tubes) and reactivity control components (control rods, control blades, or control blade followers). The geometric model of the fuel assembly is represented as surrounded by a reflective boundary condition to simulate the presence of an infinite array of fuel assemblies. Hence the calculations represent k_∞ for the fuel assembly. Validation calculations performed to support licensing submittals may choose to represent the actual fuel cell geometry in which the fuel assembly is located (e.g., a cask basket or spent fuel rack) so that k_{eff} values are obtained instead of k_∞.

MCNP calculates reactivity using collision, absorption, and track-length estimators. The average combined collision, absorption, and track-length estimator from the MCNP5 calculation was used for the value of k_∞ in this report.

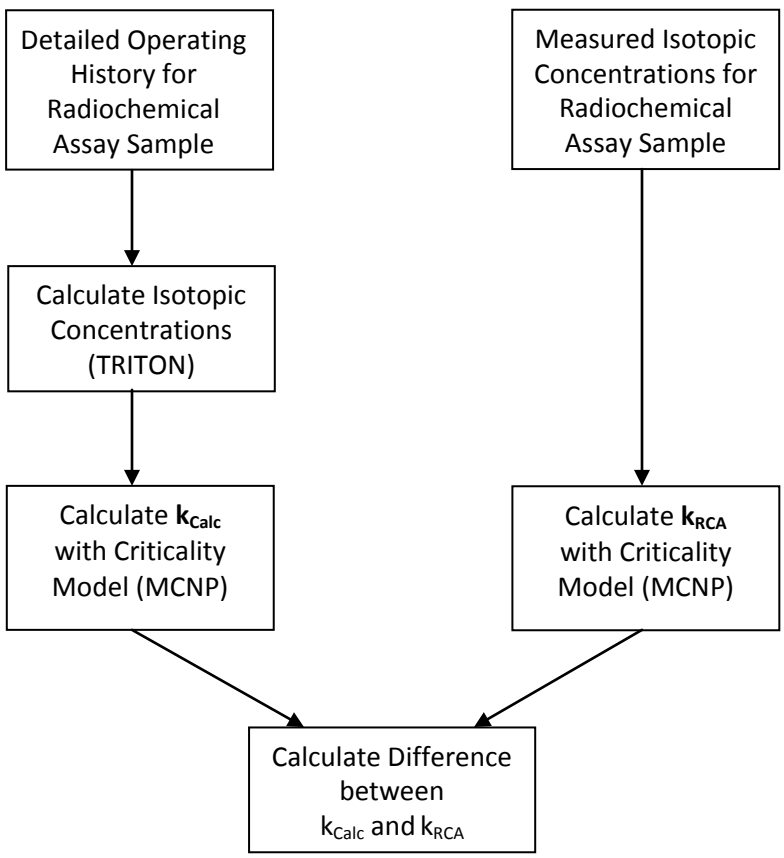

Figure 1-1
Actinide Isotopic Bias Determination

Section 2: RCA Sample Descriptions

Isotopic validation is traditionally achieved through comparisons of reactivity of spent fuel using calculated isotopic contents against reactivity using Radiochemistry Assays (RCAs). Publically available RCAs are used successfully to validate the actinide contents of spent fuel, but the number of samples contained in the typical validation dataset is 60 samples or less. Additional high-quality data, with confirmation of isotopic contents provided by comparison of radiochemical results from two or more independent laboratories, would be useful to validation for actinide-only burnup credit applications.

Many of the spent fuel data used in this report are taken from numerous Oak Ridge National Laboratory (ORNL) and Department of Energy (DOE) reports concerning isotopic validation. [4][5][6][7] These reports have been extensively peer reviewed and accepted by USNRC and other international regulatory bodies including the International Atomic Energy Agency (IAEA) for the validation of burnup credit methodologies. Summaries of the data can also be found at the Nuclear Energy Agency (NEA) website http://www.nea.fr/sfcompo/ as well as in the Japan Atomic Energy Research Institute (JAERI) database. [8] Much of the descriptive text and tables for seven reactors included in this chapter was adapted from the work of J.S. Scaglione documented in Ref. [9]. RCA data for the Bugey and Gravelines (including the high burnup 60 GWd/MTU samples) were obtained from EPRI [10]. Additional references are provided for the Yankee Rowe reactor design and for the Yankee Rowe RCA data in Chapter 3.

2.1 General Material Descriptions for TRITON and MCNP Inputs

The principal materials used in the TRITON isotopic evaluations are: UO_2 (fuel), Zircaloy-4 or steel (cladding) and H_2O (moderator). These compositions are specified on the TRITON input card images and contain basic information, i.e., alphanumeric name, id, density, isotopic distribution and temperature.

The reactor-specific UO_2 density, uranium enrichment and moderator density and associated soluble boron are provided in subsequent sections. The majority of the fuel assemblies evaluated had Zircaloy-4 cladding, except for Trino Vercelles, which used stainless steel 304 cladding. The Yankee Rowe fuel cladding was Type 348 stainless steel and the control blade followers were made of Zircaloy-2. [12]

2.2 Reactor and Radiochemical Assay Data

Radiochemical assay data currently available consist of 59 PWR RCA samples from nine PWR fuel assembly designs. A summary of the sample characteristics for each of the different assembly designs is presented in Table 2-1.

Table 2-1
PWR Radiochemical Assay Information

Reactor	Assembly Design	# of Samples/ Assemblies/ Rods	Sample Burnup (GWd/MTU)	Initial Enrichments (wt% U-235)
Trino Vercelles	Westinghouse, Irregular	14/3/6	12.042	3.897
			11.529-24.548	3.13
Turkey Point	W 15x15, 20 Guide Tubes	5/2/5	30.72-31.56	2.556
Mihama	W 15x15, 20 Guide Tubes	9/3/NA	6.92-8.3	3.208
			14.66-21.29	3.203
			29.5-34.32	3.210
Takahama	Westinghouse 17x17 (Mitsubishi)	11/2/2	14.30-47.25	4.11
H.B. Robinson	W 15x15, 20 Guide Tubes, 12 Burnable Poisons	4/1/1	16.02-31.66	2.561
Obrigheim	Siemens 14x14	6/5/special	25.93-29.52	3.13
Calvert Cliffs	CE 14x14 Burnable Poisons present	6/3/3	27.35-44.34	3.038
			18.68-33.17	2.72
			31.40-46.46	2.453
Bugey	Standard 17x17	1/1/1	38.67	3.10
Gravelines-3	Standard 17x17	3/2/3	38.24-61.08	4.5

Note that the RCA samples for typical PWR fuel were obtained from assemblies that did not contain reactivity control rods inserted to the level of the sample, so that the RCA sample neutron spectrum was not perturbed by the control components. The Yankee Rowe and Trino Vercelles reactors used control blades, but these were similarly not inserted to the level of the RCA samples. Notations for the Yankee Rowe RCAs indicating a "perturbed" spectrum refer to the presence of the control blade followers and the thin layer of water that provides clearance for the blade movement. The pellet and cladding diameters and fuel rod pitch of the Yankee Rowe fuel are equivalent to those of a Westinghouse 15x15 fuel assembly.

2.2.1 Trino Vercelles Geometry and Fuel Material Description

Radiochemical analyses for 14 spent fuel samples from three different assemblies from the Trino Vercelles [4] core were evaluated in this report. It should be noted that the Trino Vercelles assemblies were non-typical PWR fuel assemblies. The core design used two different types of fuel assemblies. The first was a Westinghouse 15x15 design that has 16 outer perimeter fuel cells removed so that the second, cruciform shaped assembly design could fit in-between the square-shaped assemblies. A cross-sectional view of the square and cruciform assemblies as they fit together is illustrated in Fig. 2-1. The Trino Vercelles cladding was made up of stainless steel Type 304 with material composition specifications as the default standard composition used by SCALE.

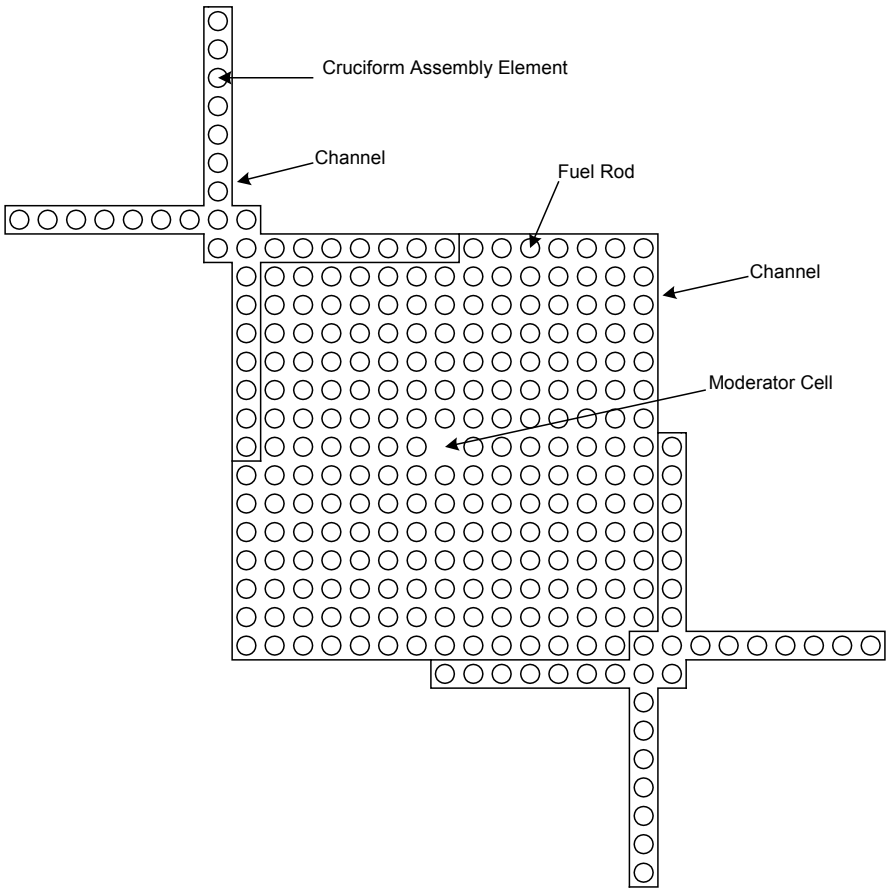

Figure 2-1
Cross-Sectional View of Trino Vercelles Fuel Assemblies

2.2.2 Turkey Point Geometry and Fuel Material Description

Radiochemical analyses for five spent fuel samples from two different assemblies from the Turkey Point Unit 3 core [4] were evaluated in this report. The assembly design is a typical Westinghouse 15x15 design with 20 guide tube

positions and one instrument tube position. A cross-sectional view of the assembly is illustrated in Fig. 2-2.

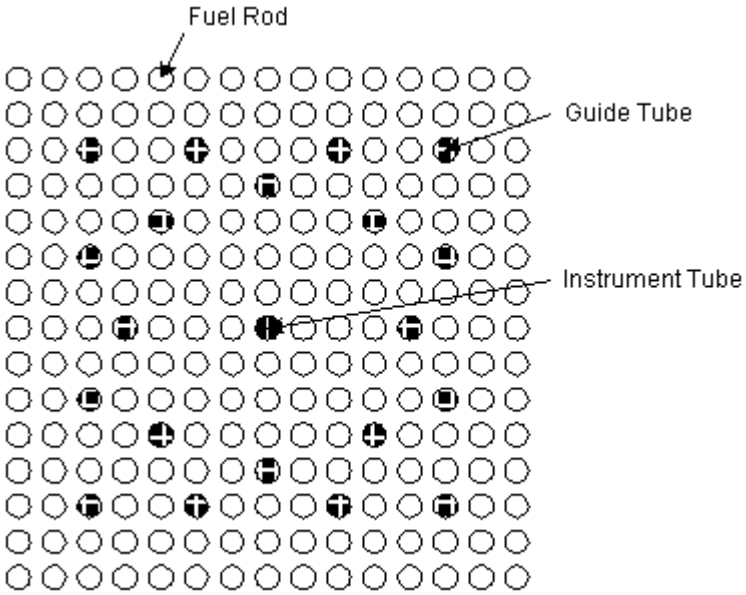

Figure 2-2
Cross-Sectional View of Turkey Point Fuel Assembly

2.2.3 Mihama Geometry and Fuel Material Description

Radiochemical analyses for nine spent fuel samples from three different assemblies from the Mihama [5] core were evaluated in this report. The assembly design is a typical Westinghouse 15x15 design with 20 guide tube positions and one instrument tube position. This assembly design is identical to the Turkey Point fuel assembly design, and a cross-sectional view for this assembly design is illustrated in Fig. 2-2.

2.2.4 Takahama Geometry and Fuel Material Description

Takahama-3 [6] is a PWR that operates with a 17×17 fuel assembly design. Spent fuel samples were obtained from assemblies operated for 2 and 3 cycles and achieved a maximum burnup of 47 GWd/MTU. Radiochemical analyses were performed on two rods having an initial enrichment of 4.11 wt % and one integral burnable absorber rod containing Gd_2O_3. The Gd_2O_3 rod is not included in this isotopic validation since insufficient data are available to validate UO_2/Gd_2O_3 fuel integral burnable absorber fuel rods. A cross section view of the assembly is shown in Figure 2-3.

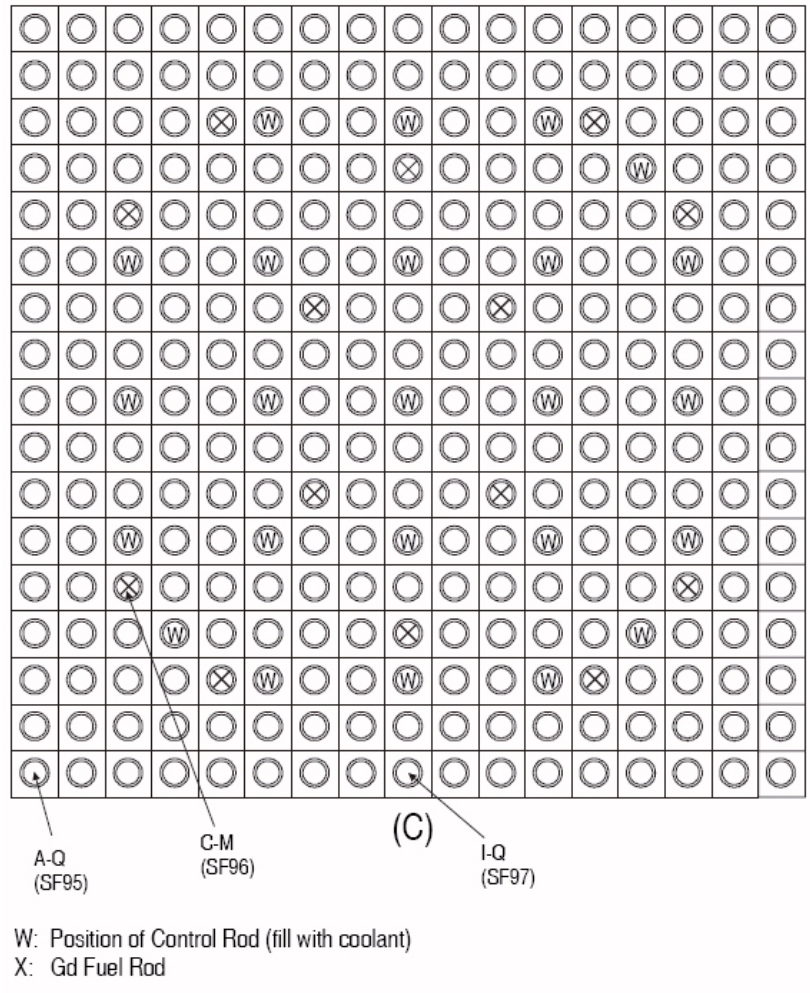

Figure 2-3
Cross-Sectional View of Takahama Fuel Assembly

2.2.5 H.B. Robinson Geometry and Fuel Material Description

Radiochemical analyses for four spent fuel samples from one rod from the H.B. Robinson [7] core were evaluated in this report. The assembly design is a typical Westinghouse 15x15 design with 20 guide tube positions and one instrument tube position. This assembly design is identical to the Turkey Point and Mihama fuel assembly design, with the exception that it was operated with burnable poison rods (BPRs) inserted in 12 guide tube locations. A cross-sectional view for this assembly design is illustrated in Fig. 2-4. The H.B. Robinson fuel assembly contained borosilicate glass burnable poison rods inserted into some of its guide tube locations during irradiation.

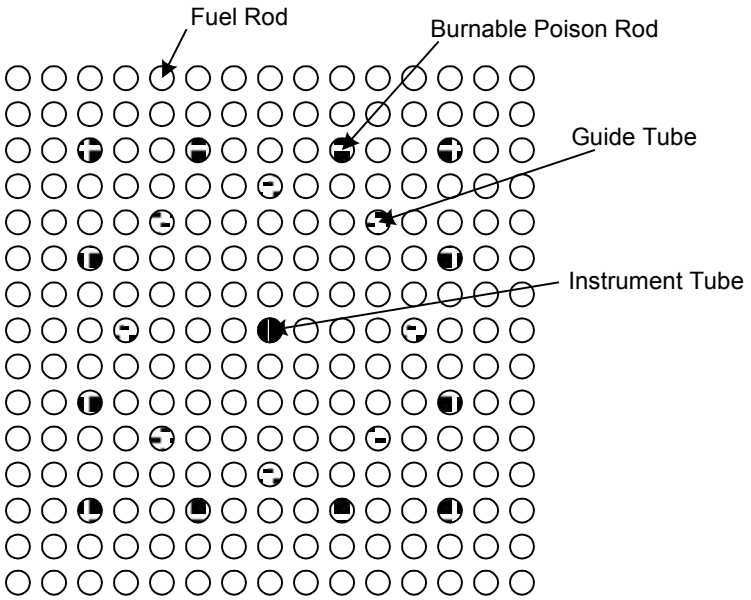

Figure 2-4
Cross-Sectional View of H.B. Robinson Fuel Assembly

2.2.6 Obrigheim Geometry and Fuel Material Description

Radiochemical analyses for six spent fuel samples from five different assemblies from the Obrigheim [7] core were evaluated in this report. The assembly design is a Siemens 14x14 design with 16 guide tube positions. The Obrigheim sample data are different from the other samples evaluated in this report. The fuel assemblies were taken from this reactor and split into two halves radially, with each half being dissolved separately from which the sample measurements were made [5]. The other samples evaluated in this report typically come from a small section of a single fuel rod within an assembly. A cross-sectional view for this assembly design is illustrated in Fig. 2-5.

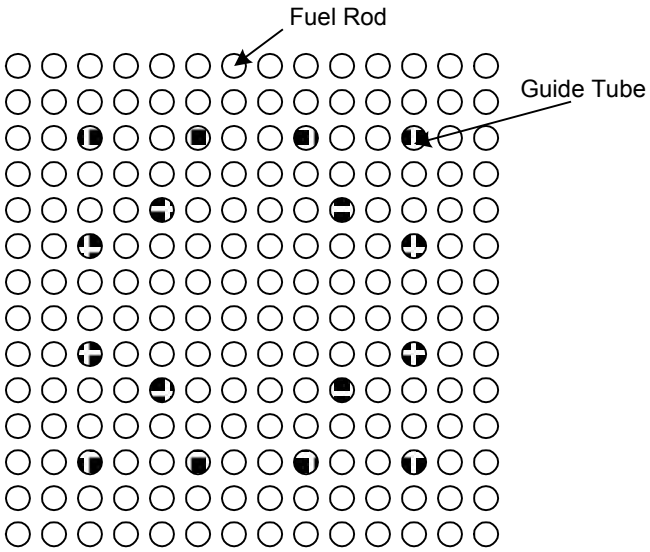

Figure 2-5
Cross-Sectional View of Obrigheim Fuel Assembly

2.2.7 Calvert Cliffs Geometry and Fuel Material Description

Radiochemical analyses for six spent fuel samples from two different assemblies from the Calvert Cliffs [7] core were evaluated in this report. The assembly design is a Combustion Engineering (CE) 14x14 design with five guide tube positions. It should be noted that the guide tubes are large (~4 unit cells) for this assembly design. Assemblies D047 and D101 were standard assemblies and assembly BT03 was non-standard. Assembly BT03 contained 12 burnable absorber rods that replaced fuel rods and four test rods that contained no fuel in them. The BT03 assembly contained Gd_2O_3 rods and is not relevant to this isotopic validation since insufficient data are available to validate UO_2/ Gd_2O_3 fuel integral burnable absorber fuel rods. Three assay samples at different axial heights were taken from a single rod from each of the assemblies. A cross-sectional view for the CE 14x14 assembly design for Assemblies D047 and D101 is illustrated in Figure 2-6.

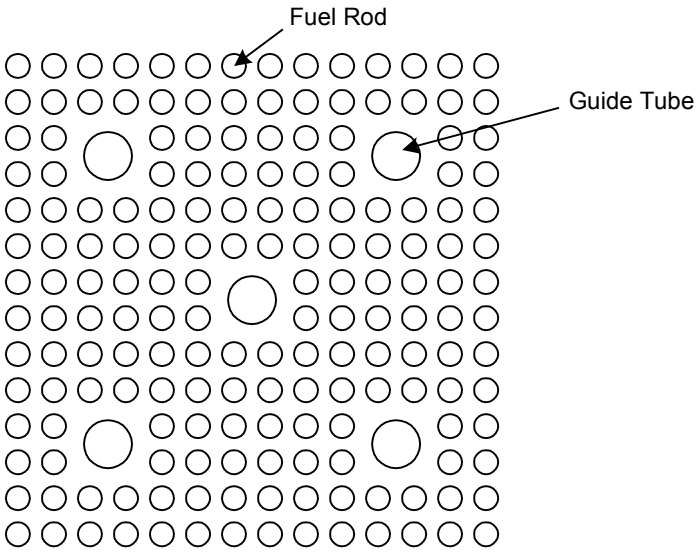

Figure 2-6
Cross-Sectional View of Calvert Cliffs Fuel Assemblies D047 and D101

2.2.8 Bugey and Gravelines-3 Geometry and Fuel Material Description

Radiochemical analyses for one spent fuel sample from Bugey and three fuel samples from Gravelines-3 were provided by EPRI [10]. The assembly design for these two plants is a Standard 17x17 design with 24 guide tube positions and one central instrument thimble. Fuel assembly information is provided in Table 2-2.

Table 2-2
Standard 17x17 Fuel Assembly Parameters

Number of Rods:	264
Number of Guide Tubes:	24
Number of Instrument Tubes:	1
Total Positions:	289
Active Fuel Length:	144 inches
Rod Pitch:	1.2621 cm
Effective Cell Radius:	0.7121 cm
Fuel Rod Dimensions	
Fuel Rod OD:	0.374 inches
Pellet OD:	0.3225 inches
Cladding Thickness:	0.0225 inches
Diametral Gap:	0.0065 inches
Fuel Rod Volume:	195.71 cm^3
UO_2 Density:	10.07 g/cm^3
Fuel Mass:	458.6 kgU
Guide Tube Dimensions	
Guide Tube OD:	0.482 inches
Tube Thickness:	0.016 inches
Instrument Thimble Dimensions	
Thimble OD:	0.545 inches
Thimble Thickness:	0.015 inches

The appearance of the fuel assembly cross-section is similar to Figure 2-3.

The RCA samples had the enrichment and burnup characteristics shown in Table 2-3.

Table 2-3
Bugey and Gravelines RCA Samples

Reactor	Assembly	Enrichment	Burnup
Bugey	K11	3.10	38.67
Gravelines	G07	4.5	38.24
Gravelines	G11	4.5	51.03
Gravelines	J07	4.5	61.08

The RCA samples were taken 1.9 meters from the bottom of the fuel rods, which have an active fuel length of 144" or 3.66 meters.

The high burnup of the G11 and J07 RCAs are of particular interest, since they extend the upper range of the isotopic validation.

2.3 Nine-Reactor RCA Dataset

Table 2-2 tabulates the results for the nine reactors/59 RCAs dataset used in the isotopic validation. The bias and uncertainty terms for the "Nine-Reactor" dataset were computed so that they could be compared to the bias and uncertainty terms for the Yankee Rowe/79 RCA dataset. Note that analyses of possible trends of burnup and enrichment contained in Chapter 5 showed no correlation with the "Nine-Reactor" Δk/k values.

Table 2-4
Calculated vs. Measured Results for the "Nine-Reactor" Dataset

Reactor	Sample	k_{Calc}	k_{RCA}	Δk_∞	σ	$\Delta k_\infty / k_{RCA}$
Calvert Cliffs	1	1.18065	1.17446	0.00619	0.00070	0.00527
	2	1.11195	1.10705	0.00490	0.00070	0.00443
	3	1.08761	1.06617	0.02144	0.00069	0.02011
	4	1.21078	1.20426	0.00652	0.00068	0.00541
	5	1.15426	1.14359	0.01067	0.00070	0.00933
	6	1.11847	1.09130	0.02717	0.00070	0.02490
Mihama	1	1.35105	1.34747	0.00358	0.00066	0.00266
	2	1.36150	1.36019	0.00131	0.00064	0.00096
	3	1.24550	1.25689	-0.01139	0.00070	-0.00906
	4	1.29500	1.29915	-0.00415	0.00068	-0.00319
	5	1.29996	1.29990	0.00006	0.00068	0.00005
	6	1.20399	1.19337	0.01062	0.00067	0.00890
	7	1.18926	1.18292	0.00634	0.00069	0.00536
	8	1.15469	1.14884	0.00585	0.00069	0.00509
	9	1.15098	1.16233	-0.01135	0.00069	-0.00976

Table 2-4 (continued)
Calculated vs. Measured Results for the "Nine-Reactor" Dataset

Reactor	Sample	k_{Calc}	k_{RCA}	Δk_∞	σ	$\Delta k_\infty / k_{RCA}$
H B Robinson	1	1.23215	1.21842	0.01373	0.00064	0.01127
	2	1.17432	1.15086	0.02346	0.00062	0.02038
	3	1.16042	1.14483	0.01559	0.00063	0.01362
	4	1.12320	1.10055	0.02265	0.00062	0.02058
Obrigheim (Assembly Average)	1	1.21664	1.20246	0.01418	0.00069	0.01179
	2	1.21182	1.19461	0.01721	0.00067	0.01441
	3	1.20243	1.18889	0.01354	0.00066	0.01139
	4	1.20048	1.18699	0.01349	0.00068	0.01136
	5	1.19702	1.18562	0.01140	0.00067	0.00962
	6	1.19420	1.17804	0.01616	0.00066	0.01372
Trino Vercelles	1	1.35380	1.35354	0.00026	0.00072	0.00019
	2	1.28996	1.28557	0.00439	0.00074	0.00341
	3	1.28676	1.27831	0.00845	0.00074	0.00661
	4	1.30944	1.30704	0.00240	0.00072	0.00184
	5	1.30280	1.30150	0.00130	0.00071	0.00100
	6	1.26560	1.25816	0.00744	0.00071	0.00591
	7	1.25203	1.23748	0.01455	0.00075	0.01176
	8	1.24655	1.23672	0.00983	0.00073	0.00795
	9	1.25151	1.24277	0.00874	0.00076	0.00703
	10	1.24698	1.23597	0.01101	0.00073	0.00891
	11	1.25139	1.24241	0.00898	0.00075	0.00723
	12	1.24645	1.23614	0.01031	0.00071	0.00834
	13	1.25065	1.24560	0.00505	0.00074	0.00405
	14	1.24780	1.23751	0.01029	0.00074	0.00832
Turkey Point	1	1.12525	1.11718	0.00807	0.00070	0.00722
	2	1.12594	1.11267	0.01327	0.00068	0.01193
	3	1.11732	1.11692	0.00040	0.00069	0.00036
	4	1.12269	1.11663	0.00606	0.00068	0.00543
	5	1.12229	1.11309	0.00920	0.00070	0.00827

Table 2-4 (continued)
Calculated vs. Measured Results for the "Nine-Reactor" Dataset

Reactor	Sample	k_{Calc}	k_{RCA}	Δk_∞	σ	$\Delta k_\infty / k_{RCA}$
Takahama	SF95-1	1.36270	1.35647	0.00623	0.00071	0.00459
	SF95-2	1.30940	1.30308	0.00632	0.00074	0.00485
	SF95-3	1.26266	1.25076	0.01190	0.00075	0.00951
	SF95-4	1.25224	1.23893	0.01331	0.00074	0.01074
	SF95-5	1.27557	1.26581	0.00976	0.00072	0.00771
	SF97-1	1.33207	1.32553	0.00654	0.00074	0.00493
	SF97-2	1.26147	1.26597	-0.00450	0.00072	-0.00355
	SF97-3	1.20815	1.21776	-0.00961	0.00075	-0.00789
	SF97-4	1.18108	1.18545	-0.00437	0.00074	-0.00369
	SF97-5	1.17372	1.18158	-0.00786	0.00071	-0.00665
	SF97-6	1.20061	1.20676	-0.00615	0.00073	-0.00510
Bugey	K11	1.08100	1.07841	0.00259	0.00024	0.00240
Gravelines	G07	1.22485	1.22308	0.00177	0.00025	0.00145
	G11	1.16848	1.16260	0.00588	0.00025	0.00506
	J07	1.08725	1.07678	0.01047	0.00024	0.00972

Section 3: Yankee Rowe RCA Sample Data

The Yankee Rowe fuel assembly was not a typical modern PWR assembly because control blades were used for reactor reactivity control instead of control rods. The control blades themselves were not inserted to the levels of the RCA samples, but even the Zircaloy-2 control blade followers and the water layer in the clearance gap caused reactivity gradients that are difficult to account for by the SAS2H computer code sequence of the SCALE 5 system; as a consequence, only eight RCA samples were used in previous SAS2H validations. These eight samples are from the central portion of the fuel assembly, termed the "asymptotic" region in Yankee Rowe terminology.

The TRITON computer code sequence of SCALE 6.0 employs the NEWT two-dimensional neutron transport code instead of the older XSDRNPM one-dimensional code used by SAS2H. Thus, TRITON has the capability to model the complex Yankee Rowe fuel assembly and provide accurate isotopic calculations.

The 59 samples from the "Nine-Reactor" dataset analyzed in this report provide a statistically valid dataset and are adequate for the determination of the isotopic bias and uncertainty of the TRITON code and the methodology used in the TRITON/MCNP5 validation. However, the addition of the 79 RCA samples available from Yankee Rowe can provide improved confidence in the isotopic validation. These 79 RCA samples were obtained and analyzed as part of a consistent program of fuel isotopic analysis, unlike the samples from other reactors, which were obtained from individual programs with assays performed at a number of different facilities. The consistency and repeatability of isotopic measurements is visible in the assay data from the Yankee Rowe program. Isotopic measurements were performed at the New Brunswick Laboratory (NBL) on the site of Argonne National Laboratory (ANL), at the Vallecitos Nuclear Center owned by GE-Hitachi Nuclear Energy and located near Livermore, CA, and by TRACERLAB, a corporation established in Massachusetts and California to perform nuclear chemical assays. A number of RCA samples were tested at two or three of the laboratories, and the cross-checking provided by the use of three different laboratories significantly improved the quality of the Yankee Rowe RCA data. A fourth laboratory at Westinghouse (Waltz Mill) was used to perform gamma measurements but was not used to obtain isotopic measurements for the samples presented in this report. Note that the Yankee Rowe fuel analysis program included RCAs that are not included here because they were at the upper end of the fuel assembly

alongside a control blade for all or part of the irradiation period. Such samples would not be representative of commercial PWR spent fuel in the United States.

The researchers who performed many of the SAS2H validation efforts over the years were aware of the additional Yankee Rowe data, but these were not deemed useful for SAS2H validation because of the inability of a one-dimensional calculational sequence to accurately represent the neutron flux gradients in the Yankee Rowe fuel assemblies. The newer two-dimensional TRITON computer code sequence allows the unused Yankee Rowe RCA data to be re-introduced into isotopic validation studies.

3.1 Yankee Rowe Fuel Design

Post-irradiation examinations (PIEs), including radiochemical analyses (RCAs), were performed on a selected set of spent fuel assemblies from Yankee Rowe Cores I, II, and IV. There were 78 samples evaluated from three different assemblies. One assembly, E6 (59 samples), was irradiated for all three of these cycles and the fuel samples had burnup up to 45 GWd/MTU. One assembly, F5 (10 samples), was in the core for the first two cycles and had burnup up to 30 GWd/MTU. Two assemblies, E5 (5 samples) and F4 (5 samples), were only used in the first core and had burnup up to 15 GWd/MTU.

It should be noted that the Yankee Rowe assemblies were non-typical PWR fuel assemblies. They were a Westinghouse design arranged in a 17x18 configuration with control blades. A cross-sectional view of the assembly design is illustrated in Figure 3-1. However, the fuel pellet and rod diameters and fuel pitch were typical of Westinghouse 15x15 fuel assemblies.

The Yankee Rowe fuel assembly is broken into nine subassemblies, identified by the center (C) and eight points of the compass (N, S, E, W, and NW, NE, SE, and SW). The central subassembly is termed the "Asymptotic" flux region, and the first two rows of fuel rods adjacent to the control blade followers are termed the "Perturbed" flux region. Rods near the "Perturbed" region are called the "Intermediate" flux region.

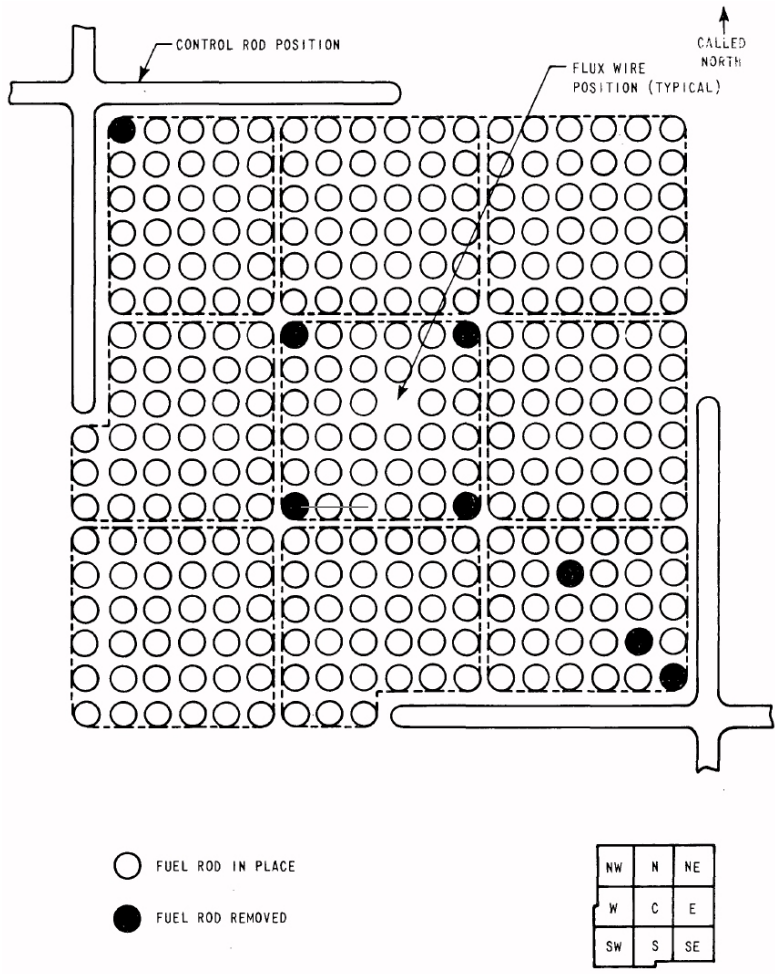

Figure 3-1
Cross-Sectional View of Yankee Rowe Assembly E6 (Note: Fuel rods removed for radiochemistry assays are marked as solid.)

The arrangement of fuel assemblies and control blades/control blade followers is shown in Figure 3-2. The symmetry of the fuel and blades is best viewed on a 45-degree diagonal across the core, unlike most PWRs which exhibit quarter-symmetry.

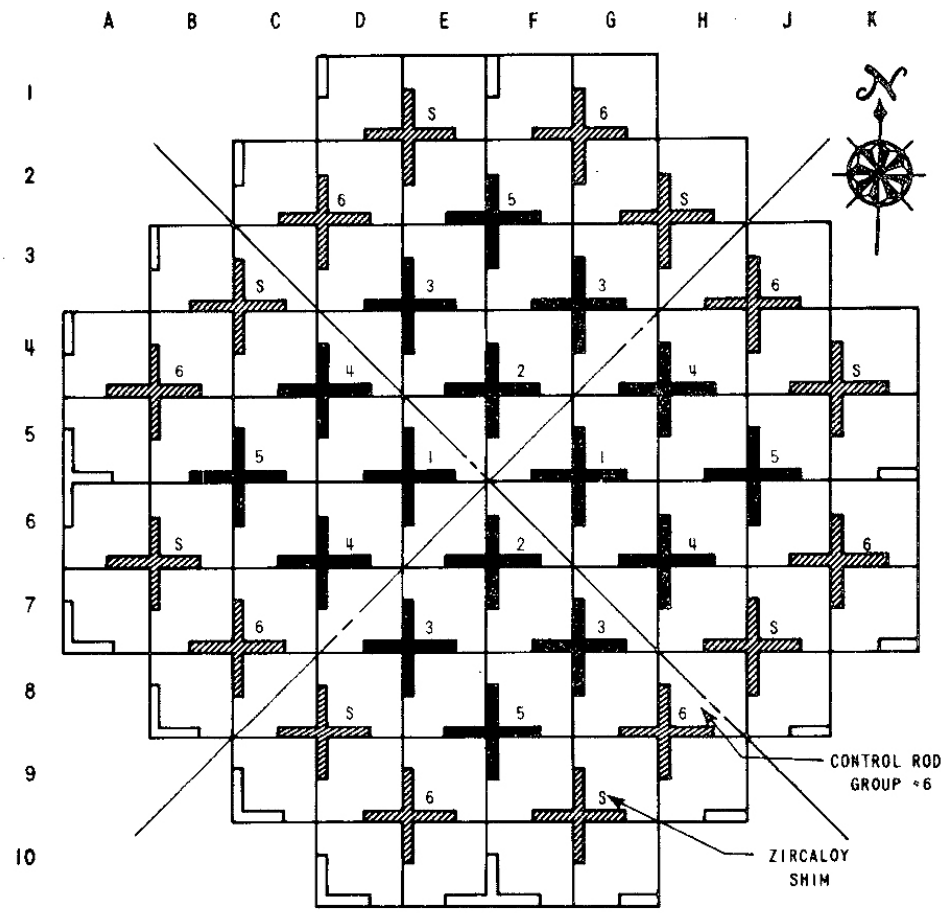

Figure 3-2
Cross-Sectional View of Yankee Rowe Core

The control blades were inserted into the fuel assemblies in slots created by the removal of fuel rods, and it can be seen that there were two types of fuel assemblies, one with two groups of fuel rods removed adjacent to each other and one with two groups of fuel rods removed at opposite ends of the fuel cross section. Zircaloy shim blades surrounded the core exterior. The shim blades consisted of borated stainless steel cruciform blades at one end and Zircaloy cruciform blades at the other end, so that adjustments to the reactivity of the core could be made at the periphery of the core. The use of the borated steel sections does not appear to have been necessary, so the additional reactivity control available via the shims was not used in Cores I through IV.

Table 3-1 presents the general characteristics for the Yankee Rowe samples [13], and Table 3-2 presents the measured isotopic concentrations for the eight samples from Assembly E6, which have been used in SAS2H validation studies for many years. Table 3-3 provides the assembly design parameters.

Table 3-1
Yankee Rowe Sample Characteristics

Assembly & Rod Number	Axial Location from Bottom of Assembly (cm)	Initial Enrichment (wt% U-235)	Burnup (GWd/MTU)	Cooling Time (days)
E6-C-f6	220.22	3.400	15.95	281.5
E6-C-f6	138.94	3.400	30.39	717.0
E6-C-f6	57.66	3.400	31.33	281.5
E6-C-f6	17.02	3.400	20.19	281.5
E6-SE-c2	138.94	3.400	32.03	281.5
E6-SE-c2	57.66	3.400	31.41	281.5
E6-SE-e4	138.94	3.400	35.97	281.5
E6-SE-e4	57.66	3.400	35.26	281.5

Table 3-2
Yankee Rowe Measured Isotopic Concentrations (g/MTU)

Sample	E6-C-f6	E6-C-f6	E6-C-f6	E6-C-f6	E6-SE-c2	E6-SE-c2	E6-SE-e4	E6-SE-e4
Location (cm)	**220.22**	**138.94**	**57.66**	**17.02**	**138.94**	**57.66**	**138.94**	**57.66**
Burnup (GWd/ MTU)	**15.95**	**30.39**	**31.33**	**20.19**	**32.03**	**31.41**	**35.97**	**35.26**
U-234	155	142	119	144	114	115	118	120
U-235	19800	12600	11900	17200	11800	11900	9780	9840
U-236	2880	4080	4150	3300	4180	4090	4450	4440
U-238	949000	937000	935000	946000[1]	935000	936000	933000	934000
Pu-238	47.3	176	214	79.1	222	211	247	240
Pu-239	5950	7870	8010	6600	7980	7680	6950	6820
Pu-240	1120	2120	2260	1440	2370	2270	2570	2480
Pu-241	663	1540	1640	915	1670	1580	1680	1620
Pu-242	80.3	346	398	145	422	400	552	529

[1] This value was shown as 936000 in Table 13 of "Calculation of Isotopic Bias and Uncertainty for PWR SNF", Scaglione, JM, CAL-DSU-NU-000001 REV A, Yucca Mountain Project, 2002, as taken from p. 2-10 of U.S. Department of Energy Report "Isotopic and Criticality Validation for PWR Actinide-Only Burnup Credit", DOE/RW-0497. Washington, D.C.: DOE Office of Civilian Radioactive Waste Management.

Table 3-3
Yankee Rowe Assembly Design Parameters

Parameter	Value	Reference Information
Uranium fuel loading (kgU)	275.557[a]	
Total number of fuel assemblies in core	76	Figure 2.1, Nodvik [13]
Number of unit cells	306	
Number of fuel rods	305	
Number of instrument cells	1	
Assembly pitch (cm)	19.46	
Type of fuel pellet	UO_2	
Pellet density (g/cm^3)	10.18	Table 2.1, Nodvik [13]
Rod pitch (cm)	1.072 (0.422 in) and 1.158 (0.456 in)[b]	Page 9, Nodvik [13]
Rod outside diameter (OD) (cm)	0.864 (0.340 in)	
Rod inside diameter (ID) (cm)	0.757 (0.298 in)[c]	
Pellet diameter (cm)	0.747 (0.294 in)	
Absorber pellet OD (cm)	0.7574	
Active fuel length (cm)	233.3(91.85 in)[d]	Page 9, Nodvik [13]
Cladding material	SS-348	Nodvik [13]
wt% U-234	0.020	
wt% U-235	3.400	
wt% U-236	0.020	
wt% U-238	96.56	
Core pressure (psia)	2000	
Instrument thimble OD (cm)	0.9525 (0.375 in)	Nodvik [13]
Instrument wall thickness (SS-348)	0.1651 (0.065 in)	(If present)
Instrument thimble ID (cm)	0.6223 (0.245 in)	

NOTES:
[a] Value calculated as follows: 20,908 kgU/core ÷ 23142 rods/core * 305 rods/assembly
[b] Pitch for row aligned with control blade vane is greater to allow clearance for vanes
[c] Calculated by subtracting the cladding thickness (0.021 inch) from the rod OD
[c] Includes SS-348 axial spacers between each set of 25 pellets

The arrangement of fuel assemblies in the first four Yankee Rowe cores is illustrated in Figure 3-3. The first core, Core I, consisted of 76 fresh fuel assemblies with an initial enrichment of 3.4 wt% ^{235}U, which was the enrichment used for all four cores. The second core retained two assemblies from Core I, Assemblies E6 and F5. The reload pattern for Core III did not include either E6 or F5, using four fresh assemblies in the center instead. Assembly E6 was replaced into its original location for Core IV, and the other three central assemblies were reinserted from Core III. Cores I and II did not employ dissolved boron for reactivity control; Cores III and IV did. Table 3-4 presents the operating history information. [1]

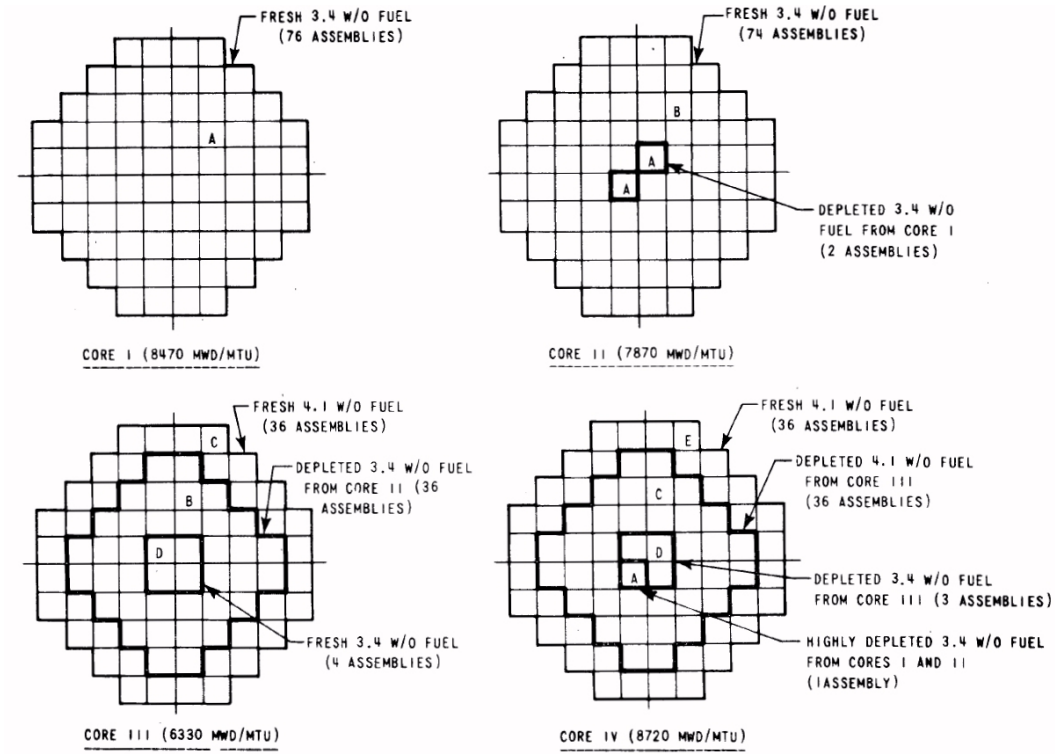

Figure 3-3
Fuel Arrangements for Yankee Rowe Cores I – IV

[1] Once the operating information for assembly E6 is documented, the others are also provided. This is because Assembly E6 was in the core for Cycles I, II, and IV. No fuel was examined from Cycle III in the Nodvik report [13]. So all of the information is needed for Assembly E6 and a subset of that information is used for the other assemblies.

Table 3-4
Yankee Rowe Operating Parameter Values

Parameter	Core I	Core II	Core IV
Days of operation	451.5	333.1	337.0
Shutdown Time (days)	129.0	370.0	N/A
Specific power (kW/kgU)			
E6-C-f6, 220.22 cm	13.07	13.00	16.97
E6-C-f6, 138.94 cm	27.74	30.79	22.57
E6-C-f6, 57.66 cm	29.44	29.86	24.01
E6-C-f6, 17.02 cm	18.14	19.07	16.76
E6-SE-c2, 138.94 cm	29.24	32.45	23.79
E6-SE-c2, 57.66 cm	29.51	29.93	24.07
E6-SE-e4, 138.94 cm	32.84	36.44	26.72
E6-SE-e4, 57.66 cm	33.13	33.60	27.02
Cumulative Burnup (GWd/MTU)			
E6-C-f6, 220.22 cm	5.90	10.23	15.95
E6-C-f6, 138.94 cm	12.53	22.78	30.39
E6-C-f6, 57.66 cm	13.29	23.24	31.33
E6-C-f6, 17.02 cm	8.19	14.54	20.19
E6-SE-c2, 138.94 cm	13.20	24.01	32.03
E6-SE-c2, 57.66 cm	13.33	23.30	31.41
E6-SE-e4, 138.94 cm	14.83	26.97	35.97
E6-SE-e4, 57.66 cm	14.96	26.15	35.26
Moderator Temperature (K)			
E6-C-f6, 220.22 cm	548	549	549
E6-C-f6, 138.94 cm	541	541	540
E6-C-f6, 57.66 cm	531	531	534
E6-C-f6, 17.02 cm	527	527	531
E6-SE-c2, 138.94 cm	541	541	540
E6-SE-c2, 57.66 cm	531	531	534
E6-SE-e4, 138.94 cm	541	541	540
E6-SE-e4, 57.66 cm	531	531	534

Table 3-4 (continued)
Yankee Rowe Operating Parameter Values

Parameter	Core I	Core II	Core IV
Cladding Temperature (K)			
E6-C-f6, 220.22 cm	558	559	561
E6-C-f6, 138.94 cm	560	563	557
E6-C-f6, 57.66 cm	551	552	550
E6-C-f6, 17.02 cm	540	540	543
E6-SE-c2, 138.94 cm	560	563	557
E6-SE-c2, 57.66 cm	540	552	550
E6-SE-e4, 138.94 cm	560	563	557
E6-SE-e4, 57.66 cm	540	552	550
Fuel Temperature (K)			
E6-C-f6, 220.22 cm	755	755	800
E6-C-f6, 138.94 cm	894	915	848
E6-C-f6, 57.66 cm	885	889	847
E6-C-f6, 17.02 cm	788	796	777
E6-SE-c2, 138.94 cm	894	915	848
E6-SE-c2, 57.66 cm	885	889	847
E6-SE-e4, 138.94 cm	894	915	848
E6-SE-e4, 57.66 cm	885	889	847
Boron concentration (ppmB[a], average)			
Startup – First hour	0	0	835
Next 2022 hours	0	0	707
Next 2022 hours	0	0	337
Final 4044 hours	0	0	0
Source: Nodvik [13] Tables 10-1 and 10-2 NOTE: [a] ppmB = Boron parts per million by mass			

The change in moderator temperature at each axial level was very small; therefore, an average temperature moderator density was used for each cycle. The moderator density values for each sample in the TRITON inputs were interpolated from page S2.5.14 of the SCALE manual using (1) a pressure of 2000 psia, and (2) for each cycle of operation, the average moderator temperature at the axial location of the RCA specimen.

The moderator temperatures and densities shown in Table 3-5 are provided for different heights along the fuel assembly length. These temperatures and densities are directly applicable to assemblies E5 and F5, which were positioned symmetrically relative to assembly E6. These assemblies were irradiated in the central zone of four assemblies in the Yankee Rowe reactor core. Further, these environmental parameters could be extended to the Core I RCA samples since explicit temperature and density data are not included in Ref. [13].

Table 3-5
Moderator Density Values for Yankee Rowe Samples

Sample ID	Average Moderator Temperature (K)	Moderator Density (g/cm^3)
E6-C-f6, 220.22 cm	548.67	0.7701
E6-C-f6, 138.94 cm	540.67	0.7834
E6-C-f6, 57.66 cm	532.0	0.7968
E6-C-f6, 17.02 cm	528.33	0.8020
E6-SE-c2, 138.94 cm	540.67	0.7834
E6-SE-c2, 57.66 cm	532.0	0.7968
E6-SE-e4, 138.94 cm	540.67	0.7834
E6-SE-e4, 57.66 cm	532.0	0.7968

Each RCA sample is identified by a sample identifier, using a scheme designating the assembly (AA); the subassembly (Center or one of the eight points of the compass); left-to-right alphabetic and top-to-bottom locations in the subassembly with reference to Fig. 3-1; and elevation at which the sample was taken (1 through 6, with 1 near the upper end of the fuel rod). The nomenclature is illustrated in Table 3-6.

Table 3-6
Sample Notation

	Sample Identifier Format: AASXY-H		
AA	Assembly Location in Core: e.g., E6		
S	Subassembly: C, N, S, E, W, NE, NW, SE, SW		
X	Rod Position in Subassembly: a – f, from left to right [1]		
Y	Rod Position in Subassembly: 1 – 6, from top to bottom [2]		
H	1	220.22	cm
	2	179.58	cm
	3	138.94	cm
	4	98.30	cm
	5	57.66	cm
	6	17.02	cm

Notes: [1] Subassembly NW only contains rods a – e (see Fig. 3-1)
[2] Subassembly SE only contains rods 1 – 5 (see Fig. 3-1)

Section 4: Yankee Rowe Calculations

The 79 RCA samples available from Yankee Rowe Cores I, II, and IV were analyzed using the TRITON computer code system of SCALE 6.0 to calculate the isotopic contents based upon the fuel rod environment and burnup. The reactivity (k_∞) of a uniform fuel rod array using the calculated isotopic contents was compared to the reactivity obtained using the measured RCA isotopic contents. The MCNP5, version 1.40, computer code was used for all reactivity calculations.

4.1 Computer Modeling Approach

Isotopic calculations for Yankee Rowe were performed using the TRITON computer code system of SCALE 6.0. Fuel irradiation parameters were obtained from Table A-1 of Ref. [13] and more precise burnup values from Table B-1 were used in place of the rounded-off values contained in Table A-1. The geometric model for the 17x18 fuel rod array creates a symmetric configuration of four fuel assemblies. The "PERIODIC" boundary of the TRITON computer code sequence might be used to simulate an infinite reactor of four-assembly blocks, but difficulties in implementing the "PERIODIC" boundary condition were encountered and the "REFLECTIVE" boundary condition was employed instead. This was facilitated by the representation of the Yankee Rowe fuel assembly as a control blade follower centered in a square array of rods that preserves the ratio of fuel, instrument tube, and control blade follower areas[2].

The computer requirements required to solve the flux distribution in the fuel assembly for each depletion time step of a TRITON model are substantial, so groups of fuel rods located in similar flux environments were assigned a common material type. The fuel rod containing the RCA sample was always assigned its own fuel type, numbered as material 1. A study was performed to identify the geometric and nuclear data library group structure that best suited the Yankee Rowe fuel assemblies, as shown in Table 4-1. Calculations were performed with the 44-group library and 238-group library structures using the CENTRM cross section processing option of the TRITON code sequence. Fuel rod materials were evaluated at a coarser division of only 4 fuel material types and also with 14 fuel material types. Experiments were also performed with larger number of fuel rod materials but did not show any improvement in predicted isotopic content (relative to the measured isotopic content), and are not reported here.

[2] This representation was suggested by Dr. Dale Lancaster of nuclearconsultants.com.

Interestingly, the combination of 4 material types to describe the geometry plus the 44-group library structure provided equivalent results to the combination of 14 material types and the 238-group library structure, while the combination of 14 materials and 44 energy groups resulted in poorer predicted isotopic results. It might be supposed that the coarser energy group structure and finer geometric detail resulted in flux convergence difficulties, but it should suffice to simply warn users of the TRITON system that an evaluation of the geometric and group structure choices is warranted. The selection of the CENTRM or NITAWL cross-section processing option did not appear to have a strong effect, but the CENTRM option was preferred.

Table 4-1
Evaluation of Geometric and Energy Group Detail

Case Number	Cross-Section Process	Energy Group Structure	Number of Fuel Materials	NEWT Grid	k_{Calc}	k_{RCA}	Δk_∞	$\Delta k_\infty/k_{RCA}$
TF1595	NITAWL	44	4	50	1.27624	1.27751	-0.00127	-0.00099
CF1595	CENTRM	44	4	50	1.27611	1.27751	-0.00140	-0.00110
LF1595	CENTRM	238	4	50	1.27700	1.27751	-0.00051	-0.00040
OF1595	NITAWL	44	14	50	1.26069	1.27751	-0.01682	-0.01317
OL1595	CENTRM	44	14	50	1.28937	1.27751	0.01186	0.00928
YL1595	CENTRM	238	14	50	1.28014	1.27751	0.00263	0.00206
YLGRD	CENTRM	238	14	70	1.27851	1.27751	0.00100	0.00078

All of the isotopic calculations reported for Yankee Rowe were obtained using the 238-group library structure and CENTRM option, with 14 material types and a selection of 50 for the NEWT grid for coarse-mesh acceleration. These parameters resulted in PC computer run times of 14 hours on a six-core AMD processor using solid state drives in RAID 0 configuration inserted directly into the computer PCI-E slots. The use of more sophisticated workstation computers could reduce the computer runtime requirements but were not evaluated for this study.

4.2 Calculations for the Eight RCAs of SAS2H Validation Studies

Eight of the Yankee Rowe RCAs have been used in SAS2H validation for many years. The investigations of use of TRITON for the 79 RCA samples began by re-analyzing these eight RCAs. Initial efforts resulted in very poor agreement between calculated and measured values, as shown in Table 4-2. Results

obtained using isotopic contents calculated using the SAS2H computer code sequence are compared to early results obtained using TRITON in Table 4-2, and reasonable agreement is shown for the samples from the Center subassembly of Assembly E6, but RCAs in the Intermediate and Perturbed regions show poor agreement. This unsatisfactory result was traced to over-burning of the fuel rods caused by thermal flux peaking when the control blades are replaced by water as the blades are withdrawn to bring the reactor to a critical state. The solution was to supplement the one-dimensional information used for SAS2H validation with the knowledge that the Yankee Rowe control blades have Zircaloy control blade followers that displace the "extra" water from the space between fuel assemblies. [12] Thus, the poor agreement shown in Table 4-2 resulted from a poor definition of the materials present in the Yankee Rowe core.

Table 4-2
Comparison of SAS2H to Early TRITON Calculations

Sample	Enrichment (wt% U-235)	Burnup (GWd/MTU)	SAS2H $\Delta k_\infty/k_{RCA}$	TRITON $\Delta k_\infty/k_{RCA}$
E6-C-f6-1	3.4	15.95	-0.005	0.0095
E6-C-f6-3	3.4	30.39	-0.018	0.0175
E6-C-f6-5	3.4	31.33	-0.017	0.0181
E6-C-f6-6	3.4	20.19	-0.01	0.0104
E6-SE-c2-3	3.4	32.03	-0.014	-0.0255
E6-SE-c2-5	3.4	31.41	-0.012	-0.0214
E6-SE-e4-3	3.4	35.97	0.0092	-0.0586
E6-SE-e4-5	3.4	35.26	0.0092	-0.0577
Note: These early TRITON calculations omitted the Zircaloy Control Blade Follower so that water not actually present caused a thermal flux peak and substantial over-burning of the fissile material.				

Inspection of Table 4-2 shows reasonable agreement between SAS2H and early TRITON calculations for the C subassembly, f6 rod samples, because this fuel rod was near the center of the fuel assembly in the asymptotic zone, and was neutronically isolated from the erroneous modeling of the control blade follower by a water gap. The SE subassembly c2 rod samples show a greater deviation from SAS2H results as they were located in the intermediate flux zone. The SE subassembly e4 rod was only one row of rods from the Zircaloy vanes of the follower, and is most affected by the "extra" water of the incorrectly modeled space between fuel assemblies.

Addition of the control blade follower and use of the modeling approaches described in Section 4.1 greatly improves the agreement between calculated and measured isotopic content reactivities, as shown in Table 4-3. Table 4-3 also provides results for individual samples (indicated with an identifier starting with

the letter "Y") as well as reactivity results for average burnup and isotopic content results as used by ORNL in SAS2H evaluations (indicated with an identifier starting with the letter "S"). The original investigators of RCAs at ORNL used averages for the burnup values and the isotopic contents, possibly because the separation of the three different sample analyses performed at the three different laboratories did not warrant individual entries as samples for the purposes of the 1D code validation. The results obtained with discrete RCA samples and results obtained with the three samples averaged show that either approach would produce valid results. The discrete results are shown for the remainder of the tables for this report in order to provide the most information to the reader.

Table 4-3
Final TRITON Results for Eight-RCA Set

Sample	Burnup (GWd/MTU)	ID	TRITON Discrete $\Delta k_\infty/k_{RCA}$	ID	TRITON Average $\Delta k_\infty/k_{RCA}$	SAS2H Average $\Delta k_\infty/k_{RCA}$
E6-C-f6-1	15.95	Y1	0.00187	S1	0.00206	-0.00524
E6-C-f6-2A	27.84	Y2A	0.00795			
E6-C-f6-2B	27.84	Y2B	0.01207			
E6-C-f6-3A	30.39	Y3A	0.01111	S2	0.00831	-0.01798
E6-C-f6-3B	30.39	Y3B	0.00818			
E6-C-f6-3C	30.39	Y3C	0.00795			
E6-C-f6-4	32.48	Y4	0.01317			
E6-C-f6-5A	31.33	Y5A	0.01231	S3	0.01187	-0.01701
E6-C-f6-5B	31.33	Y5B	0.01071			
E6-C-f6-6	20.19	Y6	0.00320	S4	0.00222	-0.01044
E6-SE-c2-3	32.03	Y7	0.00819	S5	0.00879	-0.01369
E6-SE-c2-5	31.41	Y8	0.00770	S6	0.00801	-0.01224
E6-SE-e4-3	35.97	Y9	0.00287	S7	0.00182	0.00918
E6-SE-e4-5	35.26	Y10	0.00604	S8	0.00152	0.00922

Inspection of Table 4-3 shows the superior agreement between the reactivity of isotopic contents calculated with TRITON and RCA measurements, relative to SAS2H. The IDs Y2A and Y2B as well as Y5A and Y5B are samples analyzed at two different laboratories, while IDs Y3A, Y3B, and Y3C represent a sample that was analyzed at three different laboratories. The "Average" result occurs between the three laboratory values, and the range of values is reasonably small.

Further comparisons between multiple measurements are shown for Assembly E6 in Table 4-4, especially samples Y12A through Y12G which are all from the same fuel section. Three measurements were performed by Tracerlab, two by GE-Vallecitos, and two by the New Brunswick laboratories. The average

reactivity $\Delta k_\infty / k_{RCA}$ of these measurements is -0.00213 with a range of -0.00246 to +0.00116. The small range of the seven measurements is an expression of the high quality of the Yankee Rowe data. (Note that with only 7 data points, statistical measures such as standard deviation have little meaning.)

4.3 Results of Yankee Rowe TRITON/MCNP Calculations

The results of TRITON calculations with reactivities provided by MCNP are shown in Table 4-4 for 3-cycle Assembly E6, in Table 4-5 for 2-cycle Assembly F5, and in Table 4-6 for 1-cycle Assemblies E5 and F4. These 79 RCA samples were obtained from fuel assemblies that were centrally located in the reactor core, where flat flux conditions prevailed.

A number of fuel samples were assayed by different laboratories and some assays were repeated, so the multiple assays for a sample are indicated by a capital letter added to the end of the sample identifier. The results for each assay are provided in Tables 4-4, 4-5, and 4-6. There are 79 assays of 53 fuel samples provided in the Tables. Users of this isotopic validation data may choose to combine all of the multiple radiochemical assays for a fuel sample into a single data point by calculating average burnup and measured isotopic contents, as was done by early ORNL researchers. Another approach facilitated by the speed of modern computers is to perform individual TRITON and MCNP calculations for each sample assay and then average the $\Delta k_\infty / k_{RCA}$ values. The later approach is evaluated in Chapter 5 to determine the sensitivity of the calculated bias and uncertainty factors to the treatment of the multiple samples.

Table 4-4
TRITON Results for Three-Cycle Assembly E6

Sample	k_{Calc}	k_{RCA}	ID	Δk_{∞}	σ	$\Delta k_{\infty}/k_{RCA}$	Flux Zone
E6-C-f6-1	1.28114	1.27875	Y1	0.00239	0.00019	0.00187	Asymptotic
E6-C-f6-2A	1.24399	1.23418	Y2A	0.00981	0.00020	0.00795	Asymptotic
E6-C-f6-2B	1.24689	1.23202	Y2B	0.01487	0.00020	0.01207	Asymptotic
E6-C-f6-3A	1.23858	1.22497	Y3A	0.01361	0.00020	0.01111	Asymptotic
E6-C-f6-3B	1.23821	1.22816	Y3B	0.01005	0.00020	0.00818	Asymptotic
E6-C-f6-3C	1.24188	1.23209	Y3C	0.00979	0.00020	0.00795	Asymptotic
E6-C-f6-4	1.23586	1.21979	Y4	0.01607	0.00020	0.01317	Asymptotic
E6-C-f6-5A	1.23797	1.22292	Y5A	0.01505	0.00020	0.01231	Asymptotic
E6-C-f6-5B	1.2381	1.22498	Y5B	0.01312	0.00020	0.01071	Asymptotic
E6-C-f6-6	1.26338	1.25935	Y6	0.00403	0.00020	0.00320	Asymptotic
E6-SE-c2-3	1.22984	1.21985	Y7	0.00999	0.00020	0.00819	Intermediate
E6-SE-c2-5	1.22759	1.21821	Y8	0.00938	0.00020	0.00770	Intermediate
E6-SE-e4-3	1.18754	1.18414	Y9	0.00340	0.00020	0.00287	Perturbed
E6-SE-e4-5	1.18690	1.17978	Y10	0.00712	0.00020	0.00604	Perturbed
E6-SE-f5-1	1.24648	1.26822	Y11	-0.02174	0.00019	-0.01714	Perturbed
E6-SE-f5-3A	1.15083	1.15195	Y12A	-0.00112	0.00020	-0.00097	Perturbed
E6-SE-f5-3B	1.14903	1.15274	Y12B	-0.00371	0.00019	-0.00322	Perturbed
E6-SE-f5-3C	1.14791	1.15320	Y12C	-0.00529	0.00020	-0.00459	Perturbed

Table 4-4 (continued)
TRITON Results for Three-Cycle Assembly E6

Sample	k_{Calc}	k_{RCA}	ID	Δk_{∞}	σ	$\Delta k_{\infty}/k_{RCA}$	Flux Zone
E6-SE-f5-3D	1.14790	1.14975	Y12D	-0.00185	0.00020	-0.00161	Perturbed
E6-SE-f5-3E	1.14790	1.14936	Y12E	-0.00146	0.00020	-0.00127	Perturbed
E6-SE-f5-3F	1.15038	1.15200	Y12F	-0.00162	0.00020	-0.00141	Perturbed
E6-SE-f5-3G	1.15044	1.15258	Y12G	-0.00214	0.00019	-0.00186	Perturbed
E6-SE-f5-5A	1.14905	1.15278	Y13A	-0.00373	0.00019	-0.00324	Perturbed
E6-SE-f5-5B	1.14835	1.15381	Y13B	-0.00546	0.00019	-0.00473	Perturbed
E6-SE-f5-6	1.20276	1.21331	Y14	-0.01055	0.00020	-0.00870	Perturbed
E6-NW-a1-1	1.24732	1.26408	Y15	-0.01676	0.00018	-0.01326	Perturbed
E6-NW-a1-2	1.16129	1.16605	Y16	-0.00476	0.00020	-0.00408	Perturbed
E6-NW-a1-3A	1.15095	1.14991	Y17A	0.00104	0.00020	0.00090	Perturbed
E6-NW-a1-3B	1.14913	1.15233	Y17B	-0.00320	0.00020	-0.00278	Perturbed
E6-NW-a1-3C	1.14920	1.14338	Y17C	0.00582	0.00020	0.00509	Perturbed
E6-NW-a1-4A	1.14782	1.14408	Y18A	0.00374	0.00020	0.00327	Perturbed
E6-NW-a1-4B	1.14452	1.13828	Y18B	0.00624	0.00019	0.00548	Perturbed
E6-NW-a1-4C	1.14942	1.14095	Y18C	0.00847	0.00019	0.00742	Perturbed
E6-NW-a1-5	1.15121	1.15058	Y19	0.00063	0.00019	0.00055	Perturbed
E6-NW-a1-6	1.21052	1.21896	Y20	-0.00844	0.00019	-0.00692	Perturbed
E6-C-a1-1	1.27760	1.27332	Y21	0.00428	0.00019	0.00336	Asymptotic
E6-C-a1-2A	1.24610	1.23379	Y22A	0.01231	0.00020	0.00998	Asymptotic
E6-C-a1-2B	1.24649	1.23391	Y22B	0.01258	0.00019	0.01020	Asymptotic
E6-C-a1-3A	1.23908	1.22338	Y23A	0.01570	0.00020	0.01283	Asymptotic
E6-C-a1-3B	1.23797	1.22721	Y23B	0.01076	0.00020	0.00877	Asymptotic
E6-C-a1-3C	1.24107	1.21748	Y23C	0.02359	0.00020	0.01938	Asymptotic

Table 4-4 (continued)
TRITON Results for Three-Cycle Assembly E6

Sample	k_{Calc}	k_{RCA}	ID	Δk_{∞}	σ	$\Delta k_{\infty}/k_{RCA}$	Flux Zone
E6-C-a1-3D	1.24050	1.22020	Y23D	0.02030	0.00020	0.01664	Asymptotic
E6-C-a1-3E	1.24063	1.21753	Y23E	0.02310	0.00020	0.01897	Asymptotic
E6-C-a1-3	1.23994	1.22109	Y23AVG	0.01885	0.00020	0.01544	Asymptotic
E6-C-a1-4A	1.23854	1.23735	Y24A	0.00119	0.00020	0.00096	Asymptotic
E6-C-a1-4B	1.23735	1.22064	Y24B	0.01671	0.00020	0.01369	Asymptotic
E6-C-a1-5A	1.23878	1.21945	Y25A	0.01933	0.00020	0.01585	Asymptotic
E6-C-a1-5B	1.24058	1.22146	Y25B	0.01912	0.00020	0.01565	Asymptotic
E6-C-a1-6	1.26386	1.25732	Y26	0.00654	0.00019	0.00520	Asymptotic
E6-C-a6-1	1.28024	1.27645	Y27	0.00379	0.00018	0.00297	Asymptotic
E6-C-a6-2	1.24725	1.23514	Y28	0.01211	0.00020	0.00980	Asymptotic
E6-C-a6-3	1.24188	1.22717	Y29	0.01471	0.00020	0.01199	Asymptotic
E6-C-a6-4	1.24023	1.22150	Y30	0.01873	0.00020	0.01533	Asymptotic
E6-C-a6-6A	1.26523	1.25968	Y31A	0.00555	0.00020	0.00441	Asymptotic
E6-C-a6-6B	1.26422	1.25967	Y31B	0.00455	0.00019	0.00361	Asymptotic
E6-C-f1-1A	1.28120	1.27602	Y32A	0.00518	0.00018	0.00406	Asymptotic
E6-C-f1-1B	1.27998	1.27619	Y32B	0.00379	0.00019	0.00297	Asymptotic
E6-C-f1-2	1.24624	1.24408	Y33	0.00216	0.00020	0.00174	Asymptotic
E6-C-f1-6	1.26387	1.25627	Y34	0.00760	0.00019	0.00605	Asymptotic

Table 4-5
TRITON Results for Two-Cycle Assembly F5

Sample	k_{Calc}	k_{RCA}	ID	Δk_∞	σ	$\Delta k_\infty / k_{RCA}$	Flux Zone
F5-SE-f5-3	1.18841	1.20009	Y37	-0.01168	0.00020	-0.00973	Perturbed
F5-SE-f5-6	1.24901	1.25632	Y38	-0.00731	0.00018	-0.00582	Perturbed
F5-C-a1-1	1.31496	1.31437	Y47	0.00059	0.00018	0.00045	Asymptotic
F5-C-a1-3A	1.25923	1.25535	Y48A	0.00388	0.00019	0.00309	Asymptotic
F5-C-a1-3B	1.25828	1.25646	Y48B	0.00182	0.00019	0.00145	Asymptotic
F5-C-f6-1	1.31136	1.31146	Y49	-0.00010	0.00018	-0.00008	Asymptotic
F5-C-f6-2	1.26604	1.26428	Y50	0.00176	0.00020	0.00139	Asymptotic
F5-C-f6-4	1.25668	1.24781	Y51	0.00887	0.00019	0.00711	Asymptotic
F5-C-f6-5	1.25568	1.25163	Y52	0.00405	0.00020	0.00324	Asymptotic
F5-C-f6-6	1.28123	1.28048	Y53	0.00075	0.00018	0.00059	Asymptotic

Table 4-6
TRITON Results for One-Cycle Assemblies E5 and F4

Sample	k_{Calc}	k_{RCA}	ID	Δk_∞	σ	$\Delta k_\infty / k_{RCA}$	Flux Zone
E5-SW-a5-3	1.25680	1.26406	Y35	-0.00726	0.00018	-0.00574	Perturbed
E5-SW-a5-5	1.26052	1.26814	Y36	-0.00762	0.00018	-0.00601	Perturbed
E5-C-a6-3	1.29366	1.29514	Y44	-0.00148	0.00018	-0.00114	Asymptotic
E5-C-a6-5	1.29349	1.29345	Y45	0.00004	0.00020	0.00003	Asymptotic
E5-C-a6-6	1.32325	1.32337	Y46	-0.00012	0.00018	-0.00009	Asymptotic
F4-C-f1-1	1.33292	1.33624	Y39	-0.00332	0.00018	-0.00248	Asymptotic
F4-C-f1-4	1.29126	1.29266	Y40	-0.00140	0.00018	-0.00108	Asymptotic
F4-C-f1-5	1.29280	1.29325	Y41	-0.00045	0.00018	-0.00035	Asymptotic
F4-C-f6-3	1.29250	1.29162	Y42	0.00088	0.00018	0.00068	Asymptotic
F4-C-f6-4	1.29962	1.29918	Y43	0.00044	0.00018	0.00034	Asymptotic

4.4 Gamma Shielding Validation by ^{137}Cs RCAs

The uranium and plutonium RCA data used to calculate $\Delta k_{\infty}/k_{RCA}$ measured versus calculated results were supplemented by assays of ^{137}Cs isotopic content. The ^{137}Cs isotope dominates the gamma radiation source for the range of cooling times after discharge from a reactor applicable to the storage and transport of spent fuel. Thus the comparison of measured assay ^{137}Cs contents against TRITON calculations can provide insight into the accuracy of gamma shielding calculations. The ^{137}Cs isotopic content data were obtained from Table A-1 of Ref. [13], and consists of 60 measurements. The measured and TRITON calculated isotopic contents (in Curies per gram of uranium) are tabulated in Table 4-7. The fractional difference between measured and calculated values, normalized by the measured value, is also tabulated for each data point.

Table 4-7
^{137}Cs Measured and Calculated Isotopic Contents

Rod Location	Sample ID	Axial Sample Zone	RCA Measured Ci/gU	TRITON Calculated Ci/gU	ΔCs-137 Normalized Fraction
E6-C-f6	Y1	1	0.0546	0.0512	0.0625
	Y2A	2	0.1014	0.0901	0.1108
	Y2B	2(Ext)	0.0865	0.0883	-0.0209
	Y3A	3	0.1074	0.1005	0.0639
	Y3C	3(Ext)	0.0914	0.0945	-0.0338
	Y4	4	0.1084	0.1057	0.0248
	Y5A	5	0.1081	0.1017	0.0591
	Y6	6	0.0704	0.0648	0.0786
E6-SE-c2	Y7	3	0.1190	0.1022	0.1415
	Y8	5	0.1086	0.1022	0.0588
E6-SE-e4	Y9	3	0.1253	0.1177	0.0603
	Y10	5	0.1158	0.1153	0.0044
E6-SE-f5	Y11	1	0.0633	0.0584	0.0773
	Y12A	3	0.1300	0.1379	-0.0600
	Y13A	5	0.1377	0.1355	0.0160
	Y14	6	0.0926	0.0850	0.0814

Table 4-7 (continued)
^{137}Cs Measured and Calculated Isotopic Contents

Rod Location	Sample ID	Axial Sample Zone	RCA Measured Ci/gU	TRITON Calculated Ci/gU	ΔCs-137 Normalized Fraction
E6-NW-a1	Y15	1	0.0590	0.0578	0.0194
	Y16	2	0.1221	0.1250	-0.0233
	Y17A	3	0.1358	0.1381	-0.0173
	Y17C	3	0.1366	0.1400	-0.0251
	Y18A	4	0.1337	0.1425	-0.0655
	Y18C	4	0.1299	0.1403	-0.0798
	Y19	5	0.1305	0.1336	-0.0235
	Y20	6	0.0818	0.0794	0.0300
E6-C-a1	Y21	1	0.0554	0.0539	0.0279
	Y22A	2	0.0941	0.0902	0.0416
	Y22B	2	0.0929	0.0900	0.0315
	Y23A	3	0.0998	0.1010	-0.0119
	Y23C	3	0.0974	0.0976	-0.0023
	Y24A	4	0.1063	0.1062	0.0004
	Y25A	5	0.0996	0.1018	-0.0215
	Y25B	5	0.0999	0.0981	0.0177
	Y26	6	0.0638	0.0644	-0.0082
E6-C-a6	Y27	1	0.0551	0.0524	0.0502
	Y28	2	0.1012	0.0918	0.0922
	Y29	3	0.1103	0.1030	0.0664
	Y30	4	0.1076	0.1068	0.0076
	Y31A	6	0.0702	0.0643	0.0836
E6-C-f1	Y32A	1	0.0547	0.0513	0.0613
	Y33	2	0.0975	0.0933	0.0433
	Y34	6	0.0693	0.0656	0.0535

Table 4-7 (continued)
^{137}Cs Measured and Calculated Isotopic Contents

Rod Location	Sample ID	Axial Sample Zone	RCA Measured Ci/gU	TRITON Calculated Ci/gU	ΔCs-137 Normalized Fraction
E5-SW-a5	Y35	3	0.0569	0.0545	0.0417
	Y36	5	0.0529	0.0519	0.0182
F5-SE-f5	Y37	3	0.0981	0.1011	-0.0303
	Y38	6	0.0588	0.0569	0.0314
F4-C-f1	Y39	1	0.0185	0.0208	-0.1230
	Y40	4	0.0465	0.0450	0.0336
	Y41	5	0.0404	0.0434	-0.0751
F4-C-f6	Y42	3	0.0464	0.0441	0.0504
	Y43	4	0.0423	0.0390	0.0787
E5-C-a6	Y44	3	0.0372	0.0424	-0.1393
	Y45	5	0.0437	0.0418	0.0441
	Y46	6	0.0287	0.0249	0.1350
F5-C-a1	Y47	1	0.0303	0.0289	0.0443
	Y48A	3	0.0744	0.0721	0.0306
F5-C-f6	Y49	1	0.0336	0.0309	0.0811
	Y50	2	0.0628	0.0652	-0.0371
	Y51	4	0.0804	0.0766	0.0463
	Y52	5	0.0754	0.0757	-0.0039
	Y53	6	0.0505	0.0498	0.0141

4.5 Neutron Shielding Validation by ^{242}Cm and ^{244}Cm RCAs

The uranium and plutonium RCA data used to calculate $\Delta k_\infty/k_{RCA}$ measured versus calculated results were supplemented by 12 assays of ^{242}Cm and 7 assays of ^{244}Cm isotopic content. The ^{242}Cm isotope dominates the neutron radiation source for the range of cooling times applicable to transport of spent fuel in a year

or less after discharge from the reactor. At longer cooling times, the ^{244}Cm isotope dominates and determines the neutron source for storage and transport of spent fuel in the five-year or more timeframe. Thus the comparison of measured assay ^{242}Cm and ^{244}Cm contents against TRITON calculations can provide insight into the accuracy of neutron shielding calculations. The ^{242}Cm and ^{244}Cm isotopic content data is obtained from Table D-5 of Ref. [13]. The isotopic contents are expressed as the ratio of the number of ^{242}Cm or ^{244}Cm atoms to the number of ^{239}Pu atoms.

Table 4-8
^{242}Cm and ^{244}Cm Measured vs. Calculated Isotopic Contents

Core Location	Sample ID	Raw ^{242}Cm/^{239}Pu	Average of Raw ^{242}Cm/^{239}Pu	TRITON-calculated ^{242}Cm/^{239}Pu	ΔCm-242 Normalized Fraction	Raw ^{244}Cm/^{239}Pu	Average of Raw ^{244}Cm/^{239}Pu	TRITON-calculated ^{244}Cm/^{239}Pu	ΔCm-244 Normalized Fraction
E5-SW-a5-3	Y35	5.56	5.82	2.84	0.511	1.90	1.90	2.10	-0.107
		6.07							
F5-SE-f5-3	Y37	23.80	23.70	17.85	0.247	24.50	24.50	34.23	-0.397
		23.60							
E6-NW-a1-3C	Y17C	55.90	54.20	47.76	0.119	112.00	112.00	136.93	-0.223
		52.50							
E6-NW-a1-4C	Y18C	55.90	55.30	46.70	0.156	113.00	113.00	137.10	-0.213
		54.70							
E6-SE-f5-3E	Y12E	48.80	46.35	44.92	0.031	99.80	97.57	123.40	-0.265
		49.70				97.00			
		37.60							
		49.30				95.90			
				Mean	**0.213**			**Mean**	**-0.241**
Note: The raw data measurements are averaged together to allow direct comparison to TRITON calculated values.									

Section 5: Bias and Uncertainty for Actinide-only Isotopic Validation

The validation of Actinide-Only Burnup Credit follows standard industry practices and uses RCA data that measure the isotopic contents of spent fuel samples, which can be compared to those calculated using the TRITON computer code sequence. The reactivity effect of the differences between measured and calculated isotopic compositions is evaluated using the MCNP computer code.

Note that some samples provide the measured isotopic contents of ^{238}Pu, which is usually included in Actinide-Only burnup credit. The ^{238}Pu isotope was omitted in this report so that a common basis for measured isotopes could be applied to all samples. Further, the effect of gadolinia burnable absorber contained within fueled rods was omitted. The Calvert Cliffs and Takahama RCA datasets contain Gd rods, but there is insufficient data for the purpose of this report and those RCA samples were omitted.

Statistical analyses of the $\Delta k_\infty/k_{RCA}$ results are provided below. The mean and standard deviation values for the Nine-Reactor and Yankee Rowe datasets are provided so that it can be seen that the Yankee Rowe RCAs confirm the isotopic validation results for the Nine-Reactor dataset. Thus, the Yankee Rowe RCA data adds confidence to earlier isotopic validation efforts.

Plots of data are provided to illustrate the normality of the distribution of the $\Delta k_\infty/k_{RCA}$ results, both as a histogram and as a normality quantile plot. The normality of the distributions is determined by the Shapiro-Wilkes test. Trend analyses versus fuel burnup and initial enrichment are also presented.

5.1 Nine-Reactor Dataset

The comparison of measured and calculated reactivities ($\Delta k_\infty/k_{RCA}$) follows a normal distribution, as shown in Figures 5-1 and 5-2.

Figure 5-1 illustrates the distribution of $\Delta k_\infty/k_{RCA}$ data points and the equivalent normal distribution (red line). The bias for the Nine-Reactor dataset is the mean value of 0.006075 and the standard deviation is 0.007100. The isotopic bias and uncertainty correction term for Actinide-Only Burnup Credit analyses is thus

0.006075 + 1.645*0.00710, or 0.01775.[3] This is the value that must be added to calculated k_∞ values during a burnup credit analysis.

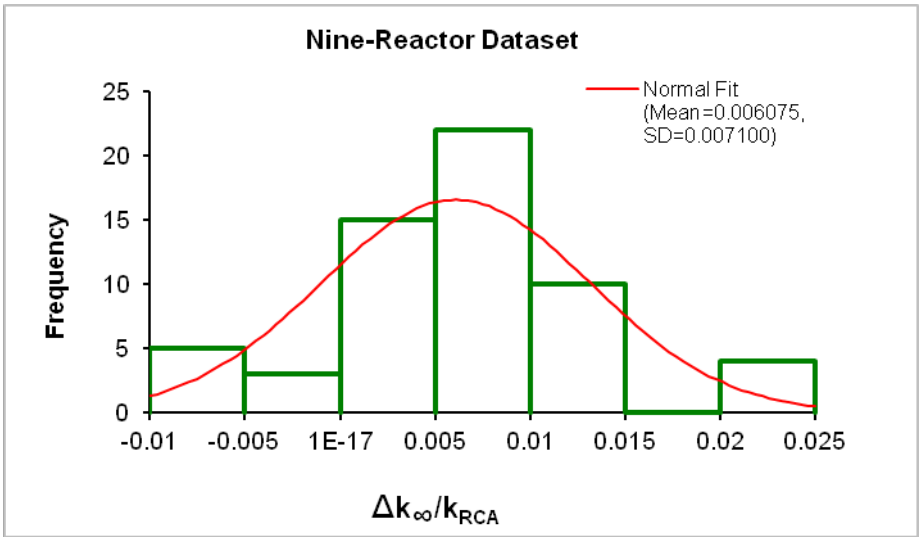

Figure 5-1
Plot of Nine-Reactor $\Delta k_\infty / k_{RCA}$ Results

Figure 5-2 illustrates the normality of the Nine-Reactor dataset and provides the results of the Shapiro-Wilkes test, p = 0.2666 and W probability = 0.98. If p were less than 0.05, then the data would have failed the test for normality. The demonstration of the normality of the $\Delta k_\infty / k_{RCA}$ distribution indicates that the bias and uncertainty values are meaningful.

Regression analyses were performed for the Nine-Reactor dataset using a second-order polynomial fit, which yielded an adjusted Residual value of 0.17 for enrichment and 0.01 for burnup, so there is no correlation between fuel enrichment or burnup and the Δk_∞ values.

[3] The 1.645 multiplier is associated with the "single-sided" tolerance limit value. The more common multiplier used is the "double-sided" tolerance limit value which is 2.0, hence the use of "2σ" in many reports in conjunction with the 95 percent Confidence Limit. The "double-sided" value protects against a value falling more that 2.5 percent above the mean or less than 2.5 percent below the mean 95 percent of the time. The "single-sided" value protects against under-predicting (or over-predicting, but not both) k-eff by more than 5 percent 95 percent of the time. Since for criticality purpose, over-predicting k-eff is conservative, the single-sided version is used.

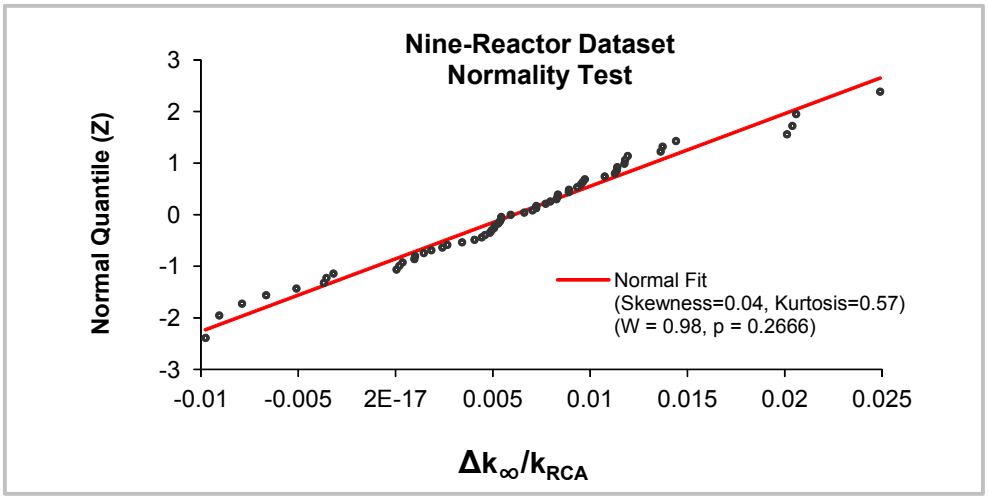

Figure 5-2
Nine-Reactor $\Delta k_\infty/k_{RCA}$ Normality Plot

5.2 Yankee Rowe Dataset

The comparison of measured and calculated reactivities ($\Delta k_\infty/k_{RCA}$) follows a normal distribution, as shown in Figures 5-3 and 5-4.

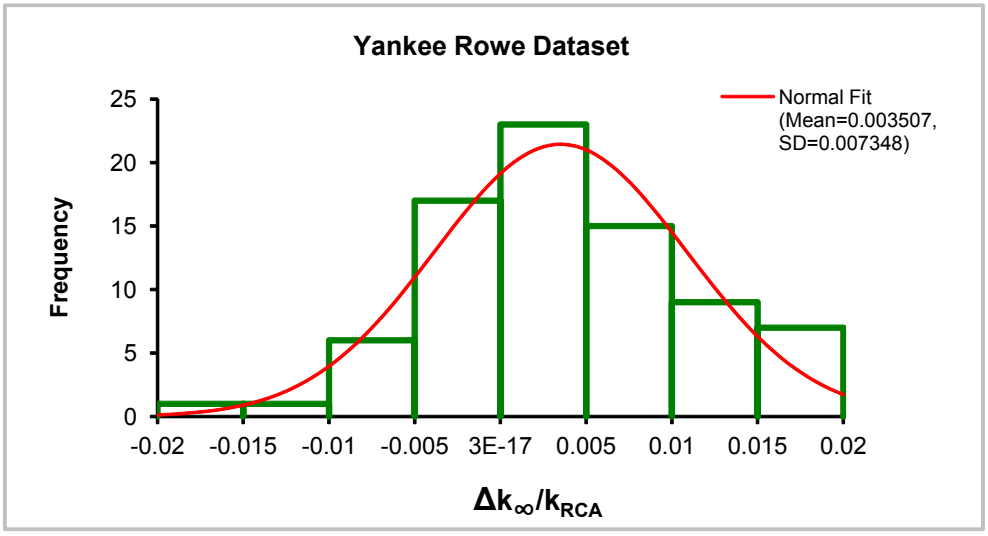

Figure 5-3
Plot of Yankee Rowe $\Delta k_\infty/k_{RCA}$ Results

Figure 5-3 shows the distribution of $\Delta k_\infty/k_{RCA}$ data points and the equivalent normal distribution (red line). The bias for the Yankee Rowe dataset is the mean value of 0.003507 and the standard deviation is 0.007348. The isotopic bias and

uncertainty correction term for Actinide-Only Burnup Credit analyses is thus 0.003507 + 1.645*0.007348, or 0.01560. This is the value that must be added to calculated k_∞ values during a burnup credit analysis based upon Yankee Rowe RCA alone.

A graphical representation of the normality of the Yankee Rowe dataset of 79 RCAs is shown in Figure 5-4. The Shapiro-Wilkes test for normality is used and provides a W probability = 0.99 and p = 0.6348, indicating that the Yankee Rowe dataset is normally distributed. Note that the Yankee Rowe dataset fits a normal distribution significantly better than the Nine-Reactor dataset.

Regression analyses were performed for the Yankee Rowe dataset using a second-order polynomial fit, which yielded an adjusted Residual value of 0.21 for burnup, so there is no correlation between burnup and the Δk_∞ values.

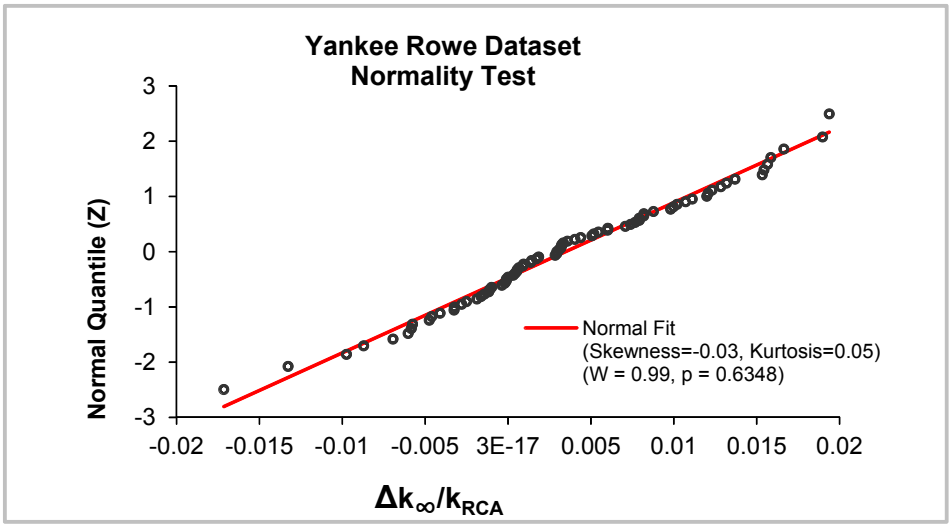

Figure 5-4
Yankee Rowe $\Delta k_\infty/k_{RCA}$ Normality Plot

5.3 Combined Dataset

The purpose of adding the Yankee Rowe data to the validation of isotopics for burnup credit is to improve the confidence in the validation, so the results of the Nine-Reactor and Yankee Rowe RCA calculations are added into a combined dataset. The combined dataset comparison of measured and calculated reactivities ($\Delta k_\infty / k_{RCA}$) follows a normal distribution, as shown in Figures 5-5 and 5-6.

The combination of the RCA data from the Nine-Reactor and Yankee Rowe datasets is shown in Figure 5-5.

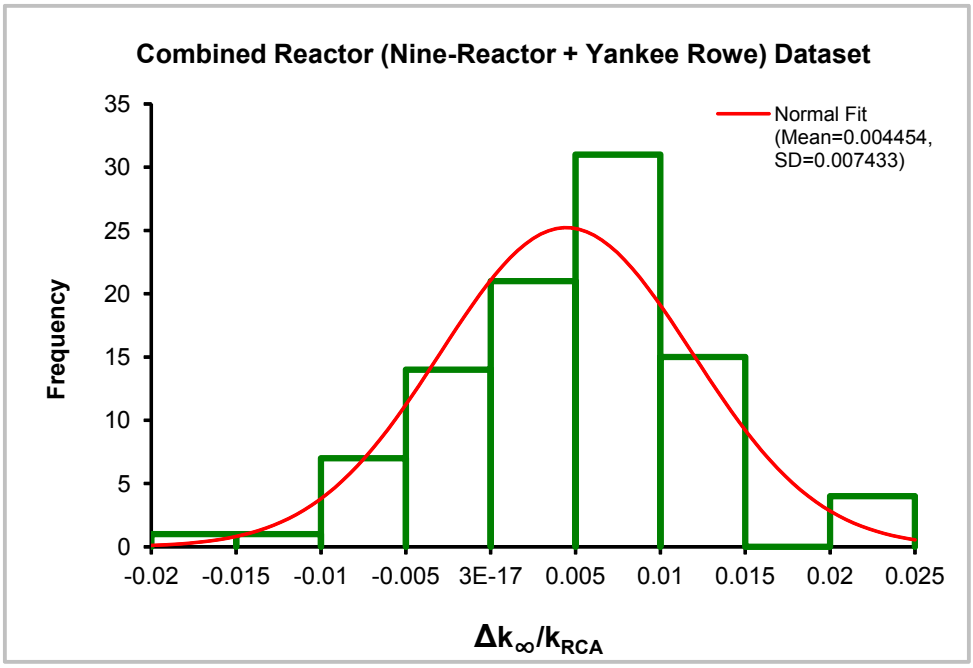

Figure 5-5
Plot of Combined Nine-Reactor + Yankee Rowe $\Delta k_\infty/k_{RCA}$ Datasets

Figure 5-3 shows the histogram distribution of $\Delta k_\infty / k_{RCA}$ data points and the equivalent normal distribution (red line). The bias for the combined dataset is the mean value of 0.004454 and the standard deviation is 0.007433. The isotopic bias and uncertainty correction term for Actinide-Only Burnup Credit analyses is thus 0.004454 + 1.645*0.007433, or 0.01668. This is the value that must be added to calculated k_∞ values during a burnup credit analysis based upon the combined 138 RCA dataset.

A graphical representation of the normality of the combined dataset of 138 RCAs is shown in Figure 5-6. The Shapiro-Wilkes test for normality is used and provides a W probability = 1.00 and p = 0.9976, indicating that the combined dataset is normally distributed. Note that the combined dataset fits a normal distribution very well indeed, and therefore the statistical results for the mean (bias) and standard deviation can be trusted.

Regression analyses were performed for the combined Nine-Reactor + Yankee Rowe dataset using a second-order polynomial fit, which yielded an adjusted Residual value of 0.07 for enrichment and 0.01 for burnup, so there is no correlation between fuel enrichment or burnup and the Δk_∞ values.

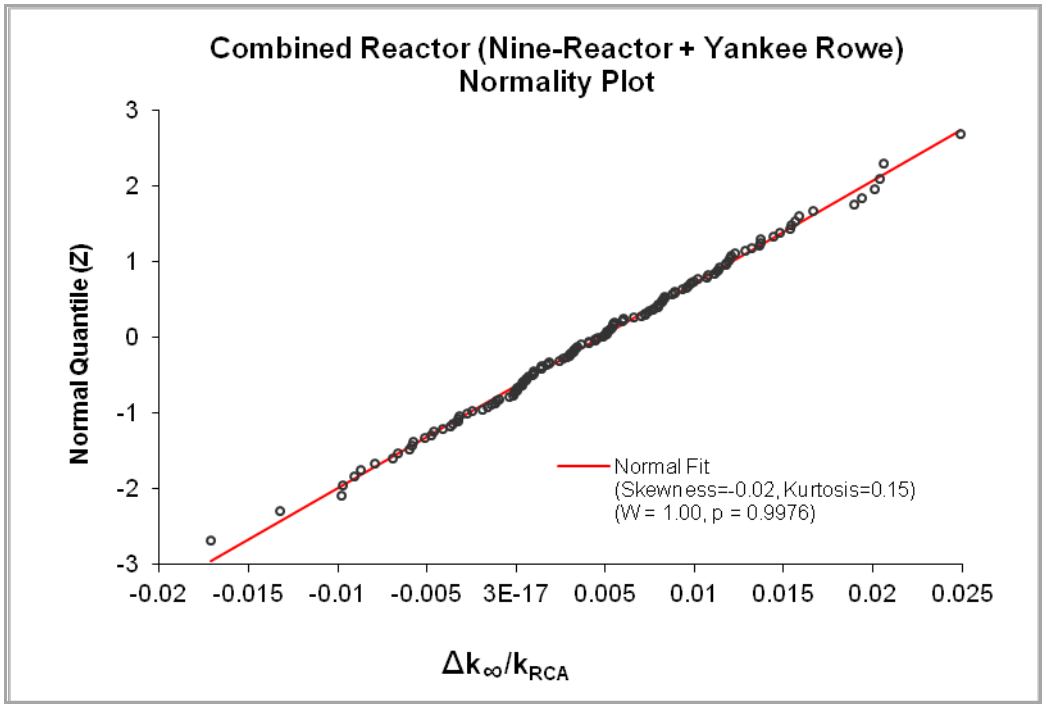

Figure 5-6
Combined Nine-Reactor + Yankee Rowe $\Delta k_\infty/k_{RCA}$ Normality Plot

Comparison of the isotopic bias and uncertainty correction terms shows that the Nine-Reactor dataset has the higher value (0.01775), the Yankee Rowe dataset has the smaller value (0.01560), and as would be expected, the combined dataset value falls in between (0.01668). These values are all quite close together, indicating that the addition of the Yankee Rowe data adds confidence to the Nine-Reactor dataset. All three values are lower than the equivalent value for SAS2H validation, which might be expected based upon the improved accuracy of the TRITON depletion code sequence. The increase in the number of RCAs from 59 to 138 thus significantly improves the confidence in the statistical validity of the isotopic calculations using TRITON, and the application of Actinide-Only Burnup Credit, either by itself or as a component of Full burnup credit.

Section 6: Gamma and Neutron Radiation Sources

The Yankee Rowe RCA program included measurement of the ^{137}Cs isotopic content, which is the dominant contributor to gamma radiation from spent fuel.[4] The RCA program also evaluated the uncertainty of heavier actinides, especially ^{242}Cm and ^{244}Cm, which are the dominant neutron source. At low burnup, ^{239}Pu is the dominant neutron emitter.

The TRITON depletion code system calculates these isotopes and the quantity of each isotope may be related to ^{238}U, in order to allow comparison to the measured quantities. The accuracy of the radiochemical measurement of ^{137}Cs is 6 percent. The accuracy of the Cm radiochemical measurements is also about 6 percent.

6.1 Gamma Radiation Source

The ^{137}Cs RCA gamma source data is plotted in Figure 6-1 as the difference between measured and TRITON calculated, normalized by the measured value.

[4] ^{60}Co is the dominant contributor to the gamma radiation from the end fittings.

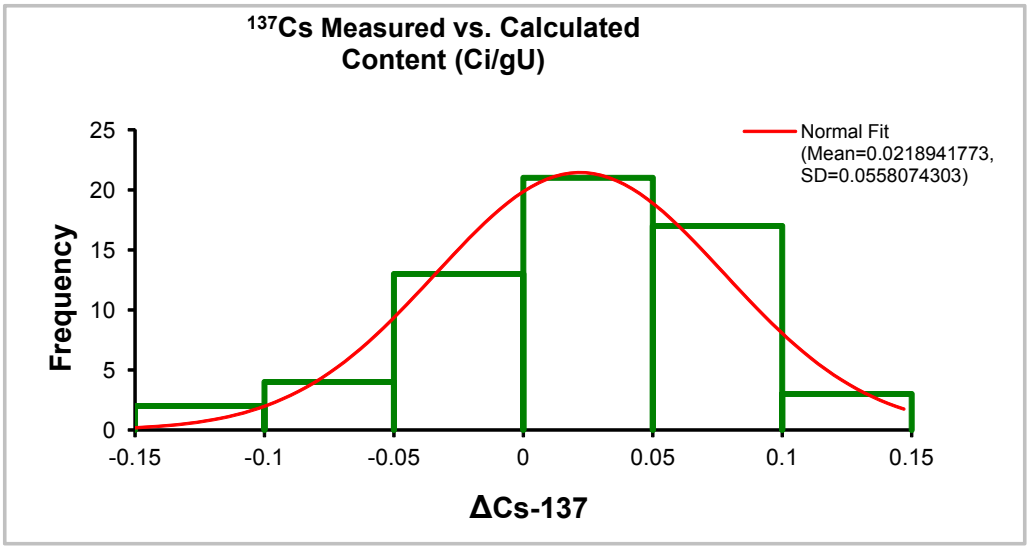

Figure 6-1
Plot of Normalized ΔCs-137 for the Yankee Rowe Dataset

The mean value of 0.02189 indicates that the gamma source (as represented by ΔCs-137) has a bias of 2.19 percent and a standard deviation of 0.0558 or 5.6 percent for the 60-sample dataset. These results would suggest that the gamma radiation source for transport and storage systems has a bias correction factor of 0.02189 + 1.645*0.0558 or 0.1137. This result is consistent with the expectation that gamma sources are known within 5 to 10 percent. [14]

A graphical representation of the normality of the ^{137}Cs gamma source dataset of 60 RCAs is shown in Figure 6-2. Inspection of figure 6-2 shows that the distribution of ΔCs-137 is normal, with a probability W of 0.97 and p value of 0.2151. Thus the statistical results for ΔCs-137 are meaningful.

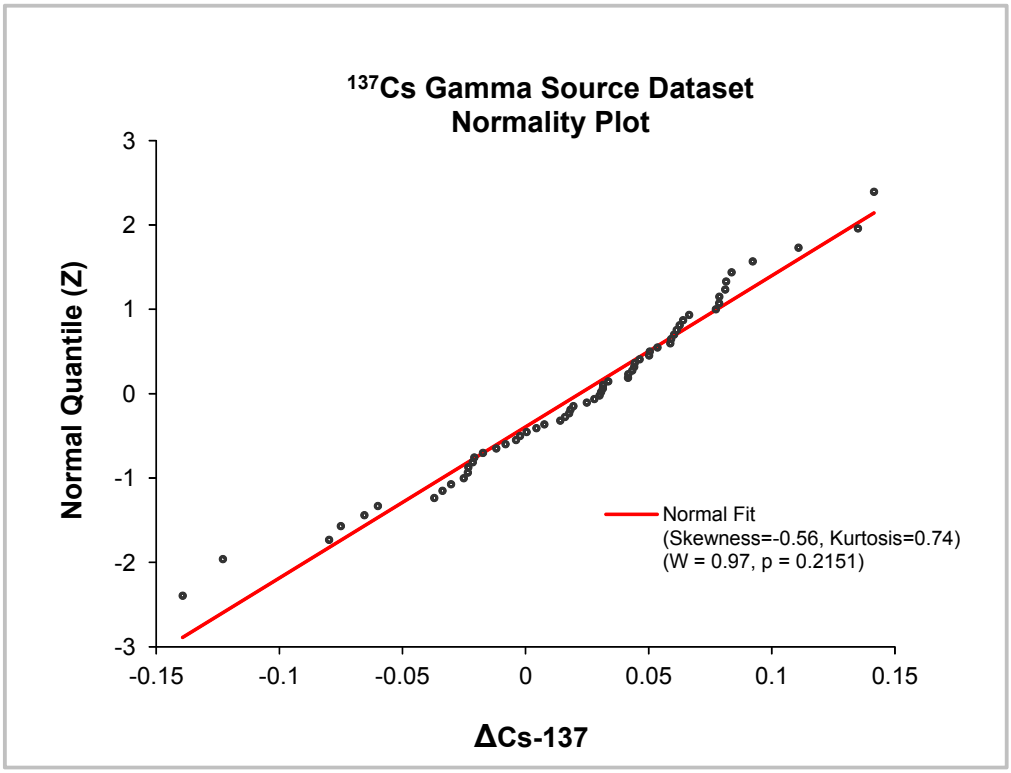

Figure 6-2
Normality Plot of ΔCs-137 for the Yankee Rowe Dataset

6.2 Neutron Radiation Source

The ^{242}Cm and ^{244}Cm RCA neutron source data are tabulated in Table 6-1 as the difference between measured and TRITON calculated, normalized by the measured value. The isotopic contents are expressed as the ratio of the number of ^{242}Cm or ^{244}Cm atoms to the number of ^{239}Pu atoms. The Yankee Rowe RCA data includes 12 assays of ^{242}Cm and 7 assays of ^{244}Cm, taken from five RCA samples. Two to four measurements (at two different laboratories) were made for each sample, but were averaged together for this report. The ^{242}Cm isotopic content is under-predicted by 21.3 percent at end-of-life and discharge from the reactor, but the importance of ^{242}Cm decreases rapidly at cooling times over one year. The ^{244}Cm isotopic content is over-predicted by 24.1 percent, which is more important for transport and storage with cooling times greater than one year. The over-prediction of ^{242}Cm of 24 percent is consistent with the conventional understanding that the neutron source is accurate to 30 percent. [14]

Table 6-1
Curium Neutron Radiation Source

Core Location	ΔCm-242 [Percent]	ΔCm-244 [Percent]
E5-SW-a5	51.1	-10.7
F5-SE-f5	24.7	-39.7
E6-NW-a1	11.9	-22.3
E6-NW-a1	15.6	-21.3
E6-SE-f5	3.1	-26.5
Mean	21.3	-24.1

Section 7: Conclusions

The Yankee Rowe RCA data represent a substantial contribution to the validation of the TRITON depletion computer code sequence for Actinide-Only Burnup Credit. The addition of so many RCA samples to the Nine-Reactor dataset enhances confidence in the validity of the data and the statistical results that are derived from the dataset, i.e., bias and uncertainty. The combined Nine-Reactor + Yankee Rowe dataset follows a normal distribution very well, so the statistical results are trustworthy.

The bias, standard deviation, and bias and uncertainty terms for the three RCA datasets are shown in Table 7-1. "Quality" is difficult to ascribe to RCA results, but the similarity of the Nine-Reactor and Yankee Rowe bias and standard deviation indicate that these two RCA datasets have similar quality.

Table 7-1
Isotopic Validation Results

Radiochemistry Assay Dataset	Bias ($\Delta k_\infty/k_{RCA}$)	Standard Deviation ($\Delta k_\infty/k_{RCA}$)	Bias and Uncertainty Term ($\Delta k_\infty/k_{RCA}$)
Nine-Reactor	0.006075	0.007100	0.01775
Yankee Rowe	0.003507	0.007348	0.01560
Combined Ten Reactor	0.004454	0.007433	0.01668

The Yankee Rowe radiochemical assay data includes ^{137}Cs, the dominant contributor to transport and storage cask gamma dose rates, and ^{242}Cm and ^{242}Cm, the dominant contributors to the neutron dose rates. The sixty (60) ^{137}Cs data points have a normal distribution and indicate that the accuracy of the gamma radiation source is about 10 percent, with a bias correction factor of 0.02189 (bias) + 1.645*0.0558 (standard deviation), or 0.1137. The Cm data were averaged into five ^{242}Cm and five ^{244}Cm data points and compared to the isotopic content calculated by TRITON. The mean of the data points indicates an under-prediction of ^{242}Cm of 21.3 and an over-prediction of ^{244}Cm by 24.1 percent, for a neutron source accuracy of about 30 percent.

Section 8: References

[1] "SCALE 6.0, "A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation for Workstations and Personal Computers," RSICC Code Package CCC-545, March 2000.

[2] M. D. DeHart, NUREG/CR-0200, Revision 7, Volume 2, Section F21, "TRITON – A TWO-DIMENSIONAL DEPLETION SEQUENCE FOR CHARACTERIZATION OF SPENT NUCLEAR FUEL", ORNL/NUREG/CSD-2/R7, January 2009.

[3] "MCNP/MCNPX Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," RSICC Code Package CCC-730, January 2006.

[4] DeHart, M.D. and Hermann, O.W. 1996. "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel". ORNL/TM-13317. Oak Ridge, Tennessee Oak Ridge National Laboratory, September, 1996.

[5] Rahimi M., Fuentes E., and Lancaster D., DOE/RW-0497, "Isotopic and Criticality Validation for PWR Actinide-only Burnup Credit," May 1997.

[6] Sanders, C. E., Gauld, I. C., "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor", NUREG/CR-6798, ORNL/TM-2001/259, Oak Ridge National Laboratory, Oak Ridge, Tennessee, May 2002.

[7] Hermann, O.W.; Bowman, S.M.; Brady, M.C.; and Parks, C.V. 1995. "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analyses", ORNL/TM-12667. Oak Ridge, Tennessee: Oak Ridge National Laboratory.

[8] Mochizuki, H., Suyama, K., Nomura, Y., and Okuno, H., "Spent Fuel Composition Database System on WWW - SFCOMPO on WWW Ver.2 -," JAERI-Data/Code 2001-020, Japan Atomic Energy Research Institute (August 2001).

[9] Scaglione, J.M., "Calculation of Isotopic Bias and Uncertainty for PWR Spent Nuclear Fuel". CAL DSU-NU-000001, Yucca Mountain Project.

[10] Chanaron, B. and Machiels, A., "Bugey and Gravelines Assay Measurement Data", Final Report 1016633, Electric Power Research Institute, December 2008.

[11] ANSI/ANS-8.27, "Burnup Credit for LWR Fuel", ANSI/ANS-8.27-2008, 555 North Kensington Avenue, La Grange Park, Illinois USA

[12] Graves H.W., Jr., Janz R. F., and Poncelet C. G., "The Nuclear Design of the Yankee Core", YAEC-136, Yankee Atomic Electric Power Company, February 1961, page 13.

[13] Nodvik R.J., et al., "Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium Through Curium", TID 4500, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, August 1969.

[14] Personal communication from Hermann O.W., ORNL, 1985.

Appendix A:Example Yankee Rowe Triton and MCNP Input

The TRITON input file shown here is for the E6-SE-e4-5 RCA sample from Yankee Rowe Assembly E6, SE subassembly, rod e4, axial elevation 57.66 cm from the bottom of the core.

EXHIBIT A-1: TRITON INPUT for E6SEe4-5

```
=t-depl parm=(centrm)
Yankee Rowe, Assembly E6, SE Section Pin e4 Height  57.66
' burnup=35.26 GWd/MTU
'
'************'
'  Library  *'
'***********'
'
238groupndf
'
'***********
'  Alias    '
'***********'
'
read alias
  $fuel 1-14 end
end alias
'
'***********************'
' Material Composition  '
'***********************'
'
read composition
'Fuel
'
'  fuel volume fraction decreased to smear inside the clad
(.747/.757)**2=.9738
'
'
  uo2  $fuel den=10.18 .9738 876 92234 0.020 92235 3.400 92238 96.58
end
```

```
  ss304  15 1   550  end
  h2o  16 den=0.7968  1  532   end
  arbm-bormod  0.7968 1 1 0 0 5000 100 16   1.0e-6  532  end
  h2o  66 den=0.9304  1  532   end
  arbm-bormod  0.9304 1 1 0 0 5000 100 66   1.0e-6  532  end
  zirc2  17 1   532  end
end composition
'
'********************'
' Cell Data          '
'********************'
'
read celldata
  latticecell squarepitch pitch=1.072 6 fueld=0.757 1 cladd=.864  5  end
end celldata
'
'*******************'
' Depletion         '
'*******************'
'
read depletion -1 2 3 4 5 6 7 8 9 10 11 12 13 14 end
end depletion
'
'
'********************
'* Timetable        *
'********************
'
read timetable
' this first spec. is for boron letdown (average boron conc. in mod.)
' 1.000 = boron conc. = 1 ppm
density 16 2 5010 5011
' Core I
     0.0    0.001
   451.5    0.001
' Core II
   580.5    0.001
   913.6    0.001
  1283.5    0.001
' Core IV
'  Short restart at 835 is incorporated into 707 burn.
  1283.6  707.
  1367.8  707.
  1367.9  337.
  1452.1  337.
  1452.2  0.001
  1620.7  0.001 end
end timetable
'
```

```
'*******************'
' Burn Data         '
'*******************'
'
read burndata
  power=33.128 burn=451.54  down=129.0   nlib=5 end
  power=33.604 burn=333.125 down=370.0   nlib=4 end
  power=27.021 burn=84.25  down=0.0     nlib=1 end
  power=27.021 burn=84.25  down=0.0     nlib=1 end
  power=27.021 burn=168.5  down=281.5   nlib=2 end
end burndata
'
'********************
'* Opus data        *
'********************
'
read opus
units=gram sort=no
symnuc=u-234 u-235 u-236 u-238
pu-238 pu-239 pu-240 pu-241 pu-242
np-237 am-241 am-242m am-243 cm-242 cm-243
cs-134 cs-137 mo-95 tc-99 ru-101 rh-103
ag-109 nd-143 nd-145 nd-148
sm-147 sm-149 sm-150 sm-151 sm-152
eu-151 eu-153 gd-155 end
matl=0 1 end
end opus
'
'*******************'
' NEWT Model        '
'*******************'
'
read model
YANKEE
read parm
  prtflux=no  drawit=yes echo=yes
  run=yes sn=6 inners=8 outers=200
 epsilon=1e-4 cmfd=yes xycmfd=3
end parm
read materials
  $fuel  1 ! fuel  !  end
  15     1 ! clad  !  end
  16     2 ! water !  end
  17     1 ! zirc  !  end
  66     2 ! vwater ! end
end materials
read geom
unit 1
 cylinder 10 0.3785
```

```
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  1 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 2
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  2 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 3
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  3 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 4
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  4 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 5
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  5 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 6
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  6 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 7
 cylinder 10 0.3785
 cylinder 20 0.432
```

```
 cuboid   30 4p0.536
 media  7 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 8
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  8 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 9
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media  9 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 10
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media 10 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 11
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media 11 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 12
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media 12 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 13
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
```

```
 media 13 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
unit 14
 cylinder 10 0.3785
 cylinder 20 0.432
 cuboid   30 4p0.536
 media 14 1 10
 media 15 1 20 -10
 media 16 1 30 -20 -10
 boundary 30
'  Empty Thimble Location
unit  22
 cuboid   30 4p0.536
 media 16 1 30
 boundary 30
'Vertical Zircaloy Control Blade Follower
unit 55
 cuboid 20  0.296 -0.296  0.5359 -0.5359
 cuboid 30 4p0.536
 media  17 1 20
 media  66 1 30 -20
 boundary 30 2 2
'Horizontal Zircaloy Control Blade Follower
unit 56
 cuboid 20  0.5359 -0.5359  0.296 -0.296
 cuboid 30 4p0.536
 media  17 1 20
 media  66 1 30 -20
 boundary 30 2 2
'Center Zircaloy Control Blade Follower
unit 57
  cuboid 20  0.5359 -0.5359  0.296 -0.296
  cuboid 22  0.296 -0.296  0.5359  0.296
  cuboid 24  0.296 -0.296  -0.296 -0.5359
  cuboid 30 4p0.536
  media  17 1 20
  media  17 1 22
  media  17 1 24
  media  16 1 30 -20 -22 -24
  boundary 30 2 2
global unit 21
 cuboid 10 4p13.4
 array 1 10 place 13 13 0.0 0.0
 media 16 1 10
 boundary 10 50 50
end geom
'
```

```
'
'  real assemblies have 305 fuel pins and 19 water pins.  In order to get
'  the thickness of the control rods correct the model has placed the
control
'  rod in the center of the assembly.  The control rod is 37 pin
locations.  To preserve
'  the water there should be (38/19)*305 fuel pins or 610 fuel pins.
Note that we still need
'  the instrumentation hole (making it 38 rather than 37).  The total
number of pins is 648.
'  The square root of 648 is 25.46.  Therefore a 25 by 25 assembly will
be modeled with the
'  control rod in the center.  One of the control blade holes furthest
from the sample locations
'  was removed to better preserve the water to fuel ratio.
'
'
read array
  ara=1  nux=25  nuy=25
   fill  2 2  2 2  2  2  2  2  2  2  2  2  2   2  2  2  2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2  2  2  2  2   2  2  2  2  2  2  2  2  2  2 2 2
         2 2 10 10 10 2  2  2  2  2  2  2  3   2  2  2  2  2  2  2  2  2  2 2 2
         2 2 10 22 10 2  2  2  2  2 11  3  55  3  2  2  2  2  2  2  2  2  2 2 2
         2 2 10 10 10 2  2  2  2  2 11  3  55  3  11 2  2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2 12 11  3  55  3  11 12 2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2 12 12 12 11  3  55  3  11 12 12 2  2  2  2  2  2 2 2
         2 2  2 2  2  2 12 12  8  9 11  3  55  3  11 9  8  9  2  2  2  2  2 2 2
         2 2  2 2  2  2  12 8  9  8 14  3  55  3  14 8  8  8  12 2  2  2  2 2 2
         2 2  2 2  2  12 12 9  8  7  4 13  55 13  4  7  8  9  12 12 2  2  2 2 2
         2 2  2 11 11 11 11 11 14 4  1  5  55  5  4  4  14 11 11 11 11 11 2 2 2
         2 2  2 3  3  3  3  3  3 13  5  6  55  6  5 13  3  3  3  3  3  3  2 2 2
         2 2  3 56 56 56 56 56 56 56 56 56 57 56 56 56 56 56 56 56 56 56  3 2 2
         2 2  2 3  3  3  3  3  3 13  5  6  55  6  5 13  3  3  3  3  3  3  2 2 2
         2 2  2 11 11 11 11 11 14 4  4  5  55  5  4  4  14 11 11 11 11 11 2 2 2
         2 2  2 2  2  12 12 9  8  7  4 13  55 13  4  7  8  9  12 12 2  2  2 2 2
         2 2  2 2  2  2  12 8  8  8 14  3  55  3  14 8  8  8  12 2  2  2  2 2 2
         2 2  2 2  2  2  2  9  8  9 11  3  55  3  11 9  8  9  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2 12 12 11  3  55  3  11 12 12 2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2 12 11  3  55  3  11 12 2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2  2 11  3  55  3  11 2  2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2  2  2  2  3   2  2  2  2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2  2  2  2  2   2  2  2  2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2  2  2  2  2   2  2  2  2  2  2  2  2  2  2 2 2
         2 2  2 2  2  2  2  2  2  2  2  2  2   2  2  2  2  2  2  2  2  2  2 2 2
end fill
 end array
read bounds
  all=reflect
end bounds
```

```
end model
end
```

EXHIBIT A-2: MCNP INPUT for E6SEe4-5 with Measured RCA Isotopics

```
Yankee Rowe, Assembly E6, SE Section Pin e4 Height  57.66
C     NEW Measured Isotopics, Burnup 35.26 GWd/MTU
C     E6-SE-E4  57.66 cm
C     FUEL ROD
1     6270  -10.018 -1              IMP:N=1   U=2 $ UO2 PELLET
3     2     -6.4  1 -3              IMP:N=1   U=2 $ ZIRCONIUM
4     3     -1.0   3                IMP:N=1   U=2 $ WATER
C     WATER ROD
5     3     -1.0   -3               IMP:N=1   U=3 $ water cell
6     3     -1.0   3                IMP:N=1   U=3 $ WATER CELL
C
C     Zirc Follower
10    2     -6.4 -51 52 -53 54      IMP:N=1   U=8 $ Vertical Zr Cell
11    3     -1.0 -55 56 -57 58 (51:-52:53:-54)   IMP:N=1   U=8
C
12    2     -6.4 -61 62 -63 64      IMP:N=1   U=9 $ Horizontal Zr Cell
13    3     -1.0 -65 66 -67 68 (61:-62:63:-64)   IMP:N=1   U=9
C
20    0             -10 11 -12 13 30 -31  IMP:N=1 LAT=1 U=1
      FILL= -13:13 -13:13 0:0
     3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 9 9 9 9 9 9 9 9 9 9 9 9 9 9 9 9 9 9 9 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 3
```

```
     3 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 8 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3
     3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
C
22    0     -16 17 -18 19 14 -15  IMP:N=1  FILL=1  $ FUEL ASBL
C     REFLECTOR
C     OUTSIDE WORLD
30    0     (16:-17:18:-19:-14:15)   IMP:N=0  $ VOID

C     SURFACE SPECIFICATIONS
1    CZ      0.3785       $ UO2 PELLET  W15 is 0.4096
C    Pellet/Clad gap is not modeled in TRITON
3    CZ      0.432      $ ZIRC OUTER RADIUS  W15 is 0.4750
10   PX      0.536      $
11   PX     -0.536       $ ACTUAL CELL BOUNDARIES
12   PY      0.536       $  W15 is 0.630
13   PY     -0.536       $
14*   PZ     -182.9        $ BOTTOM END OF FUEL
15*   PZ      182.9        $ TOP END OF FUEL
16*   PX      13.400       $
17*   PX     -13.400       $ FUEL ASSEMBLY
18*   PY      13.400       $ BOUNDARIES
19*   PY     -13.400       $
30   PZ     -182.92       $ END OF FUEL FOR FILL
31   PZ      182.92       $ TOP END FOR FILL
C    Zircaloy control blade follower
51   PX      0.296      $ Vertical Zirc
52   PX     -0.296
53   PY      0.5359
54   PY     -0.5359
C
55   PX      0.6      $ Vertical Zirc
56   PX     -0.6
57   PY      0.6
58   PY     -0.6
C
C
C
61   PY      0.296    $ Horizontal Zirc
62   PY     -0.296
63   PX      0.5359
64   PX     -0.5359
C
65   PX      0.6      $ Horizontal Zirc
66   PX     -0.6
67   PY      0.6
```

```
68   PY     -0.6
C
C
C

MODE  N
KCODE 5000 1 400 4400 10000
KSRC  1.072  0 0 -1.072 0 0 0 1.072 0 0 -1.072 0
      1.072 1.072 0 -1.072 1.072 0 1.072 -1.072 0 -1.072 -1.072 0
      2.144  0 0 -2.144 0 0 0 2.144 0 0 -2.144 0
C
M2   40000.60C  1.0                                  $ ZIRCONIUM
M3   1001.60C   0.667   8016.60C   0.333             $ WATER
MT3  LWTR.60T
C
M6270
        8016.50C        -11.85
        92234.50C       -0.0110
        92235.50C       -0.9038
        92236.50C       -0.4076
        92238.50C       -85.7757
        94239.50C       -0.6269
        94240.50C       -0.2272
        94241.50C       -0.1476
        94242.50C       -0.0482
C
C
C
C
C
PRINT
```

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