WCAP-6068

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WCAP-6068
UC-80, Reactor Technology

EVALUATION OF MASS SPECTROMETRIC AND
RADIOCHEMICAL ANALYSES OF YANKEE CORE I
SPENT FUEL

by

R. J. Nodvik
Nuclear Operations Analysis

March 1966

Prepared for the New York Operations Office
U. S. Atomic Energy Commission
Under AEC Contract Number AT(30-1)-3017

TECHNICAL APPROVAL:

H. W. Graves, Jr., Manager
Nuclear Engineering Section
Reactor Development

PROJECT MANAGER APPROVAL:

for L. Chajson, Manager
Yankee Core Evaluation Program

WESTINGHOUSE ELECTRIC CORPORATION
Atomic Power Division
Box 355
Pittsburgh, Pennsylvania 15230

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ABSTRACT

This report presents the results of mass spectrometric, X-ray spectrographic, and radiochemical analyses of spent fuel samples from 219 locations in the Yankee core. These sample locations were pre-selected to provide: 1) the U and Pu isotopic composition of the fuel as a function of burnup in the asymptotic and perturbed reactor neutron spectra; 2) the spatial distribution of burnup and fuel isotopes in the rods, in the assemblies, and in the core; 3) the total isotopic inventory of the core; and 4) fuel characteristics, including the specific Pu production, the effective capture-to-fission ratio in U-235, and the net mass of fissile materials destroyed per unit of energy release in the fuel.

Values of burnup are inferred over a broad range (1,200 to 31,000 MWD/MTU) from relationships between U and Pu concentrations measured before and after irradiation, and from the activities of the Cs-137 and Sr-90 fission products. The calculations used to infer burnup and the various fuel characteristics from the spent fuel data are described in detail.

The consistency and reliability of the data are established through the evaluation of the experimental results obtained from a number of inter-laboratory cross-check and monitor sample solutions.

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1.0 INTRODUCTION

This report presents the mass spectrometric, the X-ray spectrographic, and the radiochemical data from a large number of spent Yankee fuel samples which were analyzed under Phase I and Phase II of the Yankee Core Evaluation (EYC) Program.^{/1} The bases for selecting these samples are fully described, and the measured data are thoroughly evaluated with respect to fuel depletion, plutonium production, and other important nuclear fuel characteristics.

Under the EYC Program, 56 fuel rods were removed from 14 Core I fuel assemblies (Phase I), and 7 fuel rods were removed from one of two Core I assemblies recycled in Core II for extended burnup (Phase II). These fuel rods were subjected to extensive post-irradiation examinations, destructive and non-destructive. Non-destructive examinations consisted of visual inspections, dimensional checks, and gamma-activity measurements.^{/2} Destructive examinations included: chemical defilming of cladding surfaces; fission gas collection and analysis; hardness, burst, and tensile testing of cladding specimens; and X-ray spectrographic, mass spectrometric, and radiochemical analyses of spent fuel and, in some instances, cladding.

As related to this report, Phase I represents the analyses of pellet-size samples from 191 pre-selected locations within Yankee Core I, spanning a range of burnups from 1,300 to 18,000 megawatt days per metric tonne of initial uranium loading (MWD/MTU), and Phase II represents the analyses of pellet-size samples from 28 locations in the extended burnup Core I fuel assembly. The Phase II samples range from 10,000 to 31,000 MWD/MTU.

Phase III of the 5-year EYC Program consists of similar post-irradiation examinations which are being performed on a Core I assembly that was cycled through Cores I, II, and IV to achieve an estimated peak burnup of 46,000 MWD/MTU. Phase III results will be reported in late 1966.

This report is organized so that a brief description of the Yankee core is given in the next section. The EYC sampling plan is presented in Section 3, the origin of the spent fuel data is given in Section 4, and the fuel pellet burnup and a comparison of several methods used to obtain the burnup are given in Section 5. An evaluation of the isotopic composition data, in-

cluding plutonium production, is presented in Section 6. Some fuel characteristics inferred from the isotopic data are discussed in Section 7; these include the effective capture-to-fission ratio in U-235 (α^{25}) and the net mass of fissile material destroyed per unit of energy release. In Section 8, the uncertainties in the spent fuel data are discussed. The conclusions reached in this study are stated in the final section.

Several appendices have been included at the end of this report. Appendix A1 contains a listing of the mass spectrometric and radiochemical data obtained for each sample; included are the U and Pu isotopic composition of the sample solution, and the Cs-137 and Sr-90 activity. In Appendix A2, a detailed description is given of the data reduction processes used in the program. Some fuel characteristics, inferred from the results of the spent fuel analyses, are tabulated in Appendix A3; these include the data used in Sections 5, 6, and 7.

Most of the pellet data appearing in this report has been utilized to infer the assembly and core-total U and Pu isotopic inventory and burnup. A complete description of the assembly and core-total results can be found in topical report WCAP-6071.¹³

2.0 DESCRIPTION OF THE YANKEE CORE

Core I of the Yankee power reactor^{/4,5} consisted of 76 fuel assemblies, each loaded with either 304 or 305 fuel rods arranged in a square lattice. In each assembly one fuel rod, located at approximately the center of the assembly, was omitted from the lattice to provide a space for the insertion of a flux wire thimble. (Only 22 assemblies subsequently were equipped with thimbles.) The total number of fuel rods in the core was 23,142. A general description of the first core is given in Table 2.0.1.

The second Yankee core was nominally identical to Core I except for the presence of two spent Core I fuel assemblies located in the central region of the core, and a modified control rod design. With respect to Core I, Core II was operated at a higher power level (485 versus 392 MWt) and for a shorter burnup (7,870 versus 8,470 MWD/MTU). The final shutdown of Core I occurred May 18, 1962, while the final shutdown of Core II occurred September 2, 1963.

An accurate record exists^{/21} of the total energy release in Yankee Cores I and II based on the calorimetric measurements obtained during reactor operation. A plot of the reactor power versus time for the periods during which Cores I and II were operated at power is shown on Figure 2.0.1. These periods extend from January 1961 until May 1962 for Core I, and from September 1962 until September 1963 for Core II. In addition, some low power physics and startup tests^{/29} were made on Core I between November 1960 and January 1961. The relatively small amount of energy released during those tests is included in the total for Core I and in the equivalent full power hours (EFPH), shown as a dotted line on Figure 2.0.1. The reference thermal power used in calculating the total core EFPH is 392 MWt for Core I and 485 MWt for Core II.

The basic component of both "batch-loaded" cores was a stainless steel tube filled with approximately 150 slightly enriched UO_2 pellets, each with a nominal length of 0.6 inches. The average height of the UO_2 columns was 90.57 inches with a cross sectional area of 0.0590 square inches. In addition to the UO_2 pellets, each fuel rod contained stainless steel spacer discs of total height of 0.61 inches and axial expansion voids of a total height

TABLE 2.0.1

Yankee Core I Description and Operating SummaryCore Data

Uranium Fuel Loading (kg)	20,908 ^{/a}
Fuel Enrichment (w/o U-235)	3.4
Average Core Diameter (inches)	75.1
Active Core Height (inches)	91.9
Number of Ag-In-Cd Control Rods	24
Number of Zr Shims	8
Nominal System Pressure (psi)	2000
Nominal Main Coolant Temperature (°F)	514
Number of Fuel Assemblies	76
Number of Fuel Tubes	23,142
Fuel Tube Pitch (inches)	0.422 and 0.456
Fuel Tube Outside Diameter (inches)	0.340
Fuel Tube S.S. Cladding Thickness (inches)	0.021
Fuel Pellet Diameter (inches)	0.294
Unit Cell W/U	2.67

Operating Data

Initial Criticality	8/19/60
Initial Operation at Rated Power	1/16/61
Initial Commercial Operation	7/1/61
Final Shutdown	5/18/62
Average Main Coolant Temperature (°F) ^{/b}	504
Average Main Coolant Temperature Rise (°F) ^{/b}	31
Nominal Core Power Density (kw/kg U)	18.7
Nominal Power Output (MW thermal)	392
Maximum Power During Gen. (MW thermal)	485
Average Power During Gen. (MW thermal) ^{/b}	~321

TABLE 2.0.1 (continued)

Gross Energy (MWH electric)	1,330,000
Net Energy (MWH electric)	1,223,000
Gross Energy (MWH thermal)	4,250,000
Hours Critical (hours)	13,247
EFPH (hours at 392 MWt)	10,840
Core Average Burnup (MWD/MTU) ^{/c}	8,470 ^{/d}

Notes a) Determined from gravimetric measurements of pre-irradiated fuel.

b) Includes core extension at reduced power level and reduced main coolant temperature.

c) Units are megawatt days per metric tonne of initial uranium loading.

d) Determined from plant calorimetric data at Yankee-Rowe site.

Reactor Power and Equivalent Full Power Hours of Yankee Cores I and II

Figure 2.0.1

of 0.71 inches, adding up to an active core height of 91.89 inches from the bottom of the lowest UO_2 pellet to the top of the highest pellet.

The arrangement of the fuel assemblies in the core is shown in Figure 2.0.2. Also shown on the figure are the active control rods (solid black) which at some time during the operation of Core I (and Core II) presented their Ag-In-Cd absorbing sections to the neutron flux. The cross checked outer ring of control rod positions was filled with Zircaloy during power operation. (Half of these positions contain movable Group 6 control rods, while the other half contain cruciform shaped Zircaloy shim pieces.) On the periphery of the core are shown the stainless steel fillers, displacing water from the recesses in the fuel assemblies. The alpha-numeric matrix notation used to describe the core location of a particular fuel assembly is also shown on the figure.

The normal control rod withdrawal sequence 6,7 in Yankee Core I was 642351; the inverted withdrawal sequence, with Control Rod Groups 4, 3 and 1, 2 interchanged, was 631452. Thus, with respect to the fuel loading and the location of the control rod groups, the core possessed quarter-core symmetry about the E-W and N-S axes of the core. With respect to the normal and inverted control rod withdrawal sequence, the core possessed eighth-core symmetry about the SW-NE 45° diagonal axis to the degree of equivalent control rod interchange.

CORE LOCATION OF FUEL ASSEMBLIES AND
CONTROL ROD GROUPS IN THE YANKEE REACTOR

Figure 2.0.2

3.0 SELECTION OF SPENT FUEL SAMPLES FOR THE YANKEE CORE EVALUATION PROGRAM

The large size of the Yankee core, and the great number of fuel pellets (3.5×10^6) available for analysis suggested the use of a statistical method in the selection of an optimum number of samples for post-irradiation analysis. However, because of the nature of the information sought, and the availability of a large amount of data (such as burnup distributions from in-core instrumentation measurements) useful in the selection of samples, it became apparent after an initial investigation that the experimental program for the spent core evaluation was not directly suited to a statistical design. Nevertheless, in establishing the original sampling plan, effective use was made of the general principles which underlie a statistical experimental design.

Due to unforeseen difficulties and delays^{/8} encountered in the acquisition and analysis of the spent fuel samples, the original EYC sampling plan^{/9} was modified several times. These revisions^{/8} affected both the location and the total number of pellets removed from the core. Early in the program, some discrepancies in the results of mass spectrometric analyses were uncovered through the use of cross check samples. In the intermediate and later stages of the program, repeated analyses of a monitor sample solution were obtained to provide a check on the consistency and precision^{/1} of the spent fuel data. To offset the increase in the number of analyses required, and to remain within the original program funding, the number of core locations from which pellets were removed was decreased. Furthermore, as data were acquired from certain regions of the core, it became apparent that additional samples were not necessary to develop the information sought in these regions.

The number of fuel rods finally selected for analysis from Yankee Core I and Core II are summarized in Table 3.0.1. The remainder of this section presents the general objectives employed in selecting fuel samples and the exact core locations chosen.

3.1 General Objectives in the Selection of Spent Fuel Samples

In the interests of the advancement of competitive atomic power, the Yankee Core Evaluation Program has the following broad objectives:

TABLE 3.0.1
Number of Spent Fuel Rods and Number of Core Locations Sampled in EYC Phases I and II

	P H A S E I					P H A S E I I				
	Neutron Spectrum			Total No. of Fuel Rods	No. of Core Locations Sampled	Neutron Spectrum ^{/c}			Total No. of Fuel Rods	No. of Core Locations Sampled
	Asymptotic (Center)	Intermediate ^{/a} (Diagonal)	Perturbed ^{/b} (Corner)			Asymptotic (Center)	Intermediate (Diagonal)	Perturbed (Corner)		
6-Pellet Rod ^{/d}	4		1	5	30	1		1	2	12
4-Pellet Rod	20	8	9	37	148	1	2	1	4	16
2-Pellet Rod	4	1		5	10					
1-Pellet Rod	3			3	3					
0-Pellet Rod	3	3		6			1		1	
Total No. of Rods Removed from Core	34	12	10	56		2	3	2	7	
Total No. of Rods Sectioned	31	9	10	50		2	2	2	6	
Total No. of Core Locations Sampled	115	34	42		191	10	8	10		28

- Notes: a) Includes two 4-pellet and one 2-pellet fuel rods on the diagonal approaching the core reflector.
b) Includes one 4-pellet fuel rod immediately adjacent to the reflector.
c) All fuel rods removed from assembly #A21, core location F5.
d) Includes a nominal 4-pellet rod from which two additional pellets were cut from zone 3.

1. the evaluation of the performance of Core I in the Yankee plant,
2. the comparison of actual performance with predicted performance, and
3. the revision of design methods in an effort to reconcile differences between predicted and actual performance.

To achieve the broad objectives outlined above, the objectives in the selection of the spent fuel samples for the physics portion of Phase I of the EYC Program were to determine:

1. the uranium and plutonium isotopic composition of the fuel as a function of burnup in the asymptotic and perturbed reactor neutron spectra,
2. the spatial distribution of burnup and fuel composition in the rods, in the assemblies, and in the entire core,
3. the total uranium and plutonium isotopic inventory of Core I, and
4. the fuel characteristics, including the capture-to-fission ratio in U-235, plutonium production, and the net mass of fissile material destroyed per unit of energy release in the fuel.

Consistent with the objectives listed above was the desire to analyze samples which covered as broad and as complete a range of burnup as possible, and which defined the spatial variations adequately. The analyses of samples from the Phase II fuel assembly doubled the range of burnup available from the Phase I assemblies.

As used in the preceding paragraphs, the perturbed neutron spectra refer to those fuel regions in the vicinity of the water slots which surround the cruciform control rod positions. When the absorber sections of a particular control rod group are in the core, the thermal neutron flux in the vicinity of the control rods is depressed. When the control rods are withdrawn from the core and replaced by the Zircaloy control rod follower sections, the excess moderation due to the water slots causes a peaking in the thermal flux. These perturbations affect only those fuel rods which are within 2 or 3 lattice pitches of the water slots. Also included in the regions of perturbed neutron spectra are those fuel rods which occupy positions near the core reflector.

The perturbed neutron spectra also refer to those fuel pellets which are adjacent to the stainless steel spacer discs and axial expansion voids which

separate stacks of pellets within a fuel rod, and also to those fuel pellets which are in the neighborhood of the stainless steel ferrules used to braze together the individual fuel rods to form a fuel subassembly. (Each fuel assembly consists of 9 subassemblies, in a 3 x 3 array.) The spacer discs and axial voids cause an increase in the water-to-uranium ratio, thus producing a peak in the local neutron flux. The ferrules, which are in the coolant channels, displace water, thereby decreasing the water-to-uranium ratio and causing a depression in the local flux.

The so-called asymptotic neutron spectrum refers to those regions of the core which are well away from and unaffected by the perturbations described above (i.e., where the fast-to-thermal flux ratio is constant).

3.2 Selection of the Fuel Rods

Because of the geometrically symmetric placement of fuel assemblies and control rod groups in the core, a full core distribution could be obtained by sampling one quadrant of the core. To utilize the flux wire data, and to ultimately compare³ the burnup distribution inferred from time integrated flux wire activation data with the burnup distributions inferred from the results of destructive analysis, the instrumented quadrant of the core (NE in Figure 2.0.2) was chosen to be sampled. Due to the frequent control rod interchanges which occurred during reactor operation, the core possessed some degree of eighth-core symmetry. This is evident in Figure 3.2.1 which shows the normalized burnup distribution obtained with the Yankee in-core instrumentation system. A description of the Yankee in-core instrumentation and the Yankee Core I burnup distribution is given in Reference 10. The data shown in Figure 3.2.1 are slightly different than those given in the reference, due to some recent revisions made in the computer code which is used to reduce the flux wire data. The degree of eighth-core and quarter-core burnup asymmetry evident in Figure 3.2.1 is of the order of the accuracy in the flux wire data³.

Thus, it was found that eleven assemblies would define the complete core burnup and isotopic materials distribution. The location of the eleven assemblies is shown in Figure 3.2.2, along with the location of three additional assemblies chosen to check the validity of quarter-core symmetry.

NOTE: ALL DATA ARE NORMALIZED
TO THE QUADRANT AVERAGE.

YANKEE CORE I NORMALIZED BURNUP DISTRIBUTION
INFERRED FROM IN-CORE FLUX WIRE MEASUREMENTS

Figure 3.2.1

Note: 1) Those rods indicated by open circles were sectioned for isotopic and radiochemical samples.

2) All Phase II fuel rods from Assembly F5.

CORE LOCATION OF EYC PHASE I AND PHASE II
FUEL RODS SELECTED FOR POST-IRRADIATION ANALYSES

Figure 3.2.2

To obtain data on fuel located in the reactor's asymptotic neutron energy spectrum, at least two fuel rods were selected from the center subassembly of each assembly examined. More than two fuel rods were chosen from those assemblies of high burnup, F⁴ and G⁴, and from the assembly which had experienced a burnup approximately equivalent to the average core burnup, H³. To facilitate the removal of fuel rods, and to avoid the influence of the local flux perturbation caused by the flux-wire thimble, the corner fuel rods of the central subassembly were chosen to represent the asymptotic neutron spectrum.

To select fuel rods from the regions of perturbed neutron spectra, use was made of the data in Table 3.2.1. Based on the data in this table, the SW subassembly of Assembly F⁴ was selected since it contained fuel rods which experienced the longest irradiation adjacent to a control rod absorber section (Assembly F⁵ was cycled in Core II). The SE subassembly of Assembly H³ was also selected since it contained fuel rods which were adjacent to a control rod Zircaloy follower section throughout the entire irradiation. Finally, to represent the neutron spectrum in the vicinity of the water reflector, the NE subassembly of Assembly J³ was selected. To provide a check on the experimental data, and to obtain additional corner-to-center local peaking information, the corner fuel rods in Assemblies E⁵, G⁴, H⁵, and J⁴ were selected.

The core locations of the 56 Phase I fuel rods are shown in Figure 3.2.2. Also shown in the figure are the core locations (all in Assembly F⁵) of the 7 Phase II fuel rods. Those fuel rods represented by the open circles were samples to provide the data which appear later in this report. The 7 fuel rods represented by the darkened circles were subjected only to visual examination and certain mechanical testing.

3.3 Selection of Fuel Samples Within a Fuel Rod

Analysis of flux wire manganese activation data obtained during core life revealed that the axial flux distribution in the core was perturbed by the presence of ferrules, axial voids, and spacer discs, as discussed in Section 3.1. Since the number of fuel pellets affected by these perturbations is small compared to the total number of pellets in the core, it was felt that samples should not be taken from these regions. Samples more representative of the

TABLE 3.2.1

Irradiation History for Selected Fuel Subassemblies

Fuel Assembly	Subassembly	Days of Full Power ^a Operation Next To a Follower	Days of Full Power Operation Next To a Control Rod	Days of Full Power Operation Next To a Moving Rod
F4	SW	269	137	39
E5	NE	269	137	39
F4	NE	397	0	48
G4	NW	397	0	48
H3	SE	445	0	0
J4	NW	445	0	0

Note a) The table shows the approximate number of days of full power (392 MWt) operation next to a follower or control rod for fuel rods located in the corner of fuel assemblies.

asymptotic core condition could be obtained from positions along the fuel rod stack, midway between the local perturbations described above.

A typical axial flux wire trace obtained during Core I operation is shown in Figure 3.3.1. The peaks and depressions along the activation distribution are characteristic of all the flux wire traces in all assemblies. The spacer discs and voids are located at nominal 16-inch intervals along the fuel rod pellet stack, while the ferrules occur at nominal 8-inch intervals along the rod. Consequently, 6 relatively unaffected axial regions or zones, bracketed by consecutive pairs of ferrules, are available for sampling. The 6 axial sample zones, and the location of the positions which were sampled, are shown in the sketch on Figure 3.3.2. To obtain all the samples in each axial sample zone from the same x-y plane, the measurements shown on Figure 3.3.2 are from a reference point on that end of the fuel rod which was in the bottom of the core during reactor operation.

The axial location information is summarized in Table 3.3.1. The normalized weighting factors listed in the table have been used to obtain fuel rod average data by volume weighting the pellet data. It is emphasized that the dimensional data and the weighting factors are nominal, and represent the average conditions in the core for the following reasons:

1. due to manufacturing tolerances on the fuel pellets, not all fuel rods contained exactly the same number of pellets, and
2. during fabrication of the fuel assemblies, fewer handling fixtures were required if some types of subassemblies were inserted in the assembly in an inverted position.

The result of these fabrication procedures is a slight non-uniformity in the axial locations of the spacer discs and ferrules; that is, with respect to the bottom of the fuel assembly, the locations of the spacer discs (and axial voids) and ferrules are staggered, and are not all in the same x-y plane in each axial sample zone. These conditions are obvious in the photographs of some spent fuel assemblies shown, for example, on pages 340.3 and 340.4 of Reference 11.

Typical Yankee Flux Wire Activity Trace

Figure 3.3.1

AXIAL LOCATION OF SPENT FUEL SAMPLES WITHIN A FUEL ROD

TABLE 3.3.1

Axial Location of Spent Fuel Samples
Within a Fuel Rod

Type of End Plug	Axial Sample Zone	Distance From Reference Point to Sample (inches)	Length of Pellet Stack Per Sample Zone (inches)	Volume Weighting Factor
Male or Chamfered	1	86.7	13.30	0.1470
	2	70.7	15.77	0.1741
	3	54.7	15.77	0.1741
	4	38.7	15.77	0.1741
	5	22.7	15.77	0.1741
	6	6.7	<u>14.18</u>	<u>0.1566</u>
Total			90.56	1.0000
Female	1	87.1	13.30	0.1470
	2	71.1	15.77	0.1741
	3	55.1	15.77	0.1741
	4	39.1	15.77	0.1741
	5	23.1	15.77	0.1741
	6	7.1	<u>14.18</u>	<u>0.1566</u>
Total			90.56	1.0000

The validity of the volume weighting factors in Table 3.3.1 has been demonstrated for the gamma activity data obtained in Subtask EYC-354. The conclusion drawn from the demonstration, which is shown in Appendix A3 of Reference 3, is that the effects of the local flux peaks and depressions, caused by the spacer discs and ferrules, are compensating and tend to cancel in the volume weighting process. Single-channel gamma scans, showing the axial gross gamma activity distributions above 0.5 and 0.05 Mev in a typical Yankee spent fuel rod², are presented in Figure 3.3.3.

Axial Gross Gamma Activity Distribution in a Typical Yankee Spent Fuel Rod

Figure 3.3.3

4.0 SOURCE OF EYC DATA

After a cooling period of approximately 10 months following the shutdown of Yankee Core I, the 14 depleted fuel assemblies selected for sampling were shipped^{/12} from the Yankee site at Rowe, Massachusetts to the Transfer Building at the Westinghouse Post Irradiation Facility (WPIF) at Waltz Mill, Pennsylvania. The pre-selected fuel rods were removed intact from the assemblies with the fuel rod hole-saw, described in Reference 11, and transferred in a shielded cask, via a canal, to the WPIF Hot Cell Laboratory. After performing dimensional, gamma scanning, and fission gas measurements, the fuel rods were cut with an abrasive wheel at the locations given in Table 3.3.1, and up to 6 pellet-size sections of cladding and fuel were removed from each fuel rod.

The fuel was separated from the cladding by dissolution in nitric acid, and sample solutions were prepared by pipetting aliquots of the fuel solution into water. Early in the program, the final acid concentration of the sample solution was approximately 0.01 M HNO_3 . Sample solutions submitted for mass spectrometric analysis at this low acidity are referred to as "dilute" acid samples. An evaluation of the mass spectrometric results obtained for some cross check samples among the approximately 45 dilute acid samples indicated the possibility that Pu was plating out on the sides of the shipping bottles. Consequently, 2 M HNO_3 was employed for the remaining dilutions. The samples submitted for analysis at the higher acidity are referred to as "concentrated" acid samples.

The acidity of the sample solution does not affect the relative isotopic composition results since the plutonium isotopes do not plate out preferentially, however, the absolute values for plutonium concentrations are affected. The results of radiochemical analyses from Cs-137 and Sr-90 also appear to be affected by the sample solution acidity.

The following three sections give brief descriptions of the mass spectrometric, X-ray spectrographic, and radiochemical analyses performed on the spent fuel samples. In Appendix A1, all the reactor physics data obtained in Phase I and Phase II are tabulated.

4.1 Mass Spectrometric Analysis

Utilizing the technique of isotopic dilution, the mass spectrometric analyses of the Yankee fuel were performed by two subcontractors. These are:

1. TRACERLAB, Reactor Monitoring Center, A Division of Laboratory for Electronics, Inc., 2030 Wright Avenue, Richmond, California, under Messrs. Buddy L. Brock, and Leon Leventhal.
2. Vallecitos Atomic Laboratory, General Electric Atomic Products Division, P. O. Box 846, Pleasanton, California, under Messrs. J. F. Hanus, J. F. Davies, and C. P. Ruiz.

Several cross check analyses were performed by the USAEC New Brunswick Laboratories (NB) at New Brunswick, New Jersey.

The following paragraphs outline the procedure used by the Vallecitos Atomic Laboratories (GE) in the analyses of the EYC samples*.

A U-233 solution is used to spike the uranium in the samples. This is calibrated against a solution of natural uranium prepared from the NBS gravimetric standard U_3O_8 (NBS #950).

The plutonium in the samples is spiked with Pu-242. This contains lighter Pu isotopes and a separate mass spectrometric analysis of the sample plutonium is necessary to determine its isotopic composition. The Pu-242 spike is calibrated against a plutonium solution prepared from a gravimetric standard (NBS #949).

Chemical equilibrium between spike and sample Pu is affected by reducing to the III state with Fe II ions and hydrazine. Excess reducing agent is removed and the Pu is oxidized to Pu IV by addition of excess nitrite solution. The plutonium is separated from uranium ion and most fission products by absorbing it onto a Dowex 1 anion exchange column in the nitrate form and washing with 3 M HNO_3 . The Pu is finally eluted with 1 M HCl.

The uranium fraction from the nitrate column is converted to chloride and adsorbed from concentrated HCl onto a Dowex 1 anion exchange in the

*Outline obtained from W. W. Sabol, Manager of Analytical Chemistry, GE-APED.

chloride form. Washing with 6 N HCl removes most fission products and the uranium is finally eluted with 0.5 N HCl.

Both the plutonium and uranium are converted to nitrate for loading onto mass spectrometer filaments. The analysis is made on the metal ion beams.

The TRACERLAB laboratories (TL) utilize a solvent extraction separation scheme with U-233 and Pu-236 tracers. The spiking and separation technique can be found in Reference 13. After addition of U-233 and Pu-236 spikes to the sample solution, nitric acid and K Br O_4 are added to oxidize Pu to PuO_2^{++} . Uranyl and plutonyl ions are then extracted into hexone and stripped into water. Plutonium is then extracted into TTA (thenoyltrifluoroacetone) and plated onto a platinum disc for α pulse height analysis, and the sample is prepared for mass spectrometric analysis.

The residual uranium is extracted into diethyl ether and prepared for mass spectrometric analysis.

The TL technique requires the preparation of one sample for the determination of plutonium; the GE technique necessitates the preparation of two samples for the plutonium determination, one with spike and one without.

A numerical breakdown of the dilute and concentrated acid sample analyses performed by the subcontractors is given in Table 4.1.1. In addition to the 166 mass spectrometric analyses on Phase I and Phase II full-pellet samples, TRACERLAB provided analyses on 10 Phase I and 5 Phase II microsamples. The data for these microsamples, obtained along the radius of each of 3 individual pellet cross sections, are reported in Subtask EYC-234 of References 14 and 15. The reduced microsample data is evaluated in Reference 16.

4.2 X-ray Spectrographic Analysis

The results of the cross check samples submitted to the mass spectrometric laboratories of TRACERLAB and General Electric revealed an interlaboratory discrepancy in the Pu/U mass ratio of the early analyses. These discrepancies were resolved by the laboratories through their evaluations of exchanged U and Pu spiking solutions, which resulted in re-calibration of the plutonium spike by TL and the uranium spike by GE.

TABLE 4.1.1

Summary of Total Number of EYC Phase I and Phase II Mass Spectrometric Analyses

LABORATORY	NO. OF ANALYSES (Submitted in Dilute Acid Solution)					NO. OF ANALYSES (Submitted in Concentrated Acid Solution)					TOTAL ANALYSES, DILUTE AND CONC. ACID
	UNIRRADIATED	FULL PELLET	CROSS CHECK	FULL ROD COMPOSITE	TOTAL DILUTE ACID ANALYSES	FULL PELLET	CROSS CHECK	MONITOR	FULL ROD COMPOSITE	TOTAL CONC. ACID ANALYSES	
<u>Phase I</u>											
New Brunswick (NB)							21/a	2/b		23/c	23/c
TRACERLAB (TL)	1	26	2	1	30	111	7	7		125	155
General Electric - Vallecitos (GE)	1	13	4		18	40	6	4	1	51	69
Total Phase I Anal., TL + GE	2	39	6	1	48	151	13	11	1	176	224
Total Phase I Anal., NB + TL + GE	2	39	6	1	48	151	34	13	1	199	247
<u>Phase II</u>											
TRACERLAB						12				12	12
General Electric - Vallecitos						16				16	16
Total Phase II Anal., TL + GE						28				28	28
Total Phase I + Phase II Analyses, TL + GE										204	252
Total Phase I + Phase II Analyses, NB + TL + GE										227	275/d

Notes: a) Includes 9 single analyses and 6 pairs of duplicate analyses.

b) Includes one pair of duplicate analyses of a monitor sample solution.

c) Includes 9 single analyses and 7 pairs of duplicate analyses.

d) In addition, TL performed analyses on 15 radial microsamples (10 Phase I and 5 Phase II, see Reference 16), resulting in a grand total of 290 mass spectrometric analyses performed in Phases I and II of the EYC program.

At the same time, an independent X-ray fluorescence technique was developed at WAPD for the direct determination of the Pu/U mass ratio. The outline of the X-ray spectrographic method is given on page 230.14 of Reference 8 and a brief summary is related below.

The sample pellet is dissolved in nitric acid to yield a solution of 275 ± 25 micrograms of uranium per milliliter. The uranium and plutonium are precipitated with concentrated ammonium hydroxide and filtered on a plastic membrane filter. The intensities of the lines Pu $L\alpha$, and U Ln are measured with a flat crystal X-ray spectrometer in conjunction with a tungsten target X-ray tube, a topaz analyzing crystal, and a scintillation counter with pulse height discrimination. The ratio of background corrected intensities of Pu $L\alpha$, to U Ln is calculated and compared to those of standard precipitates of known composition prepared in a like manner. The percent mass ratio of plutonium to uranium is determined from a concurrently prepared linear analytical curve relating percent mass ratios to the above determined intensity ratios.

X-ray spectrographic determinations of Pu/U mass ratio were made for all the EYC concentrated acid sample solutions, and for a few dilute acid samples. The results are tabulated in Appendix A3, where all values of Pu/U include a correction for Pu-241 decay from the time of analysis back to the end of core life.

4.3 Radiochemical Analysis

Brief descriptions of the radiochemical methods used in the EYC Program are given in the following paragraphs*.

Cesium-137. Cesium-137 is separated along with Cesium-134 by a carrier free extraction of cesium tetraphenyl boron into amyl acetate. A fraction of the organic phase is taken for gamma spectrometry using a multi-channel analyzer. Correction is made for those counts due to Cesium-134 which are included in the spectrum and Cesium-137 dpm/gU is reported for each sample.

Strontium-90. Strontium carrier is added and a standard separation is used to produce a highly purified strontium carbonate source containing the isotopes Strontium-89 and Strontium-90. This source is stored for two weeks, to allow the Yttrium-90 to come to equilibrium, and is then dissolved in hydrochloric acid in the presence of yttrium carrier. The yttrium is then precipitated and purified and a yttrium oxalate source is prepared for 2 π beta counting. Corrections are made where necessary for counter dead time and counter calibration factors are used which allow correction for backscatter and self-absorption. Strontium-90 dpm/gU is reported on each sample.

Zirconium-95. Zirconium carrier is added and separation is made by precipitation of barium zirconyl fluoride. Several scavenges are made for rare earth activities and barium zirconyl fluoride is reprecipitated and converted to zirconium hydroxide. The final precipitate of zirconium mandelate is recovered for gamma spectrometry. Zirconium-95 dpm/gU is reported on each sample.

The Cs-137 and Sr-90 dpm/gU are given in Appendix A1, where all values have been decay corrected from the time of radiochemical analysis back to the end of core life.

The Zr-95 data, obtained early in the EYC Program, are reported in References 11 and 15. Later in the EYC Program, the 65-day Zr-95 had decayed to such a low level that accurate measurements were no longer possible.

*Obtained from J. A. Corbett, Chemical Development - Technical Service Laboratories, Westinghouse Atomic Power Division.

4.4 Results of Mass Spectrometric, Radiochemical, and X-Ray Fluorescence Analyses of Phase I and Phase II Spent Fuel Samples

The results obtained from 273 mass spectrometric analyses representing 219 locations in the depleted Yankee fuel, are given in Appendix A1. The data consist of the U and Pu isotopic composition of the fuel, and the atomic ratio of Pu-239 to U-238. Also included with the isotopic data are the results obtained from radiochemical analysis for Cs-137 and Sr-90 dpm/gU.

To determine the fuel burnup and other nuclear characteristics of the depleted fuel, some rather long and laborious computations must be performed with the isotopic and radiochemical data. For this reason, and due to the large mass of reactor physics data acquired in the EYC Program, the REBUF computer code was written to facilitate the data reduction. A description of the computations performed with the REBUF code is given in Appendix A2.

The burnup and fuel characteristics inferred from the measured isotopic and radiochemical data, with the use of the REBUF code, are given in Appendix A3.

5.0 FUEL BURNUP INFERRED FROM MASS SPECTROMETRIC AND RADIOCHEMICAL ANALYSES

Three methods have been used to determine experimental values of fuel burnup under the EYC Program. These include analysis of the isotopic composition of the depleted fuel, the activity of the fission product Cs-137, and the activity of the fission product Sr-90. In this report, these are referred to as the isotopic, the Cs-137, and the Sr-90 burnup, respectively. All values of spent fuel burnup obtained in the EYC Program, and the equations used to infer values of burnup from the measured data, are given in Appendices A2 and A3 of this report.

The parameters necessary to infer the isotopic burnup are the energy conversion factors (Mev/fission), the ratio of total fissions in U-238 to the sum of the total fissions in U-235, Pu-239, and Pu-241 (δ^{28}), the effective neutron capture-to-fission ratio for Pu-239 (α^{49}), and the effective neutron capture-to-fission ratio for Pu-241 (α^{41}). In addition, the initial isotopic composition of the pre-irradiated fuel must be known. The parameters necessary to infer the Cs-137 burnup and the Sr-90 burnup are Mev/fission, the half-life, and the isotopic fission yields. The values of these parameters used in this study are summarized in Table 5.0.1. The effective values of energy release per fission and the values of δ^{28} , α^{49} , and α^{41} are from a LEOPARD^{/17} calculation of the Yankee unit cell, performed in Subtask EYC-450^{/18}.

5.1 Comparison of Burnup Inferred from Isotopic, Cs-137, and Sr-90 Data

The values of isotopic and Cs-137 burnup are compared in Figure 5.1.1. The data represent all samples from the asymptotic and perturbed neutron spectra, which were submitted for analysis in concentrated acid solution. The 45° diagonal through the origin indicates equivalent values of burnup from the two methods. The dotted straight line is the least squares fit of the data. Apparently, the results are independent of the neutron spectrum, and the scatter in the data is within the estimated $\pm 6\%$ error^{/3} associated with the isotopic and Cs-137 burnup.

However, while the two methods agree to within 2 to 3% on the average, a bias is evident in the results. At a low burnup of 2,000 MWD/MTU, the Cs-137 burnup is approximately 5% low relative to the isotopic burnup; at a high burnup of 30,000 MWD/MTU, the Cs-137 burnup is approximately 3% high, relative

TABLE 5.0.1

Parameters Used to Infer Fuel Burnup
from Mass Spectrometric and Radiochemical Measurements

Parameter	Burnup Indicator			Reference
	Isotopic Composition	Cs-137	Sr-90	
U-235 Mev/fission ^a	203	203	203	(17)
U-238 Mev/fission	201	201	201	(17)
Pu-239 Mev/fission	211	211	211	(17)
Pu-241 Mev/fission	213	213	213	(17)
δ^{28} ^b	~ 0.08			(17)
α^{49}	~ 0.55			(17)
α^{41}	~ 0.38			(17)
Half life, years		29.4	28.0	(19), (20)
Fission yield, atom/U-235 fission		0.0615	0.0577	(19), (20)
Fission yield, atom/U-238 fission		0.062	0.032	(19), (20)
Fission yield, atom/Pu-239 fission		0.0663	0.022	(19), (20)
Fission yield, atom/Pu-241 fission ^c				

a) The various energy sources in the Yankee reactor (e.g., direct energy of fission, prompt and delayed gamma energy, etc.) are outlined in Reference 3.

b) The variation of δ^{28} , α^{49} , and α^{41} with burnup are shown in Appendix A2.

c) The Pu-239 fission yield data are assumed for Pu-241.

Comparison of Burnup Inferred From
Isotopic and Cs-137 Measurements

Figure 5.1.1

to the isotopic burnup. The discrepancy is probably due to a concentration dependent bias introduced in the experimental procedures, or to a bias introduced in the analytical treatment of the isotopic and radiochemical data. Since the discrepancy between the results of the two methods appears to be a function of burnup, a possible source of the discrepancy may be the use of values of U fission yields which are too high, and Pu fission yields which are too low, in the reduction of the radiochemical data. The approximation (see Section 8.4) used to account for in-core decay of the fission products may introduce a systematic error in the inferred values of Cs-137 (and Sr-90) burnup. Also, the values of δ^{28} , α^{49} , and α^{41} , used to infer the isotopic burnup, may be systematically in error.

In Figure 5.1.2, the values of isotopic and Sr-90 burnup are compared in a manner similar to the comparison described in the preceding paragraph. Relative to the isotopic burnup, the scatter in the Sr-90 data is greater than that observed for the Cs-137 data; however, the bias is opposite that which was noted for the Cs-137 burnup in Figure 5.1.1. At a low burnup of 2,000 MWD/MTU the Sr-90 burnup is approximately 25% high relative to the isotopic burnup; at a high burnup of 30,000 MWD/MTU, the Sr-90 burnup is approximately 1% low relative to the isotopic burnup. As is the case with the Cs-137 burnup data, the discrepancy between the isotopic and Sr-90 burnup data appears to be a function of burnup. This leads to the probability that the discrepancy between the isotopic and Sr-90 burnup is due to the use of values of U fission yields which may be too low, and Pu fission yields which may be too high.

The data shown on Figures 5.1.1 and 5.1.2 were obtained for samples submitted for analysis in concentrated acid solution. For those samples submitted for analysis in dilute acid, it is suspected that cesium and strontium plateout may have occurred. The basis for this suspicion is the data shown on Figures 5.1.3 and 5.1.4. These figures display the sample burnup as a function of U-235 fractional atom depletion (D25) in the asymptotic neutron spectrum. The method used to obtain the U-235 depletion from the isotopic data is given in Appendix A2, and is essentially,

$$D25 = (N_o^{25} - N^{25}) / N_o^{25}, \text{ where}$$

$$N_o^{25} = \text{atomic density of U-235 in the fresh fuel, and}$$

$$N^{25} = \text{atomic density of U-235 in the depleted fuel.}$$

Comparison of Burnup Inferred from
Isotopic and Sr-90 Measurements

Figure 5.1.2

Burnup Inferred from Isotopic Data Versus U-235 Atom Depletion in the Yankee Asymptotic Neutron Spectrum

Figure 5.1.3

Burnup Inferred from Radiochemical Data Versus U-235 Atom Depletion
in the Yankee Asymptotic Neutron Spectrum

Figure 5.1.4

In Figure 5.1.3, the isotopic burnup is shown as a function of U-235 atom depletion. The solid curve is the quadratic least squares fit of the concentrated acid data. Also shown on the figure is the isotopic burnup inferred from the spent fuel composition data obtained for samples submitted for analysis in dilute acid. Because of the probability of Pu plateout, the Pu concentrations (i.e., the Pu-239/U-238 atom ratio) measured for these samples have been adjusted with the method outlined in Subtask EYC-352 of Reference 21. Notice that there is no significant difference in the two sets of results.

In Figure 5.1.4a, the Cs-137 and Sr-90 burnup inferred for concentrated acid samples are shown as a function of U-235 depletion. Shown for reference is the least squares fit of the isotopic data from Figure 5.1.3. With respect to the isotopic least squares fit, the relative scatter and bias in the Cs-137 and Sr-90 burnup data is similar to that observed in Figures 5.1.1 and 5.1.2.

In Figure 5.1.4b, the Cs-137 and Sr-90 burnup inferred from dilute acid samples are shown as a function of U-235 depletion. Again, the least squares fit of the isotopic data from Figure 5.1.3 is shown for reference. Notice that with respect to the isotopic least squares fit, the Cs-137 and Sr-90 burnup is low, indicating the possibility of plateout of these two fission products in dilute acid sample solutions.

5.2 Comparison of Burnup Inferred from Spent Fuel Analysis and Flux Wire Measurements

The pellet burnup inferred from the destructive analysis of the spent fuel depends on the values of the parameters listed in Table 5.0.1. Another method is available to determine the fuel burnup, independent of the parameters listed in Table 5.0.1. The normalized time-integrated power distributions obtained periodically with flux wires during the operation of Yankee Core I, coupled with the core average burnup determined from calorimetric measurements at the Yankee-Rowe site, have resulted in a set of so-called flux-wire burnup values.

In Reference 3, the core average burnup, and the normalized assembly average burnup, inferred from the destructive analysis of the spent fuel have

been compared with corresponding values of burnup inferred from calorimetric and flux-wire measurements. The comparison for the core average burnup is shown in Table 5.2.1.

TABLE 5.2.1

Summary of Yankee Core I Average Burnup

<u>Type of Measurement</u>	<u>Inferred^a Burnup (MWD/MTU)</u>
Plant Calorimetrics	8,470 $\pm$ 250
Mass Spectrometric	8,400 $\pm$ 210
Radiochemical (Cs-137)	8,440 $\pm$ 240

Note a) All data shown are from Reference 3.

In Figure 5.2.1, the values of isotopic, Cs-137, and flux-wire burnup are compared for several typical Yankee Core I spent fuel rods from the asymptotic neutron spectrum. The six samples taken from each of the fuel rods from Assemblies F⁴ and K⁴ were analyzed in concentrated acid solution. The six samples taken from the fuel rod from Assembly H³ were analyzed in dilute acid solution. The solid curve represents the smoothed asymptotic flux-wire burnup; no attempt is made to depict the perturbations caused by the spacer discs, axial expansion voids, and ferrules discussed earlier in Section 3.3.

For all the samples shown on Figure 5.2.1, the values of isotopic, Cs-137, and flux-wire burnup agree to within the uncertainty of the measurements. The 2-standard deviations relative error on the isotopic burnup has been estimated at $\pm 6\%$ in Reference 3. The 2-standard deviation relative error on the Cs-137 burnup is taken as $\pm 6\%$, which is the error generally quoted for the radiochemical measurements made under Subtask EYC-234. Neither of these errors includes any uncertainty in the parameters listed in Table 5.0.1. The relative error on the flux-wire burnup is estimated to be $\pm 6\%$, which is the statistical combination of a $\pm 3\%$ error on the core average burnup obtained from calorimetric data^{/22} and a $\pm 5\%$ error on the normalized time-integrated power distribution obtained with the Yankee in-core instrumentation system^{/3}.

Comparison of Burnup Inferred from Isotopic, Cs-137
and In-Core Instrumentation Measurements for Several
Yankee Core I Spent Fuel Rods

Figure 5.2.1

5.3 Isotopic Fractional Accumulated Fissions

In Figure 5.3.1, the fraction of the total accumulated fissions due to accumulated fissions in U-235, U-238, Pu-239 and Pu-241 are shown as a function of the burnup inferred from mass spectrometric data in the asymptotic neutron spectrum. Notice that the percentage of fissions in U-235, on an accumulated basis, ranges approximately from 90% over a burnup of 2,000 MWD/MTU, to 60% over a burnup of 24,000 MWD/MTU. Over these spans of burnup, the percentage of fissions in Pu-239 ranges approximately from 4% to 30%.

The solid curves shown on Figure 5.3.1 represent the results obtained with a LEOPARD calculation of the Yankee Core I asymptotic unit cell, consisting of fuel, cladding, and moderator. The overall good agreement between the measured and calculated isotopic fractional accumulated fission data is due largely to the fact that the assumed values of α^{49} , α^{41} , and δ^{28} used in the isotopic data reduction (see Table 5.0.1) were taken from the LEOPARD calculation. For example, the inferred values of U-238 fractional accumulated fissions fall within a pencil width of the calculated curve on Figure 5.3.1, and for this reason, are not shown.

However, it can be observed that the calculation overpredicts the fission contribution of Pu-239 at low burnup, and underpredicts the Pu-239 fission contribution at high burnup. This discrepancy is expected, since the calculation was performed with a constant fuel temperature over the entire range of burnup, while the measured data represents samples of relatively low fuel temperature at low burnup, and high fuel temperature at high burnup. Thus, the calculated U-238 resonance absorption probability is relatively high at the lower values of burnup, and relatively low at the higher values of burnup.

Isotopic Accumulated Fissions
Versus Burnup in the Yankee Asymptotic Spectrum

Figure 5.3.1

6.0 URANIUM AND PLUTONIUM ISOTOPIC COMPOSITION OF THE SPENT FUEL

The relationship between plutonium buildup and U-235 depletion is of great importance in the consideration of fuel economics (plutonium credit and fuel depletion charges) and burnup dependent core parameters for batch-loaded cores. With the trend to multi-region cores, the ability to predict the fuel composition accurately is of even greater importance; the optimum equilibrium cycle and the local power peaking factors are dependent upon the relative locations of the recycled assemblies, which are in turn dependent upon the fuel composition and burnup of the recycled assemblies.

In this section, the relationship between fuel burnup and the composition, and concentration, of the U and Pu isotopes is presented. Included for reference are the results obtained with the Yankee asymptotic unit-cell LEOPARD calculation described in previous sections. As will be seen, the measured and calculated data generally are in good agreement.

As was the case with the assembly and total-core fuel burnup, the assembly and total-core U and Pu isotopic inventory has been inferred from the pellet data which appear in this report. The isotopic inventory is reported in Reference 3, where it is also compared to calculated results obtained with the original design methods and with preliminary results of more recent calculations. A more comprehensive comparison is made in Reference 18.

6.1 Uranium and Plutonium Isotopic Composition

The uranium isotopic composition of the Yankee Core I fuel is related to U-235 atom depletion in the asymptotic neutron spectrum in Figure 6.1.1. Along with the measured data, the results of the LEOPARD unit cell calculation are shown for U-235, U-236, and U-238. The LEOPARD calculation did not include the initial content of approximately 0.02 atom percent U-234 and 0.02 atom percent U-236 in the fresh fuel. Consequently, the calculated results have been adjusted to account for the initial concentration of U-234 and U-236.

On Figure 6.1.1, a comparison between the measured and calculated U-235 data is not meaningful since the U-235 atom depletion in the fuel is inferred

Uranium Composition Versus U-235 Atom Depletion in
the Yankee Asymptotic Neutron Spectrum

Figure 6.1.1

directly from the U-235 atom percent. However, the comparison of the U-236 data is meaningful and shows that the calculated U-236 production is consistently high relative to the measured values, indicating that the effective capture-to-fission ratio for U-235 (α^{25}), as contained in the LEOPARD library, is somewhat high. The effective α^{25} is discussed further in Section 7.

Figure 6.1.1 is not detailed enough to permit an evaluation of the U-238 data. This evaluation is made in the following section.

In Figure 6.1.2, the plutonium composition of the fuel is related to U-235 atom depletion in the asymptotic neutron spectrum. Note that the atom percent of a particular plutonium isotope is the percentage of atoms of that isotope present, referred to total plutonium atoms. At zero depletion, the first atom of plutonium formed is Pu-239; consequently, at the beginning-of-life, the Pu-239 atom percent is 100%, while the Pu-240, Pu-241, and Pu-242 atom percents are zero. As will be shown in more detail in Section 6.2, the LEOPARD calculation tends generally to overpredict the plutonium production at low values of burnup, and to underpredict the plutonium production at high values of burnup.

The plutonium composition of the fuel in the perturbed neutron spectra is shown on Figure 6.1.3. For comparison, the asymptotic data, plotted on Figure 6.1.2, are represented on Figure 6.1.3 by the dotted curves. Relative to the plutonium composition in the asymptotic spectrum, the abundance of Pu-239 and Pu-241 (fissile isotopes) is less in the perturbed spectra, and the abundance of Pu-240 and Pu-242 is greater.

As regards the net production of plutonium isotopes, the data on Figure 6.1.3 can be misleading. While the abundance of Pu-240 and Pu-242 is greater in the perturbed spectra, it will be shown in the following section that the specific production (kg/MTU at a given burnup) of each of the four plutonium isotopes is greater in the asymptotic spectrum.

6.2 Net Production of U and Pu Isotopes

Using the REBUF code described in Appendix A2, the net production of U and Pu isotopes has been obtained from the isotopic data measured for each spent fuel sample. In Figures 6.2.1 through 6.2.3, the net uranium isotopic

Plutonium Composition Versus U-235 Atom Depletion in the Yankee Asymptotic Neutron Spectrum

Figure 6.1.2

Figure 6.1.3

Plutonium Composition Versus U-235 Atom Depletion in the Yankee Perturbed Neutron Spectrum

Net Destruction of U-235 Versus Burnup
in the Yankee Asymptotic Neutron Spectrum

Figure 6.2.1

Specific Production of U-236 Versus
Burnup in the Yankee Asymptotic Neutron Spectrum

Figure 6.2.2

Net Destruction of U-238 Versus Burnup
in the Yankee Asymptotic Neutron Spectrum

Figure 6.2.3

production (or destruction) is plotted as a function of the burnup inferred from isotopic data in the asymptotic neutron spectrum. On each figure, the results of the LEOPARD calculation are included for comparison. No adjustments have been made for initial content, since the net production depends on the change in the isotopic concentration and does not significantly depend on the inclusion of the 0.02 atom percent of U-234 and U-236 in the fresh fuel. It is stressed that the data on Figures 6.2.1 through 6.2.3 represent the net change in the mass of U-235, U-236 and U-238; to obtain the mass of U-235, U-236, or U-238 present in the depleted fuel at a particular burnup, the values on Figures 6.2.1 through 6.2.3 must be added to, or subtracted from, the initial mass of these isotopes present in the fresh fuel.

In Figure 6.2.1, it can be observed that the calculation overpredicts the U-235 destruction by approximately 3 to 4%. There are at least two contributing factors which account for most of this discrepancy. The first of these is apparently the use of a somewhat high value of α^{25} in the calculation, which leads to a 5% overproduction of U-236, as shown in Figure 6.2.2. The second contributing factor is an apparent underprediction of Pu-239 production by the LEOPARD calculation at high values of burnup. Thus, to achieve a particular burnup in the calculation, relatively more U-235 must be allowed to fission, which again leads to an overestimate of U-235 loss as observed in Figure 6.2.1.

The destruction of U-235 and the production of U-236 in the perturbed neutron spectra are shown on Figure 6.2.4. For comparison, the asymptotic data, plotted on Figures 6.2.1 and 6.2.2, are represented on Figure 6.2.4 by the dotted curves. Relative to the destruction of U-235 in the asymptotic spectrum, the destruction of U-235 in the perturbed spectra is greater, reflecting the softer (i.e., more thermal) spectrum to which the corner fuel rods were exposed. In the case of the U-236 net production, the integrated spectral effects are apparently the same in both the asymptotic and perturbed spectra.

The discrepancies between the calculated and measured plutonium isotopic production, shown in Figures 6.2.5 through 6.2.8, are consistent with the U-238 data on Figure 6.2.3. The calculation tends to overpredict the net U-238 destruction at low values of burnup, and to underpredict the net U-238 destruction

Net Destruction of U-235 and Net Production of U-236 Versus Burnup
in the Yankee Perturbed Neutron Spectrum

Figure 6.2.4

Specific Production of Pu-239 Versus Burnup in
the Yankee Asymptotic Neutron Spectrum

Figure 6.2.5

Specific Production of Pu-240 Versus Burnup in the
Yankee Asymptotic Neutron Spectrum

Figure 6.2.6

Specific Production of Pu-241 Versus Burnup in
the Yankee Asymptotic Neutron Spectrum

Figure 6.2.7

Specific Production of Pu-242 Versus Burnup in
the Yankee Asymptotic Neutron Spectrum

Figure 6.2.8

at high values of burnup. The cause of this discrepancy is most likely the use of a constant fuel temperature in the calculation at all values of burnup, while the measured data is representative of low fuel temperature in low burnup samples, and high fuel temperatures in high burnup samples. Consequently, the calculated resonance absorption probability for U-238 is constant, and as a result is relatively high at low values of burnup, and relatively low at high values of burnup.

The discrepancies between calculation and experiment noted for the pellet data on Figures 6.2.1 through 6.2.3 and on Figures 6.2.5 through 6.2.8 are consistent with the discrepancies between the calculated and experimentally determined core total isotopic inventories discussed in Reference 3. The experimentally determined total core isotopic inventory is inferred from the measured pellet data presented herein, while the calculated inventory results from the use of fuel constants obtained with the LEOPARD code. The net plutonium production in the asymptotic neutron spectrum is summarized in Figure 6.2.9, where the specific production of total plutonium and fissile plutonium (Pu-239 and Pu-241) is shown as a function of fuel burnup.

The spectral effects on plutonium production are evident in Figures 6.2.10 and 6.2.11 which relate the net production of Pu-239, Pu-240, Pu-241, and Pu-242 to the burnup in several perturbed neutron spectra. As shown previously in Table 3.2.1, Control Rod Groups 1 and 2 remained in the core for a longer period of time than Control Rod Groups 3 and 4. Group 6 was withdrawn during all the time of power operation. Thus, the fuel rods adjacent to Control Rod Groups 1 and 2 were exposed to a relatively higher value of time integrated fast-to-thermal flux ratio than were the fuel rods which were adjacent to Control Rod Groups 3 and 4. The fuel rods adjacent to Control Rod Group 6 were exposed to the lowest fast-to-thermal nvt. Some typical values of fast-to-thermal flux ratios are given in Reference 23.

The total plutonium-to-total uranium mass ratio was obtained from the isotopic data simply by adding together the four plutonium isotopic masses as described in Appendix A2. To verify the Pu/U mass ratio obtained from the mass spectrometric measurements, each sample was subjected to the X-ray fluorescence analysis discussed in Section 4.2.

Specific Production of Total Pu and Fissile Pu Versus Burnup in the Yankee Asymptotic Neutron Spectrum

Figure 6.2.9

Specific Production of Pu-239 and Pu-240 Versus Burnup in the Yankee Perturbed Neutron Spectrum

Figure 6.2.10

Specific Production of Pu-241 and Pu-242 Versus Burnup in the Yankee Perturbed Neutron Spectrum

Figure 6.2.11

The results of the two methods are compared in Figure 6.2.12. Both methods are in statistical agreement for the Phase I samples. However, for values of Pu/U encountered in the relatively high burnup samples of Phase II, the X-ray fluorescence technique gives results which are approximately 3% high with respect to the isotopic measurements. A part of the discrepancy is due to the fact that the X-ray Pu/U mass ratio includes Pu-238, while the isotopic Pu/U does not. The abundance of Pu-238 in total Pu is less than the abundance of Pu-242, and is estimated to be of the order of 1/2% at a burnup of 15,000 MWD/MTU, and about 1% at a burnup of 25,000 MWD/MTU.

The Pu/U mass ratio in the asymptotic neutron spectrum is shown as a function of U-235 atom depletion in Figure 6.2.13. Consistent with the data given previously, the LEOPARD calculation overpredicts the Pu/U at low values of burnup, and underpredicts the Pu/U at high values of burnup.

The Pu/U mass ratio in the perturbed neutron spectra are plotted as a function of U-235 atom depletion in Figure 6.2.14. As was the case with the individual plutonium isotopes in Figures 6.2.10 and 6.2.11, the production of plutonium is relatively less in the more thermal spectra to which the corner fuel rods were exposed.

6.3 Estimate of Total Core Isotopic Inventory

In Reference 3, the total core isotopic inventory and burnup of Yankee Core I is inferred from the results of destructive analysis of approximately 150 spent fuel samples. Through rather elaborate interpolation, reflection, weighting, and summation routines performed with the MASTER code,³ the total core data were inferred from the pellet data. It is possible to infer the total core inventory with the set of curves which relate isotopic specific production (or destruction) to burnup in Figures 6.2.1 through 6.2.3, and in Figures 6.2.5 through 6.2.8. Assuming a Core I average burnup of 8,470 MWD/MTU (see Table 5.2.1), the isotopic inventory of the core has been estimated, and is compared to the more accurate results obtained with the MASTER code in Table 6.3.1. Although the specific production relationships employ data only from the asymptotic spectrum, while the MASTER code considers data from both the asymptotic and perturbed spectra, the total core isotopic inventories obtained with the two methods are in good agreement.

Comparison of Pu/U Mass Ratio Determined from Mass Spectrometric and X-ray Fluorescence Analysis

Figure 6.2.12

Pu/U Mass Ratio Versus U-235 Atom Depletion in
the Yankee Asymptotic Neutron Spectrum

Figure 6.2.13

Pu/U Mass Ratio Versus U-235 Atom Depletion in
the Yankee Non-Asymptotic Neutron Spectrum

Figure 6.2.14

TABLE 6.3.1

Estimate of Isotopic Inventory of Yankee Core I Spent Fuel

Isotope	Initial Loading (kg)	Specific ^{/a} Production at 8,470 MWD/MTU (kg/MTU)	Net Production (kg)	Inventory of Spent Fuel	
				Using Specific Production Relationship (kg)	Using ^{/b} MASTER Code (kg)
U-234	4.1	- 0.018	- 0.4	3.7	3.79 ± 0.10
U-235	711	- 8.37	- 175	536	540 ± 5
U-236	4.1	+ 1.58	+ 33.0	37.1	36.8 ± 1.1
U-238	20,188	- 6.75	- 141	20,050	20,070 ± 20
<u>Total U</u>	<u>20,908</u>	<u>- 13.55</u>	<u>- 283</u>	<u>20,630</u>	<u>20,650 ± 20</u>
Pu-239	0	4.06	84.9	84.9	81.4 ± 1.7
Pu-240	0	0.481	10.0	10.0	10.3 ± 0.4
Pu-241	0	0.205	4.29	4.29	4.95 ± 0.22
Pu-242	0	0.012	0.251	0.251	0.436 ± 0.026
<u>Total Pu</u>	<u>0</u>	<u>4.76</u>	<u>99.5</u>	<u>99.5</u>	<u>97.1 ± 1.8</u>

Notes: a) See Figures 6.2.1 through 6.2.3 and 6.2.5 through 6.2.8 (U-234 data not shown).

b) See Reference 3, page 35.

The specific inventory of uranium and plutonium isotopes, as a function of fuel burnup in the asymptotic neutron spectrum, is summarized on Figure 6.3.1. For clarity, the individual data points have been omitted, and only the best fit of the data obtained for each isotope is shown. A rough estimate of the total core inventory (kg) can be obtained from this figure by multiplying the specific inventory (kg/MTU), at the average core burnup of 8,470 MWD/MTU, by the initial uranium loading of the core, 20.908 MTU. However, due to the coarseness of the mesh on Figure 6.3.1, the values thus obtained are not as precise as those obtained in the previous paragraph.

Uranium and Plutonium Isotopic Inventory of the Yankee Core I Fuel as a Function of Burnup in the Asymptotic Neutron Spectrum

Figure 6.3.1

7.0 FUEL CHARACTERISTICS INFERRED FROM SPENT FUEL ANALYSIS

Two important fuel characteristics which greatly affect the fuel cycle are the effective capture-to-fission ratio in U-235 (α^{25}), and the net number of grams of fissile material destroyed per megawatt day (NGD/MWD). The methods used to infer these two parameters from the results of the mass spectrometric analysis of the spent fuel are given in Appendix A2. All values of α^{25} and NGD/MWD obtained in the EYC Program are tabulated in Appendix A3.

7.1 The Effective Capture-to-Fission Ratio in U-235

Alpha-25 is defined herein as:

$$\alpha^{25} = \frac{\text{total captures in U-235}}{\text{total fissions in U-235}}$$

The values of α^{25} inferred for the EYC spent fuel samples are shown on Figure 7.1.1. In the upper half of the figure, data from the Yankee asymptotic neutron spectrum are shown along with the results of a LEOPARD unit cell calculation. The values of α^{25} on Figure 7.1.1 are quite consistent with those given in Reference 24, where a statistical analysis of α^{25} is presented in detail.

Relative to a least squares fit of the measured data, the calculation gives values of α^{25} that are approximately 2, 3, and 5% high at burnups of 8,000, 12,000 and 24,000 MWD/MTU, respectively. This is consistent with the comparisons made for the U-236 data on Figures 6.1.1 and 6.2.2. Notice that the scatter in the data on Figure 7.1.1 is large, particularly at low values of burnup. Since the value of α^{25} inferred from the mass spectrometric measurements depends essentially on the change in U-235 and U-236 concentrations, the relative error in α^{25} is greatest at low values of burnup, where the changes in the concentrations of these isotopes are the smallest. Considering the scatter in the data, the calculated and inferred α^{25} are in good agreement.

Shown on the lower portion of Figure 7.1.1 are some values of α^{25} representative of the perturbed neutron spectra. These include results obtained for samples taken from corner fuel rods adjacent to Control Rod Groups 1 and 2 in the central region of the core, and from corner fuel rods adjacent to

Effective Capture-to-Fission Ratio in U-235 Versus Fuel Burnup

Figure 7.1.1

Control Rod Group 6 in the outer region of the core. The value of α^{25} from the perturbed spectrum adjacent to Control Rod Groups 1 and 2 is only slightly lower than the value of α^{25} in the asymptotic spectrum. However, at 8,000 MWD/MTU, the value of α^{25} in the perturbed neutron spectrum adjacent to Control Rod Group 6 is about 15% lower than the asymptotic value.

It is relevant to note that Control Rod Groups 1 and 2 were fully or partially inserted in the core for approximately 70% of core power operation. Apparently, the integrated spectral effect on α^{25} in the fuel adjacent to Control Rod Groups 1 and 2 is not too different than that of the asymptotic spectrum. This is also evident in the U-236 data (adjacent to Control Rod Groups 1 and 2) shown on Figure 6.2.4. Control Rod Group 6 was fully withdrawn from the core, and replaced with Zircaloy followers, for all of power operation. Thus, the values of α^{25} inferred from fuel samples adjacent to the Group 6 rods are low, reflecting the thermal flux peaking in these regions.

7.2 Net Number of Grams of Fissile Materials Destroyed Per Megawatt Day

The net number of grams of fissile materials destroyed per megawatt day is defined herein as:

$$\text{NGD/MWD} = \frac{\Delta M^{25} - \Delta M^{49} - \Delta M^{41}}{\text{MWD}}$$

where ΔM^{25} , ΔM^{49} , and ΔM^{41} are the net changes in mass of U-235, Pu-239, and Pu-241. In the case of Yankee Core I, which had no plutonium in the fresh fuel, ΔM^{49} and ΔM^{41} are just the masses of Pu-239 and Pu-241 present in the spent fuel at the end of core life.

The values of NGD/MWD inferred from the results of the mass spectrometric analysis are plotted as a function of burnup in Figure 7.2.1. Also shown for the asymptotic spectrum are the results of the unit cell LEOPARD calculation. Relative to the measured data, the calculated value of NGD/MWD is low at low burnup, and high at high burnup. This is consistent with the comparison made for Pu-239 production in Figure 6.2.5. Again, considering the scatter in the data on Figure 7.2.1, the measured and calculated values of NGD/MWD are in good agreement.

Figure 7.2.1

Net Number of Grams of Fissile Material Destroyed
Per Megawatt Day Versus Fuel Burnup

The lower portion of Figure 7.2.1 displays some non-asymptotic results obtained for fuel rods adjacent to control rod positions and the reflector. As the neutron spectrum becomes more thermal, the value of NGD/MWD increases.

It is interesting to compare the net grams of fissile materials destroyed per megawatt day with the grams of U-235 destroyed per megawatt day. Using the data on Figure 7.2.1, the U-235 net destruction curve on Figure 6.2.1, and a value of 0.25 for α^{25} , the data listed in Table 7.2.1 were obtained. Notice that, due to the production of fissile plutonium, the consumption of U-235 decreases from a value of 1.25 gm/MWD at 0 burnup, to 0.76 gm/MWD for a burnup of 24,000 MWD/MTU, a decrease of approximately 40%. Although the net mass of fissile materials consumed per unit of energy generated decreases with increased burnup, it must be recognized that the initial fuel enrichment must be increased to obtain a higher burnup.

For Yankee Core I, for an initial enrichment of 3.4 w/o U-235 and a final burnup of 8,400 MWD/MTU, approximately one gram of U-235 was destroyed, and a net half of gram of fissile plutonium was produced, for each megawatt day of thermal energy released. Thus, the net grams of fissile material destroyed per megawatt day of energy released was 0.5.

In Figure 7.2.2, the net production of fissile plutonium (i.e., Pu-239 and Pu-241) is plotted as a function of U-235 destruction (i.e., captures and fissions) in the asymptotic neutron spectrum. For every two kilograms of U-235 destroyed, a net one kilogram of fissile plutonium is produced. The constancy of the conversion ratio is indicated in Figure 7.2.2, where a straight line drawn through the origin at 26.6° (the Arctan of 1/2) provides a remarkably good fit of the data points over a burnup range of 2,000 to 24,000 MWD/MTU.

TABLE 7.2.1

Mass of Material Consumed Per Unit
of Energy Generation in Yankee Core I

<u>Burnup^{/a}</u> <u>(MWD/MTU)</u>	<u>Net gm of^{/b}</u> <u>Fissile Materials</u> <u>Destroyed/MWD</u>	<u>Gm of U-235</u> <u>Destroyed/MWD</u>	<u>Gm of U-235</u> <u>Fissioned/MWD</u>
0	0.6	1.25	1.0
0 to 8,000	0.5	1.00	0.8
0 to 24,000	0.4	0.76	0.6
Core avg. over ^{/c} 8,400 MWD/MTU	0.5	0.98	0.82

Notes: a) MWD includes energy from all sources, i.e., the effective energy release from fissions in U-235, U-238, Pu-239, and Pu-241.

$$b) \text{ NGD/MWD} = (\Delta M^{25} - \Delta M^{49} - \Delta M^{41}) / \text{MWD}$$

c) See Reference 3.

Net Production of Fissile Plutonium Versus Destruction of U-235 in the Yankee Asymptotic Neutron Spectrum

Figure 7.2.2

8.0 SOURCES OF ERROR IN SPENT FUEL DATA

Generally speaking, the relationships between the various fuel characteristics and burnup, considered in this study, are well defined due to the rather large number of spent fuel pellets which were sampled. However, referring to the functional relationships depicted in the figures in Sections 5, 6, and 7 (e.g., the Pu production versus fuel burnup on Figure 6.2.9, or alpha-25 versus burnup on Figure 7.1.1), it is apparent that the scatter in the spent fuel data is somewhat greater than one would expect, based on the uncertainties quoted for any particular measurement. It is the purpose of this section of the report to discuss qualitatively, and in some cases quantitatively, the many sources which can contribute to the statistical, and systematic, uncertainty in the radiochemical and mass spectrometric data, and in the fuel characteristics and burnup inferred from the data.

The uncertainties in the radiochemical and mass spectrometric data obtained for the EYC samples, and in the values of burnup and fuel characteristics inferred from these data, can be attributed to four causes: 1) engineering and manufacturing tolerances on core components and fuel; 2) spectral effects due to mechanical and physical phenomena which occurred during reactor operation; 3) experimental methods, including sampling and laboratory analysis; and 4) uncertainties in assumed values of some nuclear parameters and physical constants used in the data reduction process. The uncertainties due to the first three causes are generally statistical in nature, while the uncertainties due to the fourth cause can be both statistical and systematic.

Estimates of the statistical uncertainties including the overall effects of the four sources of error listed above (but not including any systematic errors), are given in Table 8.0.1 for several important quantities. These estimates are based on the scatter observed in the data used to define the functional relationships on Figures 6.2.1, 6.2.9, 7.1.1, and 7.1.2. Shown along with the overall statistical uncertainty is an estimate of the uncertainty due to the laboratory analysis (i.e., due to the precision of the mass spectrometric measurements, based on the results of eleven separate

TABLE 8.0.1

Estimate of Relative Uncertainty in Inferred Values of Burnup and Spent Fuel Characteristics

<u>Quantity^{/a}</u>	<u>2,000 to 5,000 MWD/MTU</u>	<u>12,000 to 14,000 MWD/MTU</u>	<u>Monitor Sample^{/b} at 12,000 MWD/MTU</u>
Fuel Burnup	$\pm 10\%$ ^{/d}	$\pm 6\%$	$\pm 2\%$
U-235 Depletion	$\pm 10\%$	$\pm 5\%$	$\pm 2\%$
Net Fissile Pu Production	$\pm 7\%$	$\pm 6\%$	$\pm 3\%$
NGD/MWD ^{/c}	$\pm 13\%$	$\pm 9\%$	$\pm 5\%$
Alpha-25	$\pm 20\%$	$\pm 9\%$	$\pm 5\%$

Notes: a) Inferred from results of mass spectrometric analysis

b) Relative precision based on 11 samples of a monitor sample solution. See Table A3.1.

c) Net number of grams of fissile material destroyed per megawatt of energy released.

d) Two standard deviations, relative uncertainty.

analyses of a monitor sample solution). Notice that at a burnup of approximately 12,000 MWD/MTU the overall statistical uncertainty is 2 to 3 times as great as the uncertainty due only to the experimental error in the laboratory analysis.

Also, the relative overall uncertainty is dependent upon burnup; for those quantities inferred from U-235 data (e.g., fuel burnup, U-235 depletion, α^{25}), the overall statistical error is roughly a factor of two greater at low values of burnup than at high values of burnup.

The sources of error are discussed in more detail in the following paragraphs.

8.1 Engineering and Manufacturing Tolerances on Core Components and Fuel

Engineering and manufacturing tolerances on the fuel pellet dimensions and composition, the fuel rod cladding thickness and O.D., and the fuel rod pitch (fuel assembly fabrication) can cause changes in the local water-to-uranium and water-to-steel ratios with resulting neutron spectrum changes that contribute to the scatter in the spent fuel data presented in the figures in Sections 5, 6, and 7.

The Yankee Core I fuel pellets were obtained from two vendors: Mallinckrodt and the Westinghouse Atomic Fuel Department (WAFD). The initial composition of the fuel supplied by the two vendors is given on page 15 in Reference 5, and is repeated in Table 8.1.1.

TABLE 8.1.1

Chemical Composition of UO₂ Pellets in Yankee Core I

<u>Analysis</u>	<u>Mallinckrodt</u>	<u>WAFD</u>	<u>Weighted Avg.</u> ^a
U-235/U (weight %)	3.391%	3.406%	3.399%
Total U (weight %)	88.07 %	88.05 %	88.06 %
O/U (atomic ratio)	1.996	1.98	1.988

a. The weighted average is based on an equal mixture of the respective loadings of the two vendors' pellets.

The slightly different enrichments, along with different mixtures of pellet impurities (e.g., Al, Cr, Fe, N, Si, etc.), cause slight differences in neutron spectrum which contribute to the scatter observed in the spent fuel data. As will be demonstrated in Section 8.4, the uncertainties in the initial uranium isotopic composition of the fuel can lead to somewhat larger uncertainties in the values of burnup and fuel characteristics inferred from the spent fuel data, especially for those samples which experienced relatively low exposure.

8.2 Uncertainties in Spent Fuel Data Caused by Mechanical and Physical Phenomena Which Occur During Core Operation

Some of the scatter in the measured, and inferred, spent fuel data is undoubtedly due to mechanical and physical phenomena which occur during core operation. These include fuel rod bowing, thermal flux peaking in the vicinity of flux wire thimble locations, and a rise in moderator temperature between the core inlet and outlet coolant channels.

Due to thermal stresses which occur during core operation, it is conceivable that some fuel rods can become bowed, thereby changing the water-to-uranium ratio in the vicinity of the bow. The resulting change in the local neutron spectrum can cause some scatter in the spent fuel data. That is, in correlating data (e.g., Pu/U versus fuel burnup), it is not expected that results obtained for pellets from a fuel rod that was bowed will fall on the exact same curve as the results for pellets taken from a rod that was not bowed. This has been demonstrated with the use of calculated data, representing two different Yankee neutron spectra, on pages 350.18 through 350.25 in Reference 8. Some of the calculated data are repeated in Table 8.2.1.

TABLE 8.2.1

Effect of Fuel Rod Pitch on Neutron Spectrum

<u>Cold Pitch (inches)</u>	<u>Clearance Between Adjacent Rods</u>	<u>Hot Water-to- Uranium Ratio</u>	<u>Calculated ϕ_1/ϕ_2 at 8,470 MWD/MTU</u>
0.422x0.422	82 mils	2.69	11.2
0.422x0.486	146 mils	3.55	9.6

The conclusions drawn from the study in Reference 8 are that a neutron spectrum change (i.e., a 15% change in the fast-to-thermal neutron flux ratio) caused by conjectured fuel rod bowing (i.e., a 30% change in the water-to-uranium ratio) could cause $\pm 0.5\%$ scatter in the measured U-235 atom percent data, ± 1 to 2% scatter in the inferred burnup data, and $\pm 6\%$ scatter in the measured Pu/U data. However, the example cited is felt to be in the extreme, since to achieve a 30% change in the water-to-uranium ratio, it is necessary that two adjacent fuel rods bow 32 mils, off center, in opposite directions, or that one fuel rod bow 64 mils while the adjacent fuel rod does not bow at all (or some combination of the two cases).

All the samples representing the asymptotic neutron spectrum were removed from fuel rods which occupied the corner positions of the center subassembly of a fuel assembly. A layout of the center subassembly of a Yankee fuel assembly is shown on Figure 8.2.1. Indicated on the figure are the locations from which the asymptotic rods were removed, and the location of the flux wire thimble. Notice that the fuel rod in the upper right corner of the center subassembly (position f1) is diagonally 2 lattice pitches away from the flux wire thimble location, the fuel rods in the upper left and lower right corners (a1 and f6, respectively) are about 2-1/2 lattice pitches away, and the fuel rod in the lower left corner (position a6) is 3 lattice pitches away from the flux wire thimble location. Since the flux wire thimble encloses a void, thereby increasing the local water-to-uranium ratio, there is some local flux peaking in the vicinity of the thimble location. Although the influence of the thimble is small,* some effects of the flux perturbation are felt non-uniformly by the four corner positions of the center subassembly, since these corner positions are not equidistant from the flux wire thimble location.

The results of critical experiments (at room temperature) indicate that differences in power density in the corner fuel rods of the center subassembly, due to the unsymmetric location of these rods with respect to a central water hole (caused by the removal of the central fuel rod), are of the order of 1% (see page 65 in Reference 30). If a dry thimble (i.e., a hollow steel tube) is placed in the water hole, the non-uniformity in power density is reduced by a factor of two (see page 12 in Reference 31). At operating temperatures the effect is even smaller. With regard to plutonium production, results of a 16-assembly TURBO calculation¹⁸ of the Yankee core indicate that the perturbation caused by the central flux wire thimble introduces no more than about $\pm 1\%$ of the ± 6 or 7% scatter observed, for example, in Figure 7.2.2.

Flux-Wire Thimble Location in the Center Subassembly of a Yankee Fuel Assembly

Figure 8.2.1

Finally, some scatter exists in the spent fuel data due to the spectral change along the vertical dimension of the reactor core. The temperature rise from the inlet to the outlet coolant channels causes a decrease in the moderator density, which decreases the water-to-uranium ratio and hardens the spectrum at the top of the core, relative to the bottom of the core. Attempts to correlate data with respect to the axial sample zone (i.e., attempts to establish a family of curves which relate, for example, the Pu/U versus fuel burnup in the six axial sample zones) have indicated that any trends in the data due to the axial spectral shift effect are smaller than the uncertainties in the data.

8.3 Sampling and Analytic Errors Due to Experimental Methods

Some of the uncertainty in the EYC spent fuel data is due to errors inherent in the experimental methods used in the program. These include statistical errors which arise during sampling operations, and during the performance of subsequent mass spectrometric, radiochemical and x-ray spectrographic analyses.

During the sampling process, uncertainties can be introduced during the fuel rod sectioning and the sample dissolution and pipetting operations. Loss of fuel fragments from some pellets which are cracked more than others can lead to the acquisition of a sample solution which is not typical of the fuel being sampled. Also, uncertainties can arise due to incomplete dissolution and contamination during the preparation of the sample fuel solution. Finally, the analyses themselves are subject to contamination errors and statistical errors in measurement.

To determine the precision in the mass spectrometric, radiochemical (Cs-137), and x-ray fluorescence Pu/U data, a sample solution of the spent fuel pellet which was cut from Axial Sample Zone #3 of Fuel Rod #0994, Core Location G4-C-a1 (see Appendix A1 for core-location designation), was included in a number of sample batches submitted for analysis. The results for this "monitor" sample, as processed with the REBUF-2 computer code (see Appendix A2), are shown in Figures 350.2 through 350.12 in Reference 1. The measured and inferred quantities are listed in Tables A1.1 and A3.1.

The mean value of each quantity, and the 2 standard deviations of the mean, are summarized in Table 8.3.1. The mean value is computed with:

$$\bar{A} = \frac{1}{n} \sum_{i=1}^n A_i$$

The standard deviation of the mean^{/28} is computed with:

$$S_m^2 = \frac{1}{n} \sum_{i=1}^n (\bar{A} - A_i)^2 / (n-1)$$

where,

A = the measured or inferred quantity

n = the number of separate analyses (7 for Tracerlab isotopics, 4 for General Electric isotopics, 15 for WAPD Cs-137, and 17 for WAPD x-ray Pu/U)

S_m = the standard deviation of the mean.

The New Brunswick data should be viewed with some reservation since the mean and standard deviation of each quantity are estimated from the results of a duplicate analysis of just a single solution of the monitor sample. The results of each individual analysis of the monitor sample are given in Tables A1.1 and A3.1.

Also shown in Table 8.3.1 are the absolute differences between the mean results obtained from the mass spectrometric analysis of the Tracerlab, Inc. and the GE-Vallecitos laboratories. With the exception of the U-234 data, the absolute difference between the two means, for any quantity, generally is smaller than the 2 standard deviations of the mean as obtained for either or both laboratories. In all cases, the absolute difference between the two means is smaller than the sum of the two values of mean standard deviation. The discrepancy in the U-234 data was noted previously in Sub-task EYC-234 of WCAP-6061^{/8}.

The mass spectrometric results shown in Table 8.3.1 are considered excellent, particularly in view of the following facts:

1. All solutions of the monitor sample were submitted under coded identification, along with the regular samples. The laboratories were not aware of the existence of the monitor sample.

TABLE 8.3.1

SUMMARY OF RESULTS OBTAINED FROM REPEATED MASS SPECTROMETRIC,
RADIOCHEMICAL, AND X-RAY FLUORESCENCE ANALYSES OF MONITOR SAMPLE #0994-3

	Quantity	Units ^{/a}	Multiplier ^{/b}	Mean Value $\pm$ 2 Standard Deviations of Mean				Absolute ^{/g} Difference TL-GE
				Tracerlab ^{/c}	General Electric ^{/d}	New Brunswick ^{/e}	WAPD ^{/f}	
Measured	U-234	a/o	1×10^{-2}	1.64 $\pm$ 0.03	1.8 $\pm$ 0.1	1.8 $\pm$ 0.3		0.20
	U-235	a/o	1×10^0	2.370 $\pm$ 0.008	2.36 $\pm$ 0.02	2.38 $\pm$ 0.02		0.015
	U-236	a/o	1×10^0	0.2343 $\pm$ 0.0010	0.234 $\pm$ 0.004	0.24 $\pm$ 0.01		0.0008
	U-238	a/o	1×10^0	97.380 $\pm$ 0.008	97.38 $\pm$ 0.02	97.36 $\pm$ 0.01		0.005
	Pu-239	a/o	1×10^0	81.002 $\pm$ 0.098	81.04 $\pm$ 0.13	80.93 $\pm$ 0.17		0.043
	Pu-240	a/o	1×10^0	12.03 $\pm$ 0.06	11.98 $\pm$ 0.10	12.05 $\pm$ 0.10		0.06
	Pu-241	a/o	1×10^0	6.419 $\pm$ 0.042	6.44 $\pm$ 0.04	6.47 $\pm$ 0.06		0.026
	Pu-242	a/o	1×10^0	0.548 $\pm$ 0.004	0.544 $\pm$ 0.005	0.555 $\pm$ 0.013		0.005
	U-238 ^{/h}	a/ml	1×10^{16}	7.207 $\pm$ 0.029	2.092 $\pm$ 0.007	64.30 $\pm$ 0.23		
	Pu-239/U-238	a/a	1×10^{-3}	5.414 $\pm$ 0.077	5.45 $\pm$ 0.05	5.364 $\pm$ 0.011		0.040
	Pu/U	gm/gm	1×10^{-3}	6.55 $\pm$ 0.09	6.59 $\pm$ 0.06	6.49 $\pm$ 0.03	6.56 $\pm$ 0.06	0.05
	Cs-137	dpm/gmU	1×10^{10}				8.44 $\pm$ 0.05	
Inferred	N ²⁴ /N ²⁸	a/a	1×10^{-4}	1.67 $\pm$ 0.02	1.8 $\pm$ 0.1	1.8 $\pm$ 0.3		0.18
	N ²⁵ /N ²⁸	a/a	1×10^{-2}	2.41 $\pm$ 0.01	2.40 $\pm$ 0.02	2.42 $\pm$ 0.02		0.02
	N ²⁶ /N ²⁸	a/a	1×10^{-3}	2.38 $\pm$ 0.01	2.37 $\pm$ 0.04	2.4 $\pm$ 0.1		0.02
	N ²⁸ /N ²⁸	a/a	1×10^0	0.9898 $\pm$ 0.0001	0.9897 $\pm$ 0.0001	0.9899 $\pm$ 0.0001		0.0002
	N ¹⁹ /N ²⁸	a/a	1×10^{-3}	5.358 $\pm$ 0.076	5.39 $\pm$ 0.04	5.310 $\pm$ 0.011		0.036
	N ¹⁰ /N ²⁸	a/a	1×10^{-4}	7.958 $\pm$ 0.108	7.98 $\pm$ 0.14	7.908 $\pm$ 0.096		0.027
	N ¹¹ /N ²⁸	a/a	1×10^{-4}	4.246 $\pm$ 0.062	4.28 $\pm$ 0.06	4.24 $\pm$ 0.05		0.039
	N ¹² /N ²⁸	a/a	1×10^{-5}	3.62 $\pm$ 0.04	3.62 $\pm$ 0.04	3.64 $\pm$ 0.10		0.01
	U-235 Depl.	a/a	1×10^0	0.324 $\pm$ 0.002	0.326 $\pm$ 0.005	0.321 $\pm$ 0.006		0.003
	Burnup	MWD/MTU	1×10^3	11.91 $\pm$ 0.09	11.98 $\pm$ 0.19	11.79 $\pm$ 0.06	11.62 $\pm$ 0.06	0.08
	Alpha-25	capt/fission	1×10^0	0.254 $\pm$ 0.003	0.244 $\pm$ 0.010	0.249 $\pm$ 0.009		0.002
	NGD/MWD	gm/MWD	1×10^0	0.454 $\pm$ 0.009	0.452 $\pm$ 0.011	0.454 $\pm$ 0.016		0.003

Notes: a) For the uranium isotopes, the atom percent refers to the ratio of the atoms of a particular U isotope to total U atoms; for the plutonium isotopes, the atom percent refers to the ratio of the atoms of a particular Pu isotope to total Pu atoms. The other quantities are defined in Appendix A2.

- b) The multiplier is the factor which must be applied to the mean value, and the 2 standard deviations of the mean. For example, the U-234 a/o, as determined from the Tracerlab analyses, is 0.0164 ± 0.0003 .
- c) Based on the results of separate analyses of seven solutions of the monitor sample.
- d) Based on the results of separate analyses of four solutions of the monitor sample.
- e) Estimated from the results of a duplicate analysis of just a single solution of the monitor sample.
- f) Based on the results of fifteen separate radiochemical analyses for Cs-137, and seventeen separate X-ray fluorescence analyses for Pu/U mass ratio.
- g) Differences between the means have been maximized by including roundoff error.
- h) The U-238 a/ml data shown for GE were obtained from the final 3 monitor analyses by GE. The first analysis (a different aliquot) gave 7.30×10^{16} a/ml U-238.

2. The 7 Tracerlab analyses spanned a period of eleven months; the 4 General Electric analyses spanned a period of seven months.
3. The Tracerlab and General Electric laboratories use somewhat different techniques in the preparation of samples for mass spectrometric analysis. Differences in methods are outlined in Section 4.1.

Table 8.3.2 is a summary of the relative precision in the mean, and in the data used to compute the mean, for the measured and inferred quantities listed in Table 8.3.1. The expression used to obtain the standard deviation of the mean, S_m , has been given previously. To obtain the standard deviation of the data ^{28}m used to compute the mean:

$$S^2 = \sum_{i=1}^n (\bar{A} - A_i)^2 / (n-1) .$$

Also shown in Table 8.3.2 is the relative difference in the mean values obtained by the Tracerlab and GE-Vallecitos laboratories for each quantity. For reference, the accuracies quoted by the laboratories are given for each quantity.

It is stressed that the results summarized in Tables 8.3.1 and 8.3.2 relate only to the precision in the data, and except for a comparison of the mean values of the various quantities obtained by the laboratories, are not an indication of the overall accuracy of the data. The results shown herein reflect only the uncertainties which arise during the final handling of the monitor sample solution prior to shipment to the Tracerlab and General Electric laboratories, and the uncertainties which arise during the sample preparation and mass spectrometric analysis at the laboratories. In no way do the results include the errors which are introduced in the hot cells during the acquisition and preparation of the sample, or in the subsequent analytic treatment of the sample data necessary to obtain inferred quantities (burnup, α^{25} , atom ratios, etc.).

8.4 Uncertainties in Assumed Values of Physical Constants and Nuclear Parameters Used in Data Reduction

The values of burnup and fuel characteristics inferred from the results of mass spectrometric and radiochemical analyses depend upon certain physical

TABLE 8.3.2

SUMMARY OF RELATIVE PRECISION OF MASS SPECTROMETRIC
DATA OBTAINED FROM REPEATED ANALYSES OF MONITOR SAMPLE #0994-3

DATA OBTAINED FROM REPEATED ANALYSES OF MONITOR SAMPLE #0994-3										
Quantity ^{/a}	Relative Precision (%) ^{/b}			Relative Difference ^{/c} Between Labs (%)			Quoted ^{/d} Accuracies (%)			
	2 St'd Dev. of Data			2 St'd Dev. of Mean			$\bar{A}_{TL} - \bar{A}_{GE}$			
	TL	GE	NB	TL	GE	NB	$\bar{A}_{TL}$	TL	GE	NB ^{/e}
U-234 a/o	± 4.0%	± 9.0%	±24.1%	± 1.8%	± 5.6%	±17.1%	12.2%	± 2.0%	± 2.0%	±17.0%
U-235 a/o	0.8	1.5	1.2	0.3	0.8	0.8	0.6	1.0	0.5	1.3
U-236 a/o	1.2	3.8	6.0	0.4	1.7	4.2	0.3	2.0	2.0	3.9
U-238 a/o	0.02	0.04	0.01	0.008	0.02	0.01	0.005	0.1	0.5	0.1
Pu-239 a/o	0.3	0.3	0.3	0.12	0.16	0.20	0.05	0.1	2.0	0.1
Pu-240 a/o	1.3	1.7	1.1	0.5	0.8	0.8	0.5	1.0	2.0	1.0
Pu-241 a/o	1.7	1.2	1.2	0.6	0.6	0.8	0.4	2.0	2.0	1.0
Pu-242 a/o	2.2	1.8	3.2	0.7	0.9	2.3	0.9	2.0	2.0	2.0
U-238 a/ml ^{/f}	1.1	0.7	0.5	0.4	0.3	0.4		1.0	2.0	^{/g}
Pu-239/U-238 atom ratio	3.8	1.8	0.3	1.4	0.9	0.2	0.7			
Pu/U Mass Ratio	3.7	1.8	0.6	1.4	0.9	0.4	0.8	~3.0	2.0	^{/g}
N ²⁴ /N ²⁸ a/a	± 4.0%	± 9.1%	±24.1%	± 1.2%	± 5.6%	±17.1%	10.8%			
N ²⁵ /N ²⁸ a/a	0.9	1.5	1.2	0.4	0.8	0.8	0.8			
N ²⁶ /N ²⁸ a/a	1.2	3.8	6.0	0.4	1.7	4.2	0.8			
N ²⁸ /N ²⁸ a/a	0.03	0.02	0.01	0.01	0.01	0.01	0.02			
N ⁴⁹ /N ²⁸ a/a	3.7	1.7	0.3	1.4	0.7	0.2	0.7			
N ⁴⁰ /N ²⁸ a/a	3.6	3.5	1.7	1.4	1.8	1.2	0.3			
N ⁴¹ /N ²⁸ a/a	3.8	2.6	1.8	1.5	1.4	1.3	0.9			
N ⁴² /N ²⁸ a/a	3.2	2.3	3.8	1.1	1.1	2.7	0.3			
U-235 atom depletion	± 1.8%	± 3.1%	± 2.5%	± 0.6%	± 1.5%	± 1.8%	0.9%			
Burnup	2.1	3.2	0.7	0.8	1.6	0.5	0.7			
Alpha-25	3.3	8.2	5.2	1.2	4.1	3.7	0.8			
NGD/MWD	5.3	4.9	4.8	2.0	2.4	3.4	0.7			

- Notes:
- a) All quantities are defined in Appendix A2.
 - b) Estimated from the results of seven analyses of the monitor sample by Tracerlab, four analyses of the monitor sample by GE-Vallecitos, and a duplicate analysis of just one solution of the monitor sample by New Brunswick.
 - c) The relative difference between laboratory results were obtained with the absolute differences given in Table 8.3.1.
 - d) Quoted accuracies are from Reference 8, page 230.11.
 - e) These values are based on a comparison of the stated limits of accuracy and the actual abundances found for a sample containing 2.3% U-235.
 - f) The U-238 a/ml data shown for GE were obtained from the final three monitor analyses by GE.
 - g) Not stated.
 - e) These values are based on a comparison of the stated limits of accuracy and the actual abundances found for a sample containing 2.3% U-235.
 - f) The U-238 a/ml data shown for GE were obtained from the final three monitor analyses by GE.
 - g) Not stated.

constants and fuel parameters required in the data reduction process. The physical constants necessary to infer burnup from the radiochemical data include fission product half-lives, isotopic fission yields, and energy conversion factors (Mev/fission). The value of fuel burnup inferred from fission product activity (e.g., Cs-137 dpm/gU) is also dependent to some degree upon the approximation used to account for in-core radioactive decay of the fission product. The fuel parameters required in the reduction of the mass spectrometric data are values of δ^{28} , α^{49} , and α^{41} , as well as the Mev/fission constants. Errors in any of these factors lead to systematic error, or bias, in the final results. In addition, uncertainties in the initial isotopic composition of the pre-irradiated fuel cause uncertainties in the data inferred from results of post-irradiation mass spectrometric analyses. Each of these contributors to the overall uncertainty in the spent fuel data is discussed briefly in the following paragraphs.

The value of Cs-137 half-life used in the EYC Program is 29.4 years (see Table 5.0.1). A tentative value of 30.3 years, reported in Reference 25, indicates that the relative uncertainty associated with the half-life may be of the order of 3%. (A weighted estimate of 30.76 ± 0.5 years, from specific activity measurements, ASTM - Cs-137 standard, and many published values, is reported in Reference 26.) For a typical EYC sample (about 1 1/2 years irradiation and 2 years cooling time), a +3% increase in the value of the half-life used in the data reduction process will cause a net +2.8% increase in the inferred value of burnup.

In the EYC Program, the measured Cs-137 activity has been corrected for in-core decay in converting dpm/gU to MWD/MTU burnup. As pointed out in Appendix A2, the in-core decay correction factor is approximated with $(1 - e^{-\lambda t})/\lambda t$, where λ is the decay constant and t is the irradiation time. The use of this approximation assumes that the rate of production of Cs-137 is constant with time. Since the production rate of Cs-137 is not constant with time, the use of this approximation introduces some error (bias) in the inferred value of burnup, particularly at high values of exposure. For cycled fuel, the approximation becomes worse.

The values of fission product yield used in the EYC Program are listed in Table 5.0.1. The value used for Cs-137 fission yield in U-235 is 0.0615 and is in good agreement with that given in Reference 25, 0.0622 ± 0.0014 . Increasing the value of the effective fission yield by +2% will cause a +2% decrease in the inferred value of burnup. At low values of burnup, the uncertainty in the U-235 fission yield is weighted most heavily since most of the fissions occur in U-235. However, as burnup increases, and Pu builds up, the uncertainty in the Pu yield becomes relatively more important as more fissions occur in Pu-239 and Pu-241. (See Figure 5.3.1 for the fraction of total accumulated fissions due to accumulated fissions in U-235, U-238, Pu-239, and Pu-241.)

The uncertainty in the value of burnup inferred from radiochemical and mass spectrometric data is also proportional to the uncertainty in the energy conversion factors. The values of these constants, used in the EYC Program, are listed in Table 5.0.1. As is the case with the values of isotopic fission yield, the uncertainty in Mev/fission in U-235 is most important at low values of burnup; as burnup increases, the uncertainties in the values of Mev/fission in Pu-239 and Pu-241 become relatively more important.

In reducing the mass spectrometric data, it is necessary to assume values of α^{49} , α^{41} , and δ^{28} . The nominal values used in the EYC Program are listed in Table 5.0.1; the spectrum and burnup dependent values actually used to obtain the results listed in Table A3.1 are given in Appendix A2.

The inferred spent fuel data most sensitive to the values of δ^{28} , α^{49} , and α^{41} are shown in Table 8.4.1. Notice that the most adverse combination of a 10% error in each of the assumed values of δ^{28} , α^{49} , and α^{41} leads to a 3% error in the inferred value of fuel burnup, and to a 3% error in the value of net number of grams of fissile materials destroyed per megawatt day (NGD/MWD). For fuel samples which have been subjected to lower exposures than 13,000 MWD/MTU, the effects of uncertainties in δ^{28} , α^{49} , and α^{41} are smaller; for samples which have been subjected to higher exposures (i.e., Phase II), the effects are greater.

TABLE 8.4.1

Effect of Uncertainties in δ^{28} , α^{49} , and α^{41} Used
In Reduction of Mass Spectrometric Data

Quantity/ ^a	Inferred Value/ ^b	Relative Change in Inferred ^c Quantity Caused by a +10% Change in the Nominal Assumed Value of:		
		Delta-28	Alpha-49	Alpha-41
N^{28}/N_O^{28}	0.98845 atoms/atom	- 0.01%	+0.03%	0.0
NGD/MWD	0.460 gm/MWD	- 0.8 %	+2.1 %	+ 0.3%
F^{25}/N_O^{28}	0.0106 fissions/atom	0.0	0.0	0.0
F^{28}/N_O^{28}	0.00116 fissions/atom	+10.5 %	-0.9 %	0.0
F^{49}/N_O^{28}	0.00299 fissions/atom	0.0	-9.2 %	- 0.5%
F^{41}/N_O^{28}	0.00014 fissions/atom	0.0	0.0	-11.1%
Burnup	13,780 MWD/MTU	+ 0.8 %	-1.6 %	- 0.2%

Notes:

- All quantities are defined in Appendix A2.
- These results are inferred from mass spectrometric data obtained for sample 0979-3, core location G4-C-f1. See Tables A1.1 and A3.1.
- Nominal values of the parameters δ^{28} , α^{49} , and α^{41} used for this sample are 0.0846, 0.552, and 0.381, respectively.

From the range of values of direct fission energy listed in several sources, it is estimated that the uncertainties in the Mev/fission constants are of the order of $\pm 5\%$. The uncertainties in the nuclear parameters δ^{28} , α^{49} , and α^{41} , for a given neutron spectrum averaged over time are difficult to evaluate. However, as discussed in Reference 3, a comparison of the values of total core burnup inferred from the mass spectrometric and Cs-137 results obtained for many pellet samples provides an integral check on the physical constants and nuclear parameters used in the study. For example, the total core burnup inferred from the mass spectrometric and Cs-137 data agree to within 0.5% (i.e., 8,400 versus 8,440 MWD/MTU, see Table 5.2.1), which tends to verify the set of values of half-life and fission yield used in the reduction of the Cs-137 data, as well as the values of δ^{28} , α^{49} , and α^{41} used in the reduction of the mass spectrometric data. It is true that this remarkably good agreement may be fortuitous; the result of a cancellation of errors. However, both values of total core burnup are in excellent agreement (less than 1%) with the value of total core burnup of 8,470 MWD/MTU obtained from plant calorimetric measurements, a method completely independent of the two destructive analyses.

As was mentioned earlier in Section 8.0, the values of burnup and fuel characteristics inferred from stoichiometric relationships involving mass spectrometric data are dependent upon a knowledge of the initial isotopic composition of the pre-irradiated fuel. (The burnup inferred from radiochemical data does not depend on a knowledge of the pre-irradiated fuel composition.)

The U-235 atom enrichment of the 3.5 million Yankee Core I fuel pellets supplied by WAFD and Mallinckrodt (prior to August 1960) is given in Table 8.4.2. At the outset of the EYC mass spectrometric analyses, in May 1963, sample solutions made from an unirradiated Yankee fuel pellet, thought to be typical of the fresh Core I fuel, were submitted to Tracerlab, Inc. and General Electric-Vallecitos laboratories for analysis. The purpose of these analyses was threefold: 1) to confirm the U-235 data provided by WAFD and Mallinckrodt, 2) to determine the initial U-234 and U-236 atom percent, and 3) to verify the absence of any plutonium in the fresh fuel.

TABLE 8.4.2

Measured Uranium Isotopic Composition of
Yankee Core I Pre-Irradiated Fuel

<u>Determination</u>	<u>Tracerlab/a</u>	<u>GE/a</u>	<u>WAPD/b</u>	<u>ORNL/b</u>	<u>WAFD/c</u>	<u>Mallinckrodt/c</u>	<u>Least/d Squares Fit</u>
U-234 a/o	0.02 ± 0.01	0.0214 ± 0.0008					0.021
U-235 a/o	3.53 ± 0.01	3.526 ± 0.036	3.43 ± 0.03	3.448 ± 0.01	3.448	3.433	3.440
U-236 a/o	0.02 ± 0.01	0.0914 ± 0.0008					0.020
U-238 a/o	96.43 ± 0.01	96.43 ± 0.96					96.519

Notes:

- a. It is strongly suspected that the unirradiated fuel pellet (or possibly each of the sample solutions made from it) was contaminated prior to analysis.
- b. Reference 11, page 230.17.
- c. Reference 5, page 15.
- d. Estimated from least squares fit of N^1/N_0^{28} versus $\ln(N_0^{25}/N^{25})$ as inferred from EYC Phase I and Phase II mass spectrometric data.

The extremely consistent, but highly questionable results obtained for U-235 atom percent by both laboratories (Tracerlab and GE) led to a suspicion that the unirradiated "typical" Yankee Core I fuel pellet, or possibly each of the sample solutions made from it, was contaminated prior to analysis. Consequently, another fuel pellet was carefully selected, dissolved, and analyzed by WAPD and by the Oak Ridge National Laboratory (ORNL). As can be observed for the U-235 composition data in Table 8.4.2, the WAPD and ORNL results agree to within the uncertainties in the measurements, with the WAPD results in exact agreement with the Mallinckrodt value, and the ORNL results in exact agreement with the WAFD value. Furthermore, the initial fuel composition inferred from exponential least squares fits of N^i/N_0^{28} (see Appendix A2) give values of U-234 and U-236 atom percent which agree with the Tracerlab and GE data, and a value of U-235 atom percent which agrees with the mean value of the WAPD, ORNL, WAFD, and Mallinckrodt data. It is significant to note that the consistency in the initial composition data estimated from the least squares fits may be a result of the fact that both WAFD and Mallinckrodt fuel pellets were probably sampled to provide the input to the curve-fitting code*. (Extrapolation of curves which relate the net destruction of U-234 to fuel burnup indicates that the initial U-234 atom percent in the fresh fuel was closer to 0.022 a/o, than to the value of 0.021 a/o listed in Table 8.4.2.)

To obtain an estimate of the uncertainties in the inferred values of burnup and fuel characteristics caused by the uncertainties in the assumed initial fuel composition used in the data reduction process, the mass spectrometric data for a pair of Phase I samples (one low burnup and one high burnup sample) were processed with three separate runs through the REBUF data reducing code. In the first run, the initial conditions consistent with the WAFD data were used; in the second run, initial conditions consistent with the Mallinckrodt data were used; in the third run, initial conditions corresponding to the mean of the WAFD and Mallinckrodt data were used. The values of burnup and

*A modified version of the OMNIFIT Code²⁷, supplied by Dr. R. A. Dannels and Mr. D. E. Kusner of the Reactor Physics and Mathematics Section, WAPD.

some fuel characteristics which resulted from the three runs are compared in Table 8.4.3. Notice that for the low burnup sample a $\pm 0.2\%$ uncertainty in assumed value of initial U-235 a/o causes a $\pm 3.6\%$ uncertainty in the inferred value of burnup, and a $\pm 2.7\%$ uncertainty in the inferred value of U-235 atom depletion. If two distinct types of pellets (3.406 w/o WAFD pellets and 3.391 w/o Mallinckrodt pellets) were indeed irradiated, and subsequently sampled, in Yankee Core I, it would explain some of the scatter in the data on, for example, Figures 5.1.1, 5.3.1, 6.1.1, and 6.1.2.

Also, notice that a $\pm 0.2\%$ uncertainty in the assumed value of initial U-235 atom percent could contribute (aside from the uncertainty in the burnup) approximately $\pm 3.6\%$ and $\pm 2.7\%$ of the scatter in the alpha-25 and NGD/MWD data plotted, at low values of burnup, on Figures 7.1.1 and 7.2.1, respectively. In both the figures, it is quite apparent that the scatter in the data is greatest at low values of burnup, and that the scatter diminishes as the burnup increases.

If each fuel pellet sampled in the EYC Program had the exact same initial uranium isotopic composition, the use of consistent, but inaccurate, initial conditions in the data reduction process would introduce a bias or systematic error, rather than a statistical error, in the inferred quantities.

TABLE 8.4.3

Relative Uncertainties in Inferred Burnup and Fuel Characteristics Due to Uncertainties in Pre-Irradiated Fuel Composition

	Core Location K4-C-a1, Fuel Rod #2907 Axial Sample Zone 1, Sample Code #T-32				Core Location F4-C-a6, Fuel Rod #8994 Axial Sample Zone 5, Sample Code #T-89			
	Run 1 WAFD	Run 2 Mallinckrodt	Run 3 Mean	WAFD-Mean Mean x 100%	Run 1 WAFD	Run 2 Mallinckrodt	Run 3 Mean	WAFD-Mean Mean x 100%
<u>Assumed Initial Conditions</u>								
U-234 Atom Percent	0.021	0.021	0.02		0.021	0.021	0.02	
U-235 Atom Percent	3.448	3.433	3.44	0.2	3.448	3.433	3.44	0.2
U-236 Atom Percent	0.020	0.020	0.02		0.020	0.020	0.02	
U-238 Atom Percent	96.511	96.526	96.52	0.01	96.511	96.526	96.52	0.01
<u>Inferred Burnup and Fuel Characteristics</u>								
Isotopic Burnup (MWD/MTU)	2,380	2,220	2,290	3.9	13,230	13,070	13,140	0.7
Cs-137 Burnup (MWD/MTU)	2,160	2,160	2,160	0.0	13,100	13,090	13,100	0.0
U-235 Atom Depletion	0.07992	0.07574	0.07768	2.9	0.3501	0.3471	0.3485	0.4
Total Accumulated Fissions/N _o ²⁸	0.002578	0.002404	0.002484	3.8	0.01426	0.01409	0.01417	0.6
<u>Accumulated Fractional Fissions</u>								
U-235	0.877	0.874	0.876	0.1	0.703	0.700	0.701	0.3
U-238	0.0723	0.0723	0.0723	0.0	0.0776	0.0775	0.0776	0.0
Pu-239	0.0500	0.0536	0.0519	3.7	0.210	0.213	0.211	0.5
Pu-241	0.000183	0.000196	0.000190	3.7	0.00960	0.00971	0.00966	0.6
U-235 Net Destruction (kg/MTU)	2.722	2.568	2.639	3.1	11.92	11.77	11.84	0.7
NGD/MWD	0.563	0.534	0.548	2.7	0.439	0.432	0.435	0.9
Alpha-25	0.262	0.282	0.273	4.0	0.248	0.252	0.250	0.8

9.0 SUMMARY AND CONCLUSIONS

One of the primary objectives of the Yankee Core Evaluation Program is the sampling and subsequent mass spectrometric and radiochemical analysis of a number of locations in the Yankee core to obtain information for the determination of: 1) the U and Pu isotopic composition of the fuel as a function of burnup in the asymptotic and perturbed reactor neutron spectra; 2) the spatial distribution of burnup and fuel isotopes in the rods, in the assemblies, and in the core; 3) the total isotopic inventory of the core; and 4) fuel characteristics, including the specific Pu production, the effective capture-to-fission ratio in U-235, and the net mass of fissile materials destroyed per unit of energy released in the fuel. The measured and inferred spent fuel data presented in this report represent the accomplishment of that objective.

In the asymptotic neutron spectrum, reactor physics data have been obtained over a burnup range of 2,000 to 15,000 MWD/MTU for the Core I fuel, and up to 25,000 MWD/MTU for the Core I fuel which was cycled in Core II for extended burnup. In the perturbed neutron spectra, data have been obtained over a burnup range of 1,200 to 18,000 MWD/MTU for the Core I fuel, and up to 31,000 MWD/MTU in the fuel cycled in Core II.

At a burnup of 8,470 MWD/MTU in the asymptotic spectrum, the U-235 destruction was found to be 8.37 kg/MTU (i.e., a U-235 depletion of 24.8%), while the net production of U-236 was 1.58 kg/MTU. The Pu/U mass ratio was 0.49%, corresponding to a specific production of 4.06 kg/MTU, 0.481 kg/MTU, 0.205 kg/MTU, and 0.012 kg/MTU of Pu-239, Pu-240, Pu-241, and Pu-242, respectively. With the exception of the Pu-242, which is present in the fuel in relatively small quantities, these asymptotic values are consistent with both the calculated total core U and Pu isotopic inventory reported in Reference 18 and the experimentally determined inventory reported in Reference 3. In both the latter cases, consideration is given to the results from the perturbed spectra of the core, as well as the asymptotic spectrum. Thus, in the case of the batch-loaded Yankee core, it is concluded that the effects of the perturbed neutron spectra are small with regard to the total-core U and Pu inventory. Approximately 13% of the 23,142 fuel rods in the Yankee core are adjacent to the water slots which surround the control rod positions.

At a burnup of 25,000 MWD/MTU in the asymptotic spectrum, the U-235 destruction was found to be 18.8 kg/MTU (i.e., a depletion of 55.5%), while the net production of U-236 was 3.46 kg/MTU. The Pu/U mass ratio at this burnup was 1.17%, corresponding to a specific production of 7.75 kg/MTU, 1.86 kg/MTU, 1.34 kg/MTU, and 0.25 kg/MTU of Pu-239, Pu-240, Pu-241, and Pu-242, respectively.

With respect to the economic considerations of production and utilization of fissile materials, it was found that for every two grams of U-235 destroyed in the asymptotic spectrum, approximately one net gram of fissile plutonium was produced. The resulting net number of grams of fissile materials destroyed per unit of energy released in the fuel was a function of burnup, ranging from 0.55 gm/MWD over 2,000 MWD/MTU burnup, to 0.40 gm/MWD over 25,000 MWD/MTU burnup. Over the average core burnup of 8,470 MWD/MTU, the net number of grams of fissile materials destroyed per megawatt day was 0.48.

The consistency and reliability of the data obtained in Phases I and II of the EYC Program have been established in four separate ways: 1) the isotopic data have been obtained from two laboratories, with some cross check data from a third laboratory. Generally, the resulting data have been found to be in statistical agreement. In particular, the results obtained by the laboratories for repeated analysis of one specific sample (monitor solutions) indicate that the analyses of the laboratories are in excellent agreement. 2) The consistency of the data has been verified by comparing values of burnup inferred from the mass spectrometric data, with corresponding values of burnup obtained from radiochemical (Cs-137) analysis. The two methods give values of burnup which agree within the uncertainty of the measurements, however, a small bias of 2 to 3% is evident in the 25,000 to 30,000 MWD/MTU burnup range. While this discrepancy does not significantly affect the Phase I and Phase II results, it may be significant for the samples which will be obtained in Phase III of the EYC program, which considers spent fuel samples which have undergone burnup to approximately 46,000 MWD/MTU. 3) An integral check on the isotopic and Cs-137 data has been obtained by comparing the values of burnup inferred from flux wire burnup distributions used in conjunction with the total core burnup determined from plant calorimetric measurements. Comparisons indicate that the destructive and non-destructive methods agree to within the uncertainty of the

measurements (about 6% for each method). 4) A partial check on the isotopic data has been made with the results of the X-ray fluorescence analysis for the Pu/U mass ratio. For the Phase I samples, the Pu/U data from the two methods are in statistical agreement. However, for the higher values of Pu/U found in the Phase II samples, there is an apparent 2 to 3% bias in the data obtained by the two methods which, as in the case of the isotopic versus Cs-137 burnup comparison, could be significantly worse in the Phase III samples.

Generally, it was found that the scatter in the data used to define certain functional relationships (e.g., Pu/U versus U-235 atom depletion, the effective capture-to-fission ratio in U-235 versus fuel burnup) was greater than expected, based on the accuracies quoted for the measurements. Investigation of this anomaly, through the evaluation of the results obtained for repeated analyses of a monitor sample, revealed that roughly 40 to 60% of the scatter was due to statistical errors in the experimental methods. A large portion of the uncertainty in the data was due to uncontrollable factors such as engineering and manufacturing tolerances on core components and fuel, and spectral effects caused by non-uniform moderator density and possible fuel rod bowing. Nevertheless, due to the large number of core locations sampled, the statistical error was sufficiently reduced so that the relationships between various fuel parameters were adequately defined.

The spent fuel data presented in this report have been useful in the evaluation of reactor core design methods, as well as in the direct evaluation^{/18} of core performance.

ACKNOWLEDGMENTS

The author gratefully acknowledges the efforts of the many individuals of the Westinghouse Atomic Power Division who were responsible for the acquisition of the spent fuel data presented in this report.

Appreciation is extended to the following: Messrs. H. W. Graves, Jr., and F. L. Langford, Jr., for their guidance during the execution of this program; Messrs. J. Corbett, W. Hudec, J. Jedruch, and C. G. Poncelet, who contributed in various ways to the completion of this study; Messrs. P. G. Lacey and R. E. Radcliffe, for making available the results of the LEOPARD calculations; and Messrs. G. K. Clayton and J. Cornyn who assisted in the preparation of this report.

Special thanks are extended to Margie Peterman and Dolly Stafiej for their excellent efforts in the preparation of both the rough draft and final version of this report.

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APPENDIX A1

Results of Mass Spectrometric and Radiochemical Analyses of EYC Phase I and Phase II Spent Fuel Samples

The results of mass spectrometric and radiochemical analyses of 273 spent fuel samples are given in Table A1.1, where the samples are grouped in 4 major categories according to the neutron spectrum. These include the asymptotic spectrum in assembly centers, the perturbed spectrum in assembly corners, the intermediate spectrum along the diagonal between the assembly corner and center, and the spectrum on the core periphery, adjacent to the water reflector.

In each of the 4 major groups, the fuel rods which were sampled are listed in alphabetical order according to their core location designation. The method used to indicate the core location of a particular fuel rod is given in Reference 9, and is briefly summarized in the following paragraph.

Each fuel rod is designated with an alpha-numeric symbol which consists of three parts: the first part is an alpha-numeric matrix notation which indicates one of the 76 fuel assemblies in the core; the second part is a compass-point notation which indicates one of the 9 subassemblies within the fuel assembly; and the third part is an alpha-numeric matrix notation which indicates one of the 25, 30, or 36 (35 fuel rods in a central subassembly) fuel rods within the subassembly. When the fuel assemblies are oriented as shown in Figure 3.2.2, the upper left subassembly in each assembly is always "NW", and the upper left fuel rod in each subassembly is always "al". For example, H3-C-f6 refers to the fuel rod in the right lower corner of the central subassembly of Fuel Assembly H3; G4-NW-al refers to the fuel rod in the left upper corner of the upper left subassembly in Fuel Assembly G4.

The chronological order of sampling and analysis can be found in the various EYC progress reports listed in the references.

A fuel rod identification number was engraved on each fuel tube during fabrication. The identification number listed for each fuel rod in Table A1.1 was read from the rod in the hot cell during the post-irradiation visual examination.

RESULTS OF MASS SPECTROMETRIC AND RADIOCHEMICAL ANALYSES OF EYC PHASE I AND PHASE II SPENT FUEL SAMPLES

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number ^a	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc.	Pu-239/U-238	Cs-137 ^c	Sr-90 ^c
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	(a/ml x 10 ⁻¹⁶)	(a/a x 10 ³)	(dpm/gmU x 10 ⁻¹⁰)	(dpm/gmU x 10 ⁻¹⁰)
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES															
C3-C-a1	9578	3A ^d 3B	T-132	.0184 ± .0005 ^e	2.551 ± .004	.2066 ± .0025	97.224 ± .005	84.056 ± .037 ^e	10.662 ± .037	4.949 ± .017	.332 ± .003	6.128 ± .010 ^e	4.570 ± .042 ^e	6.63 ± .33 ^f	
			T-133	.0181 ± .0005	2.544 ± .004	.2046 ± .0013	97.234 ± .004	84.032 ± .027	10.676 ± .024	4.958 ± .019	.335 ± .002	7.284 ± .159	4.682 ± .111	6.64 ± .33	
C3-C-f6	9595	1 3 5 6	T-100	.0198 ± .0006	2.922 ± .020	.1338 ± .0012	96.924 ± .021	89.782 ± .019	7.753 ± .018	2.374 ± .008	.0906 ± .0007	6.976 ± .071	3.10 ± .06	3.88 ± .19	
			T-101	.0180 ± .0002	2.439 ± .012	.2362 ± .0015	97.307 ± .014	81.706 ± .028	11.732 ± .025	6.064 ± .020	.498 ± .005	5.794 ± .033	5.344 ± .080	8.06 ± .40	
			T-102	.0180 ± .0001	2.432 ± .005	.2337 ± .0009	97.316 ± .005	81.966 ± .021	11.682 ± .019	5.883 ± .013	.469 ± .002	4.693 ± .020	5.19 ± .08	7.87 ± .39	
			T-103	.0195 ± .0001	2.808 ± .008	.1605 ± .0007	97.012 ± .008	87.965 ± .046	8.790 ± .044	3.078 ± .021	.168 ± .007	6.288 ± .038	3.49 ± .05	4.66 ± .23	
C8-C-a6	569	3A ^d 3B	T-134	.0196 ± .0002	2.574 ± .003	.1942 ± .0005	97.212 ± .003	84.274 ± .041	10.540 ± .035	4.872 ± .029	.314 ± .005	5.746 ± .010	4.528 ± .053	6.61 ± .33	
			T-135	.0194 ± .0002	2.574 ± .004	.2002 ± .0007	97.207 ± .004	84.160 ± .019	10.621 ± .014	4.894 ± .017	.324 ± .003	6.508 ± .010	4.552 ± .050	6.60 ± .33	
C8-C-f1	5551	1 3 5 6	T-105	.0200 ± .0005	2.904 ± .004	.1351 ± .0015	96.941 ± .004	89.477 ± .025	7.999 ± .025	2.432 ± .009	.0918 ± .0006	5.746 ± .013	3.04 ± .04	3.99 ± .20	
			T-106	.0184 ± .0002	2.423 ± .010	.2319 ± .0012	97.326 ± .010	81.676 ± .015	11.780 ± .012	6.053 ± .013	.4909 ± .0013	5.531 ± .018	5.202 ± .075	8.16 ± .41	
			T-107	.0189 ± .0001	2.450 ± .011	.2287 ± .0008	97.303 ± .011	81.902 ± .033	11.708 ± .024	5.912 ± .030	.477 ± .006	5.480 ± .010	5.098 ± .074	8.15 ± .41	
			T-108	.0198 ± .0002	2.772 ± .013	.1548 ± .0007	97.053 ± .013	87.899 ± .016	8.829 ± .013	3.125 ± .011	.1468 ± .0006	5.895 ± .025	3.44 ± .05	4.76 ± .24	
E5-C-a6	7992	1 3 ^g 3 ^g 3 ^g 5 6	T-11	.0182 ± .0003	2.666 ± .013	.1700 ± .0010	97.146 ± .014	85.863 ± .051	9.776 ± .049	4.094 ± .025	.267 ± .004	6.391 ± .058	3.52 ± .13	4.97 ± .25	
			T-12	.0172 ± .0002	2.342 ± .007	.2364 ± .0013	97.404 ± .007	79.864 ± .031	12.517 ± .018	6.958 ± .020	.661 ± .004	6.103 ± .020 ^e	5.842 ± .220 ^e	9.95 ± .50	
			N-10-1 ^h	.015 ± .003 ^e	2.26 ± .03	.26 ± .02	97.47 ± .06	78.80 ± .08 ^e	13.06 ± .08	7.40 ± .04	.733 ± .002	78.86 ^e	5.727 ^e		
			N-10-2	.016 ± .003	2.26 ± .03	.26 ± .02	97.47 ± .06	78.73 ± .08	13.12 ± .08	7.41 ± .04	.739 ± .002	79.16	5.685		
			T-13	.0159 ± .0003	2.205 ± .009	.2554 ± .0013	97.524 ± .010	79.124 ± .152	12.904 ± .095	7.239 ± .062	.732 ± .012	7.216 ± .038	5.541 ± .191	9.95 ± .50	
			T-14	.0179 ± .0005	2.639 ± .012	.1799 ± .0018	97.163 ± .013	85.676 ± .102	10.011 ± .072	4.074 ± .037	.238 ± .004	6.892 ± .035	4.007 ± .144	5.82 ± .29	
E5-C-f1	7980	1 3 ^g 3 ^g 3 ^g 5 6	G-15	.018 ± .004 ^e	2.75 ± .02	.152 ± .004	97.08 ± .20	87.77 ± .40	8.87 ± .08	3.21 ± .07	.150 ± .004	21.4 ± .8	3.57 ± .08	4.74 ± .24	
			G-1	.0158 ± .0006	2.24 ± .02	.258 ± .002	97.49 ± .62	79.22 ± .48	12.75 ± .08	7.33 ± .49	.701 ± .012	6.17 ± .24	5.81 ± .12	10.48 ± .63	8.17 ± .65
			T-2 ^j	.0165 ± .0002	2.249 ± .020	.2596 ± .0044	97.475 ± .023	78.921 ± .028	12.949 ± .020	7.398 ± .026	.732 ± .004	6.305 ± .060	5.910 ± .096		
			T-2 ^j	.0169 ± .0002	2.272 ± .006	.2601 ± .0010	97.451 ± .007	79.324 ± .065	12.786 ± .032	7.212 ± .057	.677 ± .006	6.222 ± .020	5.904 ± .155		
			G	.0158 ± .0008	2.22 ± .02	.255 ± .004	97.51 ± .46	79.08 ± .80	12.89 ± .12	7.31 ± .06	.718 ± .024	5.48 ± .20	5.67 ± .10	10.11 ± .91	7.78 ± .62
			G-16	.0180 ± .0040	2.64 ± .02	.187 ± .004	97.15 ± .20	85.13 ± .40	10.26 ± .08	4.34 ± .04	.268 ± .006	31.8 ± 1.2	4.06 ± .08	5.96 ± .30	
F4-C-a1	8982	3 ^g 3 ^g 3 ^g	G-2	.0172 ± .0026	2.24 ± .032	.245 ± .008	97.49 ± .96	79.51 ± .76	12.59 ± .30	7.14 ± .15	.762 ± .030	5.65 ± .22	5.85 ± .20	9.22 ± .46	
			T-9 ^j	.0168 ± .0003	2.282 ± .005	.2578 ± .0012	97.443 ± .006	79.541 ± .064	12.615 ± .054	7.155 ± .048	.688 ± .018	5.412 ± .015	6.001 ± .210		
			N-5 ^j	.017 ± .003	2.30 ± .03	.26 ± .02	97.42 ± .06	80.55 ± .16	12.08 ± .16	6.71 ± .09	.656 ± .020	5.434	5.788		
F4-C-a6	8994	1 3 5 6	T-87	.0181 ± .0001	2.834 ± .007	.1494 ± .0012	96.988 ± .007	88.206 ± .018	8.610 ± .017	3.045 ± .009	.1395 ± .0010	5.220 ± .015	3.511 ± .043	4.62 ± .23	
			T-88	.0164 ± .0002	2.300 ± .007	.2527 ± .0013	97.431 ± .008	79.774 ± .045	12.589 ± .033	6.981 ± .043	.6560 ± .0040	5.397 ± .020	5.798 ± .062	9.54 ± .48	
			T-89	.0161 ± .0001	2.289 ± .007	.2531 ± .0015	97.442 ± .008	79.463 ± .038	12.749 ± .034	7.110 ± .028	.6784 ± .0050	4.349 ± .018	5.857 ± .108	9.52 ± .48	
			T-90	.0175 ± .0001	2.676 ± .004	.1762 ± .0006	97.130 ± .004	85.835 ± .024	9.911 ± .024	4.022 ± .007	.2308 ± .0030	6.735 ± .013	4.094 ± .042	5.86 ± .29	
F4-C-f1	8977	1 2 3 ^g 3 ^g 4 5 6	T-37	.0173 ± .0002	2.778 ± .011	.1532 ± .0009	97.052 ± .012	87.715 ± .025	8.913 ± .023	3.223 ± .013	.1494 ± .0020	6.004 ± .035	3.504 ± .078	4.48 ± .22	
			T-38	.0170 ± .0004	2.374 ± .009	.2421 ± .0020	97.367 ± .010	80.765 ± .047	12.157 ± .042	6.510 ± .035	.5682 ± .0030	5.971 ± .033	5.485 ± .147	7.81 ± .39	
			T-39	.0172 ± .0003 ^e	2.273 ± .004	.2603 ± .0023	97.450 ± .005	79.668 ± .119	12.629 ± .109	7.022 ± .084	.6804 ± .0090	6.619 ± .020 ^e	5.718 ± .142	8.91 ± .44	
			N-14-1 ^h	.016 ± .003 ^e	2.27 ± .03	.26 ± .02	97.46 ± .06	79.29 ± .06	12.74 ± .06	7.29 ± .02	.686 ± .004	73.77 ^e	5.692 ^e		
			N-14-2	.017 ± .003	2.28 ± .03	.26 ± .02	97.45 ± .06	79.38 ± .10	12.72 ± .10	7.21 ± .07	.686 ± .006	74.04	5.781		
			T-40	.0159 ± .0002	2.236 ± .008	.2660 ± .0018	97.483 ± .009	78.529 ± .052	13.145 ± .048	7.555 ± .035	.7699 ± .0060	5.872 ± .033	5.904 ± .145	9.62 ± .48	
			T-41	.0158 ± .0001	2.251 ± .005	.2618 ± .0011	97.471 ± .006	78.868 ± .056	13.034 ± .052	7.369 ± .043	.7268 ± .0050	6.616 ± .025	5.727 ± .139	9.67 ± .48	
			T-42	.0174 ± .0004	2.630 ± .009	.1816 ± .0023	97.172 ± .009	85.282 ± .031	10.226 ± .029	4.236 ± .019	.2566 ± .0020	3.894 ± .018	4.078 ± .075	6.06 ± .30	
F4-C-f6	19579	3 ^g 3 ^g 3 ^g	G-3	.0189 ± .0026	2.24 ± .02	.262 ± .008	97.48 ± .96	79.71 ± .80	12.53 ± .30	7.05 ± .13	.705 ± .028	6.05 ± .24	5.98 ± .22	10.74 ± .53	
			T-10 ^j	.0186 ± .0002	2.293 ± .003	.2678 ± .0011	97.421 ± .004	79.513 ± .043	12.641 ± .032	7.156 ± .038	.6901 ± .0110	5.804 ± .020	6.164 ± .20		
			N-6 ^j	.017 ± .003	2.31 ± .03	.26 ± .02	97.41 ± .06	79.38 ± .04	12.70 ± .04	7.19 ± .06	.726 ± .02	5.937	5.903		

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	ATOM PERCENT URANIUM ^{a/b}				ATOM PERCENT PLUTONIUM ^c				U-238 Conc.	Pu-239/U-238	Cs-137 ^c	Sr-90 ^c
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	(a/ml x 10 ⁻¹⁶)	(a/a x 10 ³)	(dpm/gmU x 10 ⁻¹⁰)	(dpm/gmU x 10 ⁻¹⁰)
PHASE II, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES															
F5-C-a1	553	1	G-43	.019 ± .004	2.40 ± .02	.231 ± .004	97.34 ± .80	81.63 ± .60	11.86 ± .10	6.02 ± .06	.484 ± .010	24.1 ± .8	5.18 ± .10	8.18 ± .41	9.05 ± .63
		3	G-44	.016 ± .004	1.65 ± .02	.369 ± .008	97.97 ± .80	70.53 ± .60	16.13 ± .12	11.42 ± .11	1.923 ± .020	25.5 ± 1.0	8.01 ± .16	17.14 ± .86	14.40 ± 1.01
		5	G-45	.017 ± .004	1.64 ± .02	.361 ± .008	97.98 ± .80	70.37 ± .60	16.24 ± .12	11.44 ± .11	1.953 ± .040	29.7 ± 1.2	7.84 ± .16	17.08 ± .85	11.32 ± .79
		6	G-46	.017 ± .004	2.12 ± .02	.267 ± .006	97.59 ± .80	77.84 ± .60	13.56 ± .12	7.77 ± .06	.827 ± .016	29.7 ± 1.2	5.89 ± .12	10.63 ± .53	7.73 ± .54
F5-C-f6	541	1	T-117	.0194 ± .0001	2.449 ± .0024	.2328 ± .0004	97.2987 ± .0024	81.590 ± .042	11.812 ± .041	6.109 ± .020	.4890 ± .0036	6.035 ± .006	5.198 ± .054	8.32 ± .42	9.14 ± .64
		2	T-118	.0164 ± .0005	1.8084 ± .0014	.3482 ± .0073	97.827 ± .0073	72.215 ± .033	15.496 ± .023	10.678 ± .036	1.6109 ± .0024	5.847 ± .018	7.459 ± .077	15.35 ± .77	12.79 ± .90
		3	T-119	.0158 ± .0002	1.7097 ± .0087	.3766 ± .0007	97.8978 ± .0087	70.455 ± .060	16.136 ± .047	11.480 ± .062	1.9296 ± .0149	5.938 ± .028	7.968 ± .089	17.57 ± .88	12.62 ± .88
		4	T-120	.0142 ± .0001	1.6435 ± .0020	.3779 ± .0006	97.9645 ± .0021	69.735 ± .018	16.418 ± .018	11.772 ± .009	2.0754 ± .0071	5.257 ± .011	8.186 ± .102	18.50 ± .92	11.77 ± .82
		5	T-121	.0144 ± .0001	1.6810 ± .0020	.3680 ± .0005	97.9366 ± .0021	70.106 ± .022	16.368 ± .020	11.553 ± .019	1.9726 ± .0037	5.779 ± .014	7.742 ± .093	17.06 ± .85	10.71 ± .75
		6	T-122	.0162 ± .0001	2.2031 ± .0061	.2707 ± .0012	97.5099 ± .0062	77.923 ± .012	13.474 ± .009	7.783 ± .010	.8202 ± .0033	6.427 ± .025	6.022 ± .063	10.48 ± .52	6.58 ± .46
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES															
G4-C-a1	0994	1/k	T-43	.0179 ± .0002	2.805 ± .007	.1517 ± .0005	97.026 ± .008	87.865 ± .016	8.810 ± .015	3.174 ± .008	.151 ± .002	4.526 ± .020	3.568 ± .046	4.81 ± .24	6.85 ± .48
		3/k	T-18	.0171 ± .0003	2.354 ± .007	.2322 ± .0011	97.397 ± .008	81.141 ± .186	11.980 ± .120	6.334 ± .071	.544 ± .011	7.216 ± .038	5.433 ± .170	8.544 ± .427	
		3	T-36	.0162 ± .0001	2.379 ± .004	.2366 ± .0004	97.368 ± .004	81.073 ± .032	11.979 ± .032	6.400 ± .015	.547 ± .004	7.219 ± .030	5.260 ± .117	8.505 ± .425	
		3	T-58	.0161 ± .0001	2.371 ± .025	.2342 ± .0004	97.378 ± .025	80.829 ± .070	12.131 ± .064	6.489 ± .047	.553 ± .006	7.183 ± .116	5.437 ± .149	8.471 ± .423	
		3	T-91	.0164 ± .0002	2.375 ± .003	.2341 ± .0005	97.375 ± .004	81.048 ± .046	12.000 ± .043	6.412 ± .030	.541 ± .003	7.165 ± .013	5.551 ± .057	8.563 ± .428	
		3	T-104	.0162 ± .0001	2.373 ± .008	.2354 ± .0010	97.375 ± .009	80.923 ± .021	12.057 ± .019	6.466 ± .012	.554 ± .006	7.208 ± .030	5.411 ± .057	8.36 ± .42	
		3	T-144	.0164 ± .0002	2.378 ± .004	.2335 ± .0006	97.372 ± .004	81.141 ± .058	11.940 ± .043	6.376 ± .058	.542 ± .008	7.176 ± .020	5.499 ± .047	8.33 ± .42	
		3	T-150	.0164 ± .0001	2.357 ± .005	.2344 ± .0015	97.392 ± .005	80.861 ± .071	12.166 ± .068	6.459 ± .043	.554 ± .008	7.282 ± .018	5.305 ± .066	8.52 ± .51	
		3	G-6	.017 ± .004	2.34 ± .02	.232 ± .006	97.41 ± .40	81.03 ± .80	11.95 ± .12	6.47 ± .06	.550 ± .010	7.30 ± .28	5.40 ± .20	8.45 ± .42	
		3	G-17	.019 ± .004	2.37 ± .02	.230 ± .004	97.38 ± .20	81.22 ± .40	11.86 ± .10	6.38 ± .07	.539 ± .010	20.9 ± .8	5.42 ± .10	8.56 ± .43	
		3	G-42	.018 ± .004	2.37 ± .02	.232 ± .004	97.38 ± .20	80.95 ± .79	12.05 ± .12	6.46 ± .07	.544 ± .010	20.87 ± .82	5.50 ± .12	8.40 ± .42	
		3	G-51	.018 ± .004	2.38 ± .02	.240 ± .004	97.36 ± .80	80.94 ± .60	12.08 ± .12	6.44 ± .07	.542 ± .010	21.0 ± .8	5.48 ± .10	8.44 ± .42	
		3	N-12-1/h	.019 ± .003	2.39 ± .03	.23 ± .02	97.36 ± .06	80.84 ± .06	12.10 ± .06	6.49 ± .02	.562 ± .006	64.19	5.370	8.40 ± .42	
		3	N-12-2	.016 ± .003	2.37 ± .03	.24 ± .02	97.37 ± .06	81.01 ± .04	12.00 ± .04	6.44 ± .02	.549 ± .002	64.42	5.359	8.40 ± .42	
		5	T-44	.0162 ± .0003	2.240 ± .008	.2603 ± .0014	97.484 ± .009	78.418 ± .025	13.176 ± .029	7.625 ± .013	.780 ± .003	5.827 ± .038	6.002 ± .115	10.13 ± .51	
		6/g	T-45	.0174 ± .0004	2.635 ± .006	.1729 ± .0019	97.175 ± .007	85.267 ± .045	10.252 ± .043	4.221 ± .021	.260 ± .004	6.629 ± .023	4.105 ± .074	5.94 ± .30	
		6/g	N-13-1/h	.020 ± .003	2.64 ± .03	.18 ± .02	97.16 ± .06	85.49 ± .04	10.03 ± .04	4.21 ± .02	.263 ± .008	73.95	4.070		
		6/g	N-13-2	.018 ± .003	2.64 ± .03	.18 ± .02	97.17 ± .06	85.48 ± .02	10.05 ± .02	4.21 ± .04	.252 ± .002	73.48	4.076		
G4-C-a6	0985	1	T	.02 ± .01	2.76 ± .01	.16 ± .01	97.06 ± .01	87.74 ± .05	8.88 ± .04	3.22 ± .02	.16 ± .01	4.91 ± .02	3.58 ± .04/m	4.77 ± .29	3.87 ± .31
		3/g	T	.02 ± .01	2.24 ± .01	.26 ± .01	97.48 ± .01	79.17 ± .21	12.82 ± .13	7.29 ± .09	.72 ± .01	4.807 ± .025	5.89 ± .09/m	9.75 ± .58	7.20 ± .58
		5/g	G	.0157 ± .0012	2.19 ± .02	.258 ± .004	97.53 ± .46	78.88 ± 1.21	13.00 ± .12	7.40 ± .06	.717 ± .010	4.74 ± .18	5.97 ± .04/m	9.82 ± .59	7.22 ± .58
		5	T	.02 ± .01	2.22 ± .01	.26 ± .01	97.51 ± .01	79.03 ± .11	12.92 ± .06	7.29 ± .08	.77 ± .02	4.858 ± .025	5.93 ± .07/m	6.95 ± .42	4.71 ± .38
		6	T	.02 ± .01	2.61 ± .01	.18 ± .01	97.18 ± .01	86.36 ± 1.23	9.06 ± .79	4.31 ± .58	.27 ± .03	4.402 ± .025	3.97 ± .12/m		
G4-C-f1	0979	1	T	.02 ± .01	2.80 ± .01	.15 ± .01	97.03 ± .01	88.04 ± .05	8.72 ± .02	3.09 ± .05	.16 ± .01	4.76 ± .02	3.50 ± .04/m	4.36 ± .27	3.54 ± .28
		3/g	T	.02 ± .01	2.23 ± .02	.25 ± .01	97.50 ± .03	79.47 ± .04	12.67 ± .02	7.16 ± .02	.70 ± .01	5.16 ± .15	5.82 ± .18/m	9.19 ± .55	8.22 ± .66
		5/g	G	.0171 ± .0004	2.24 ± .02	.254 ± .002	97.49 ± .44	79.47 ± .82	12.67 ± .12	7.17 ± .06	.689 ± .024	4.78 ± .18	5.82 ± .06/m	9.58 ± .57	7.81 ± .62
		5	T	.02 ± .01	2.24 ± .01	.26 ± .01	97.48 ± .01	79.32 ± .10	12.81 ± .06	7.15 ± .05	.72 ± .02	4.86 ± .02	5.85 ± .07/m	5.89 ± .35	5.38 ± .43
		6	T	.02 ± .01	2.61 ± .01	.18 ± .01	97.20 ± .01	85.55 ± .08	9.95 ± .05	4.20 ± .04	.30 ± .01	4.45 ± .02	4.19 ± .05/m		
G4-C-f6	0982	1	G-22	.019 ± .004	2.79 ± .02	.149 ± .004	97.04 ± .20	88.12 ± .40	8.63 ± .08	3.11 ± .04	.144 ± .004	20.9 ± .8	3.57 ± .08	4.92 ± .25	5.88 ± .35
		3	G-23	.017 ± .004	2.37 ± .02	.237 ± .004	97.37 ± .20	81.15 ± .40	12.01 ± .08	6.28 ± .07	.559 ± .010	25.6 ± 1.0	5.55 ± .10	9.36 ± .47	
		5	G-24	.018 ± .004	2.22 ± .02	.267 ± .006	97.49 ± .20	78.57 ± .40	13.10 ± .08	7.56 ± .07	.768 ± .014	17.6 ± .6	6.21 ± .12	10.41 ± .62	
		6	G-25	.018 ± .004	2.63 ± .02	.178 ± .004	97.17 ± .20	85.46 ± .40	10.06 ± .08	4.23 ± .09	.250 ± .006	20.8 ± .8	4.16 ± .08		

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code ^a Number	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc.	Pu-239/U-238	Cs-137 ^c	Sr-90 ^c
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	(a/ml x 10 ⁻¹⁶)	(a/a x 10 ³)	(dpm/gmU x 10 ⁻¹⁰)	(dpm/gmU x 10 ⁻¹⁰)
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES															
H3-C-a1	8537	5	G	.0175 ± .0016	2.44 ± .02	.204 ± .004	97.34 ± .76	82.93 ± .74	11.26 ± .08	5.41 ± .04	.395 ± .012	5.69 ± .22	4.90 ± .04 ^m	7.59 ± .46	4.88 ± .39
H3-C-a6	8510	Comp ⁿ	T	.02 ± .01	2.58 ± .01	.20 ± .01	97.20 ± .01	83.63 ± .06	10.86 ± .04	5.11 ± .02	.40 ± .01	7.14 ± .05	5.11 ± .07 ^m		
H3-C-a6	8510	1	T	.02 ± .01	2.89 ± .01	.13 ± .01	96.96 ± .01	89.77 ± .06	7.80 ± .06	2.33 ± .03	.10 ± .01	5.04 ± .02	3.02 ± .04 ^m	3.67 ± .18	2.37 ± .19
		2	T	.02 ± .01	2.52 ± .01	.20 ± .01	97.26 ± .01	83.36 ± .07	11.06 ± .07	5.21 ± .03	.36 ± .01	4.50 ± .02	4.79 ± .06 ^m	7.01 ± .35	4.33 ± .34
		3 ^g	T	.02 ± .01	2.43 ± .01	.22 ± .01	97.33 ± .01	81.92 ± .06	11.68 ± .06	5.94 ± .03	.46 ± .01	4.63 ± .02	5.17 ± .06 ^m	7.90 ± .39	5.08 ± .41
		4 ^g	G	.0192 ± .0012	2.468 ± .024	.2270 ± .0136	97.11 ± .96	81.64 ± .60	11.805 ± .086	6.041 ± .080	.515 ± .024	4.36 ± .16	5.25 ± .02 ^m	7.87 ± .39	5.22 ± .42
		4	T	.02 ± .01	2.39 ± .01	.22 ± .01	97.38 ± .01	81.44 ± .04	11.96 ± .03	6.12 ± .04	.49 ± .01	4.55 ± .02	5.30 ± .06 ^m		
		5	T	.02 ± .01	2.42 ± .01	.21 ± .01	97.35 ± .01	82.04 ± .04	11.68 ± .03	5.82 ± .02	.46 ± .01	3.90 ± .02	5.14 ± .06 ^m	7.90 ± .39	5.44 ± .44
H3-C-f1	8528	6	T	.02 ± .01	2.76 ± .01	.14 ± .01	97.08 ± .01	88.02 ± .03	8.82 ± .03	3.01 ± .02	.15 ± .01	4.73 ± .02	3.51 ± .05 ^m	4.51 ± .22	2.52 ± .20
		1	G	.0226 ± .0014	2.99 ± .02	.115 ± .002	96.87 ± .60	91.42 ± .40	6.76 ± .02	1.76 ± .02	.0548 ± .0002	5.64 ± .22	2.55 ± .04 ^m	2.92 ± .18	3.01 ± .24
		3A ^d	T-142	.0192 ± .0003	2.577 ± .007	.1998 ± .0011	97.204 ± .007	84.050 ± .009	10.717 ± .006	4.900 ± .009	.333 ± .002	5.450 ± .013	4.625 ± .04 ^m	6.74 ± .34	
		3	G	.0210 ± .0004	2.56 ± .02	.202 ± .002	97.22 ± .48	83.98 ± .56	10.83 ± .10	4.84 ± .19	.345 ± .004	5.82 ± .24	4.62 ± .04 ^m	6.26 ± .38	5.53 ± .44
		3B	T-143	.0192 ± .0004	2.736 ± .018	.1778 ± .0013	97.067 ± .020	85.599 ± .025	9.860 ± .021	4.272 ± .018	.270 ± .002	4.658 ± .051	4.087 ± .06 ^m	5.84 ± .30	
		5 ^g	G	.0204 ± .0004	2.58 ± .02	.202 ± .004	97.20 ± .38	84.28 ± .80	10.65 ± .08	4.76 ± .04	.310 ± .006	6.63 ± .26	4.54 ± .04 ^m	5.80 ± .35	4.28 ± .34
H3-C-f6	8574	5 ^g	T	.020 ± .01	2.62 ± .01	.20 ± .01	97.15 ± .01	84.19 ± .04	10.72 ± .03	4.77 ± .02	.32 ± .01	6.76 ± .05	4.56 ± .06 ^m		
		6	G	.0184 ± .0008	2.89 ± .02	.126 ± .002	96.96 ± .66	89.98 ± .48	7.68 ± .04	2.26 ± .02	.0845 ± .0028	6.75 ± .26	2.96 ± .04 ^m	3.79 ± .23	2.71 ± .22
		1	T-138	.0210 ± .0005	2.955 ± .005	.1225 ± .0002	96.901 ± .005	90.721 ± .014	7.204 ± .008	2.008 ± .015	.067 ± .001	7.401 ± .030	2.685 ± .029	3.34 ± .17	
		3	T-139	.0187 ± .0003	2.466 ± .005	.2146 ± .0017	97.300 ± .005	82.930 ± .038	11.204 ± .035	5.462 ± .026	.403 ± .002	6.573 ± .023	4.792 ± .047	7.19 ± .36	
H3-C-f6	8574	5	T-140	.0174 ± .0002	2.504 ± .012	.2036 ± .0012	97.275 ± .013	83.199 ± .081	11.902 ± .076	5.321 ± .047	.387 ± .005	6.991 ± .048	4.843 ± .053	7.24 ± .36	
		6	T-141	.0186 ± .0002	2.847 ± .009	.1356 ± .0004	96.993 ± .009	89.073 ± .024	8.188 ± .023	2.629 ± .010	.110 ± .002	5.607 ± .020	3.174 ± .034	4.16 ± .21	
H4-C-a1	2021	1	G-11	.021 ± .004	2.82 ± .02	.152 ± .004	97.00 ± .20	88.87 ± .40	8.24 ± .08	2.76 ± .02	.120 ± .002	19.8 ± .8	3.26 ± .06	3.97 ± .20	
		3	G-4	.018 ± .004	2.34 ± .02	.249 ± .006	97.40 ± .40	80.87 ± .80	11.96 ± .12	6.62 ± .06	.544 ± .010	3.69 ± .14	5.44 ± .20	10.25 ± .51	
		5	G-5	.019 ± .004	2.36 ± .02	.253 ± .006	97.37 ± .40	80.42 ± .80	12.32 ± .12	6.66 ± .06	.598 ± .012	4.95 ± .20	5.50 ± .20	9.25 ± .46	
		6	G-12	.019 ± .004	2.70 ± .02	.174 ± .004	97.10 ± .20	87.06 ± .40	9.22 ± .08	3.54 ± .04	.175 ± .004	2.59 ± .10	3.78 ± .08	4.79 ± .29	
H5-C-a1	1000	1 ^g	T-25	.0181 ± .0001	2.800 ± .010	.1448 ± .0006	97.037 ± .011	88.397 ± .041	8.540 ± .029	2.938 ± .021	.125 ± .001	6.427 ± .030	3.362 ± .087	4.49 ± .22	
		3 ^g	G-1	.0173 ± .0038	2.30 ± .032	.240 ± .008	97.47 ± .98	80.03 ± .80	12.45 ± .30	6.84 ± .13	.687 ± .056	7.28 ± .30	5.84 ± .20	9.33 ± .47	
		3 ^g	T-8	.0170 ± .0002	2.334 ± .008	.2523 ± .0014	97.397 ± .009	80.171 ± .037	12.394 ± .035	6.818 ± .020	.617 ± .008	7.206 ± .040 ^g	5.829 ± .16		
		3 ^g	N-4 ⁱ	.021 ± .003	2.38 ± .03	.25 ± .02	97.35 ± .06	80.22 ± .04	12.38 ± .04	6.77 ± .06	.63 ± .02	7.10 ± .10 ^g	5.71 ± .10		
		5	T-26	.0166 ± .0001	2.255 ± .006	.2480 ± .0010	97.480 ± .006	79.576 ± .042	12.741 ± .022	7.019 ± .034	.664 ± .003	6.548 ± .023	5.552 ± .121	9.65 ± .48	
		5 ^g	N-11-1 ^h	.017 ± .003	2.30 ± .03	.25 ± .02	97.43 ± .06	79.52 ± .04	12.77 ± .04	7.05 ± .02	.666 ± .010	75.98 ± .023	5.661 ± .121		
H8-C-a1	21305	5 ^g	N-11-2	.018 ± .003	2.31 ± .03	.25 ± .02	97.42 ± .06	79.33 ± .06	12.80 ± .06	7.19 ± .04	.676 ± .010	76.68 ± .023	5.552 ± .121		
		6	T-27	.0185 ± .0003	2.664 ± .005	.1714 ± .0015	97.147 ± .006	86.218 ± .071	9.677 ± .050	3.888 ± .025	.218 ± .004	8.233 ± .020	4.046 ± .089	4.87 ± .24	
		1	T-113	.0193 ± .0001	2.889 ± .010	.1318 ± .0008	96.960 ± .010	89.960 ± .046	7.645 ± .043	2.312 ± .021	.0835 ± .0013	6.662 ± .025	3.02 ± .04	3.87 ± .19	
		3	T-114	.0174 ± .0009	2.430 ± .003	.2273 ± .0042	97.326 ± .005	82.012 ± .029	11.594 ± .025	5.924 ± .023	.4701 ± .0022	6.171 ± .010	5.206 ± .074	8.19 ± .41	
H8-C-a1	21305	5	T-115	.0183 ± .0007	2.437 ± .004	.2259 ± .0032	97.319 ± .005	82.08 ± .14	11.664 ± .046	5.79 ± .17	.465 ± .003	5.665 ± .018	5.164 ± .075	8.15 ± .41	
		6	T-116	.0192 ± .0034	2.753 ± .021	.1527 ± .0025	97.075 ± .021	88.381 ± .023	8.539 ± .023	2.951 ± .007	.1290 ± .0014	5.146 ± .043	3.322 ± .054	4.53 ± .23	
H8-C-f6	1306	3A ^d	T-136	.0181 ± .0004	2.554 ± .007	.2014 ± .0014	97.227 ± .008	84.415 ± .020	10.514 ± .014	4.757 ± .020	.313 ± .003	3.793 ± .005	4.249 ± .061	6.47 ± .32	
		3B	T-137	.0181 ± .0003	2.549 ± .008	.2038 ± .0007	97.229 ± .008	84.190 ± .070	10.643 ± .067	4.854 ± .038	.313 ± .005	4.096 ± .015	4.579 ± .069	6.55 ± .33	

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number ^a	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc.	Pu-239/U-238	Cs-137 ^c	Sr-90 ^c
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	(a/ml x 10 ⁻¹⁶)	(a/a x 10 ³)	(dpm/gmU x 10 ⁻¹⁰)	(dpm/gmU x 10 ⁻¹⁰)
J3-C-a1	3477	3/G	G	.0202 ± .0004	2.92 ± .02	.146 ± .002	96.92 ± .66	88.94 ± .32	8.20 ± .04	2.75 ± .02	.114 ± .006	5.67 ± .22	3.40 ± .04	4.11 ± .25	
			T	.0199 ± .0002	2.811 ± .011	.1503 ± .0021	97.019 ± .012	88.839 ± .021	8.277 ± .020	2.765 ± .010	.119 ± .002	5.720 ± .040	3.350 ± .046		
			T-1 ¹	.0204 ± .0003	2.836 ± .020	.1506 ± .0014	96.993 ± .021	88.975 ± .041	8.207 ± .028	2.711 ± .024	.107 ± .003	5.655 ± .053	3.38 ± .09		
			N-1 ¹	.017 ± .003	2.86 ± .03	.15 ± .02	96.97 ± .06	88.93 ± .06	8.17 ± .06	2.78 ± .04	.117 ± .014	5.69	3.53		
			T-3 ¹	.0202 ± .0002	2.837 ± .018	.1494 ± .0012	96.994 ± .019	89.01 ± .13	8.204 ± .094	2.666 ± .040	.117 ± .005	5.690 ± .043	3.39 ± .07		
			N-2 ¹	.021 ± .003	2.93 ± .03	.16 ± .02	96.90 ± .06	88.98 ± .02	8.18 ± .02	2.72 ± .02	.118 ± .004	5.49	3.38		
			G	.0194 ± .0008	2.92 ± .02	.143 ± .002	96.91 ± .44	89.29 ± .46	8.04 ± .04	2.57 ± .02	.106 ± .002	5.16 ± .22	3.31 ± .04		
														4.05 ± .25	4.10 ± .33
J3-C-f6	3480	1	T-19	.0206 ± .0003	3.111 ± .002	.0844 ± .0003	96.784 ± .002	94.574 ± .037	4.636 ± .030	0.7725 ± .0132	.0171 ± .0007	6.950 ± .008	1.649 ± .058	1.46 ± .07	
			T-20	.0199 ± .0004	2.812 ± .006	.1463 ± .0028	97.022 ± .007	89.234 ± .063	8.044 ± .057	2.612 ± .036	.1097 ± .0023	10.60 ± .06	3.260 ± .092	3.24 ± .16	
			T-21	.0202 ± .0003	2.844 ± .010	.1389 ± .0022	96.977 ± .011	89.310 ± .249	8.162 ± .177	2.429 ± .135	.0998 ± .0050	7.871 ± .046	3.111 ± .084	3.62 ± .18	
			T-22	.0212 ± .0008	3.083 ± .006	.0943 ± .0008	96.802 ± .007	93.525 ± .074	5.441 ± .051	1.006 ± .049	.0276 ± .0017	3.180 ± .010	1.858 ± .059	2.77 ± .14	
J4-C-a1	305	1	T	.02 ± .01	2.96 ± .01	.11 ± .01	96.92 ± .01	91.12 ± .04	6.97 ± .03	1.85 ± .02	.06 ± .01	5.14 ± .02	2.63 ± .03 ^m	2.96 ± .18	2.38 ± .19
			T	.02 ± .01	2.54 ± .02	.20 ± .01	97.24 ± .02	83.72 ± .11	10.87 ± .07	5.06 ± .04	.36 ± .01	5.84 ± .05	4.69 ± .06 ^m	6.60 ± .40	5.07 ± .40
			G	.0171 ± .0014	2.52 ± .02	.1912 ± .0030	97.27 ± .68	84.13 ± 1.62	10.77 ± .20	4.763 ± .076	.339 ± .046	6.18 ± .12	4.58 ± .04 ^m	6.67 ± .40	5.22 ± .42
			T	.02 ± .01	2.51 ± .01	.20 ± .01	97.28 ± .01	83.81 ± .06	10.90 ± .04	4.96 ± .03	.34 ± .01	6.38 ± .02	4.67 ± .05 ^m		
			T	.02 ± .01	2.82 ± .01	.13 ± .01	97.03 ± .01	89.64 ± .04	7.89 ± .03	2.37 ± .01	.10 ± .01	5.29 ± .02	3.06 ± .04 ^m	3.78 ± .23	2.61 ± .21
J4-C-f6	0918	1/G	T-73	.0190 ± .0001	3.029 ± .009	.1085 ± .0009	96.844 ± .009	91.647 ± .039	6.640 ± .039	1.664 ± .013	.0484 ± .0008	5.425 ± .023	2.445 ± .040	3.12 ± .16	
			G	.0175 ± .0012	2.59 ± .02	.213 ± .004	97.18 ± .38	84.36 ± .88	10.52 ± .10	4.75 ± .02	.370 ± .004	2.02 ± .08	4.65 ± .10	6.96 ± .41	
			T-5 ¹	.0188 ± .0009	2.608 ± .007	.1964 ± .0020	97.177 ± .008	84.582 ± .103	10.427 ± .070	4.698 ± .035	.292 ± .007	5.769 ± .023	4.516 ± .098		
			N-7 ¹	.017 ± .003	2.65 ± .03	.20 ± .02	97.14 ± .06	83.92 ± .04	10.48 ± .04	5.28 ± .02	.32 ± .02	5.699	4.758		
			G	.0178 ± .0008	2.75 ± .02	.185 ± .002	97.05 ± .58	84.81 ± .96	10.36 ± .10	4.51 ± .06	.312 ± .012	6.10 ± .20	4.40 ± .12	6.36 ± .38	5.87 ± .47
			T-6 ¹	.0178 ± .0001	2.594 ± .007	.1906 ± .0008	97.198 ± .007	84.724 ± .088	10.461 ± .058	4.534 ± .044	.280 ± .003	6.037 ± .020	4.645 ± .126		
			N-8 ¹	.018 ± .003	2.62 ± .03	.20 ± .02	97.17 ± .06	84.50 ± .06	10.53 ± .06	4.60 ± .09	.36 ± .02	6.150	4.469		
			T-74	.0183 ± .0002	2.922 ± .005	.1246 ± .0014	96.936 ± .005	90.114 ± .017	7.602 ± .016	2.203 ± .007	.0806 ± .0008	6.115 ± .013	2.836 ± .040	3.78 ± .19	
J5-C-a1	3422	1	T-69	.0196 ± .0004	2.911 ± .004	.1309 ± .0009	96.938 ± .005	90.449 ± .014	7.367 ± .014	2.110 ± .004	.074 ± .001	12.27 ± .03	2.834 ± .040	3.12 ± .16	
			T-70	.0183 ± .0002	2.487 ± .004	.2136 ± .0015	97.281 ± .005	83.093 ± .059	11.102 ± .056	5.415 ± .034	.390 ± .004	5.204 ± .018	4.958 ± .062	7.43 ± .37	
			T-71	.0181 ± .0003	2.495 ± .006	.2192 ± .0012	97.258 ± .007	83.075 ± .041	11.141 ± .043	5.386 ± .011	.398 ± .002	5.706 ± .020	4.820 ± .070	7.26 ± .36	
			T-72	.0191 ± .0001	2.845 ± .002	.1479 ± .0005	96.988 ± .003	89.061 ± .018	8.142 ± .017	2.688 ± .007	.109 ± .002	8.499 ± .015	3.229 ± .033	3.74 ± .18	
K4-C-a1	2907	1	T-32	.0209 ± .0005	3.185 ± .004	.0766 ± .0005	96.717 ± .005	95.249 ± .019	4.164 ± .019	0.575 ± .005	.0123 ± .0004	6.100 ± .018	1.421 ± .023	1.56 ± .08	
			T-33	.0198 ± .0003	2.949 ± .008	.1189 ± .0008	96.912 ± .008	91.697 ± .012	6.596 ± .011	1.657 ± .004	.0502 ± .0004	6.472 ± .028	2.463 ± .027	2.89 ± .14	
			N-15-1 ^h	.020 ± .003	3.00 ± .03	.12 ± .02	96.86 ± .06	91.72 ± .06	6.57 ± .06	1.66 ± .04	.051 ± .002	70.71	2.515		
			N-15-2	.021 ± .003	3.00 ± .03	.13 ± .02	96.86 ± .06	91.65 ± .02	6.60 ± .02	1.70 ± .04	.051 ± .002	71.24	2.498		
			G	.0189 ± .0010	2.88 ± .02	.1270 ± .0018	96.97 ± .32	90.78 ± .38	7.12 ± .30	2.02 ± .02	.0714 ± .0010	6.21 ± .22	2.85 ± .04	3.48 ± .21	3.17 ± .25
			T-4 ¹	.0204 ± .0003	2.942 ± .006	.1283 ± .0005	96.910 ± .006	90.710 ± .040	7.234 ± .025	1.980 ± .029	.0752 ± .0044	6.318 ± .023	2.80 ± .05		
			N-9 ¹	.020 ± .003	3.01 ± .03	.13 ± .01	96.83 ± .06	90.72 ± .04	7.19 ± .04	2.02 ± .02	.069 ± .016	6.148	2.87		
			T-34	.0195 ± .0001	2.937 ± .004	.1337 ± .0011	96.910 ± .004	90.344 ± .044	7.435 ± .043	2.142 ± .016	.0788 ± .0013	6.368 ± .010	2.844 ± .035	3.52 ± .18	
			G	.0204 ± .0004	2.95 ± .02	.122 ± .002	96.91 ± .58	91.21 ± .54	6.89 ± .04	1.84 ± .04	≤ .07	4.01 ± .16	2.73 ± .06	3.50 ± .18	2.75 ± .22
			T-24	.0224 ± .0017	2.952 ± .015	.1254 ± .0012	96.900 ± .016	91.094 ± .167	6.836 ± .120	2.005 ± .073	.0652 ± .0060	3.972 ± .023	2.715 ± .074		
			T-35	.0199 ± .0002	3.143 ± .003	.0828 ± .0007	96.755 ± .003	94.616 ± .015	4.627 ± .014	0.740 ± .006	.0164 ± .0007	6.761 ± .010	1.566 ± .026	1.49 ± .07	
			N-16-1 ^h	.022 ± .003	3.21 ± .03	.084 ± .009	96.68 ± .06	94.73 ± .02	4.52 ± .02	0.716 ± .007	.036 ± .004	74.42	1.599		
			N-16-2	.022 ± .003	3.18 ± .03	.086 ± .009	96.72 ± .06	94.75 ± .02	4.51 ± .02	0.710 ± .011	.036 ± .004	74.27	1.608		

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc. (a/ml x 10 ⁻¹⁶)	Pu-239/U-238 (a/a x 10 ³)	Cs-137 ^c (dpm/gmU x 10 ⁻¹⁰)	Sr-90 ^c (dpm/gmU x 10 ⁻¹⁰)
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241					
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES															
K4-C-f6	2905	1	T-15	.0211 ± .0006	3.203 ± .008	.0671 ± .0015	96.709 ± .008	95.861 ± .027	3.663 ± .024	0.4592 ± .0061	Not Detected	4.557 ± .015	1.203 ± .028	1.39 ± .07	
		3	T-16	.0209 ± .0001	2.969 ± .012	.1146 ± .0009	96.896 ± .013	91.886 ± .047	6.468 ± .037	1.598 ± .017		4.607 ± .028	2.371 ± .089	3.25 ± .16	
		5	T-17	.0201 ± .0002	2.978 ± .009	.1110 ± .0007	96.891 ± .010	92.131 ± .023	6.336 ± .017	1.486 ± .013		5.427 ± .023	2.342 ± .059	2.69 ± .13	
		6	T-23	.0208 ± .0004	3.196 ± .006	.0756 ± .0006	96.707 ± .007	95.482 ± .029	3.979 ± .024	0.540 ± .011		10.26 ± .04	1.386 ± .030	1.53 ± .08	
K5-C-a1	0877	1	T-28	.0197 ± .0005	3.075 ± .011	.0816 ± .0004	96.824 ± .012	93.457 ± .010	5.474 ± .009	1.041 ± .006		2.27 ± .01	1.82 ± .04	2.06 ± .10	
		3	T-29	.0176 ± .0002	2.730 ± .012	.1505 ± .0009	97.102 ± .012	87.864 ± .033	8.787 ± .025	3.198 ± .028	8.043 ± .043	3.540 ± .087	4.67 ± .23		
		5	T-30	.0183 ± .0002	2.757 ± .004	.1470 ± .0008	97.078 ± .004	88.077 ± .035	8.693 ± .031	3.078 ± .021	5.723 ± .010	3.465 ± .063	4.67 ± .23		
		6	T-31	.0188 ± .0003	3.024 ± .007	.0931 ± .0015	96.864 ± .007	92.507 ± .014	6.098 ± .012	1.352 ± .009	3.363 ± .015	2.050 ± .046	2.76 ± .14		
K5-C-f6	011	1	G-18	.019 ± .004	3.14 ± .02	.070 ± .014	96.77 ± .20	95.18 ± .40	4.20 ± .04	0.615 ± .013		32.6 ± 1.2	1.54 ± .04	1.79 ± .09	
		3	G-19	.018 ± .004	2.88 ± .02	.124 ± .004	96.97 ± .20	90.38 ± .40	7.32 ± .08	2.22 ± .04		27.7 ± 1.0	2.91 ± .06	4.66 ± .28	
		5	G-20	.018 ± .004	2.88 ± .02	.121 ± .002	96.98 ± .20	90.52 ± .40	7.28 ± .06	2.12 ± .02		30.5 ± 1.2	2.87 ± .06	5.68 ± .28	
		6	G-21	.020 ± .004	3.13 ± .02	.079 ± .014	96.77 ± .20	94.25 ± .40	4.88 ± .06	0.849 ± .016		19.0 ± .8	1.73 ± .04	2.17 ± .11	
PHASE I, PERTURBED NEUTRON SPECTRUM, CORNER SUBASSEMBLIES															
E5-NE-f1	2300	1	G-38	.019 ± .004	2.60 ± .02	.190 ± .004	97.18 ± .20	83.37 ± .84	11.92 ± .12	4.40 ± .04	0.309 ± .006	22.39 ± .88	4.08 ± .08	6.88 ± .34	
		3	G-39	.016 ± .004	2.06 ± .02	.278 ± .006	97.64 ± .20	74.20 ± .73	16.18 ± .16	8.53 ± .09	1.10 ± .02	25.75 ± 1.02	5.49 ± .10	11.24 ± .56	
		5	G-40	.014 ± .002	2.22 ± .02	.253 ± .004	97.51 ± .20	77.50 ± .77	14.56 ± .14	7.16 ± .07	0.779 ± .016	23.68 ± .94	5.55 ± .12	9.73 ± .49	
		6	G-41	.018 ± .004	2.66 ± .02	.168 ± .004	97.14 ± .20	85.58 ± .86	10.52 ± .10	3.67 ± .04	0.218 ± .008	19.41 ± .80	3.66 ± .08	5.68 ± .28	
E5-SW-a5	15025	1	T-46	.0197 ± .0004	2.627 ± .014	.1858 ± .0029	97.167 ± .015	84.320 ± .041	11.191 ± .042	4.218 ± .009	0.271 ± .004	6.029 ± .038	3.738 ± .072	6.29 ± .31	
		3	T-47	.0169 ± .0001	1.990 ± .004	.3080 ± .0016	97.685 ± .005	73.316 ± .058	16.411 ± .052	9.003 ± .050	1.270 ± .012	6.075 ± .018	5.779 ± .094	12.87 ± .64	
		5	T-48	.0177 ± .0003	2.049 ± .001	.2903 ± .0015	97.643 ± .002	74.387 ± .050	15.990 ± .050	8.504 ± .033	1.120 ± .007	6.505 ± .013	5.536 ± .100	11.89 ± .60	
		6	T-49	.0194 ± .0003	2.550 ± .007	.2051 ± .0017	97.226 ± .007	82.734 ± .034	12.110 ± .033	4.797 ± .018	0.360 ± .003	6.052 ± .018	4.223 ± .085	7.15 ± .36	
F4-NE-f1	20036	1	T-50	.0192 ± .0002	2.589 ± .006	.1924 ± .0007	97.200 ± .007	84.247 ± .030	11.257 ± .028	4.214 ± .016	0.281 ± .004	6.877 ± .028	3.735 ± .047	6.40 ± .32	
		3	T-51	.0171 ± .0002	1.897 ± .007	.3214 ± .0016	97.764 ± .008	72.070 ± .048	17.018 ± .044	9.430 ± .036	1.483 ± .007	5.956 ± .038	5.393 ± .123	12.61 ± .63	
		5	T-52	.0171 ± .0002	1.879 ± .011	.3259 ± .0030	97.779 ± .013	71.763 ± .063	17.113 ± .067	9.581 ± .029	1.543 ± .014	6.758 ± .066	5.443 ± .142	13.35 ± .67	
		6	T-53	.0188 ± .0001	2.406 ± .004	.2334 ± .0009	97.342 ± .004	80.841 ± .044	13.014 ± .045	5.647 ± .018	0.498 ± .004	7.067 ± .035	4.285 ± .105	8.14 ± .41	
F4-SW-a5	17383	1	T-75	.0179 ± .0001	2.740 ± .003	.1595 ± .0004	97.083 ± .003	86.348 ± .042	10.116 ± .040	3.361 ± .021	0.1757 ± .0009	6.012 ± .010	3.481 ± .049	5.33 ± .27	
		3	T-76	.0159 ± .0003	2.204 ± .004	.2671 ± .0016	97.513 ± .005	76.561 ± .040	14.993 ± .031	7.600 ± .038	0.847 ± .005	7.826 ± .020	5.480 ± .089	10.32 ± .52	
		5	T-77	.0162 ± .0002	2.153 ± .004	.2695 ± .0009	97.561 ± .004	75.995 ± .020	15.308 ± .018	7.777 ± .017	0.920 ± .003	6.477 ± .020	5.327 ± .056	10.88 ± .54	
		6	T-78	.0175 ± .0001	2.606 ± .002	.1855 ± .0008	97.191 ± .002	84.088 ± .024	11.391 ± .024	4.242 ± .012	0.280 ± .002	6.217 ± .015	3.960 ± .048	6.26 ± .31	
PHASE II, PERTURBED NEUTRON SPECTRUM, CORNER SUBASSEMBLIES															
F5-NW-a1	8604	1	G-47	.017 ± .004	2.37 ± .02	.231 ± .004	97.38 ± .80	80.86 ± .60	12.30 ± .12	6.34 ± .06	0.507 ± .010	25.8 ± 1.0	5.19 ± .10	8.45 ± .42	7.85 ± .55
		3	G-48	.014 ± .002	1.39 ± .02	.416 ± .006	98.18 ± .80	63.59 ± .59	19.78 ± .16	13.33 ± .13	3.31 ± .06	28.7 ± 1.0	6.95 ± .14	20.72 ± 1.03	14.55 ± 1.02
		5	G-49	.015 ± .002	1.30 ± .02	.422 ± .006	98.26 ± .80	62.23 ± .59	20.60 ± .18	13.50 ± .13	3.67 ± .06	27.4 ± 1.0	6.46 ± .12	21.06 ± 1.05	14.27 ± 1.00
		6	G-50	.018 ± .004	1.89 ± .02	.310 ± .006	97.78 ± .80	72.25 ± .60	16.65 ± .16	9.68 ± .09	1.42 ± .02	19.4 ± .8	5.79 ± .12	12.85 ± .64	8.09 ± .57
F5-SE-f5	8717	1	T-123	.0189 ± .0002	2.254 ± .003	.2614 ± .0011	97.465 ± .003	78.266 ± .017	13.573 ± .015	7.419 ± .013	0.7413 ± .0031	5.298 ± .011	5.464 ± .053	9.41 ± .47	10.23 ± .72
		2	T-124	.0157 ± .0004	1.406 ± .002	.4151 ± .0089	98.164 ± .009	63.687 ± .052	20.048 ± .048	13.082 ± .049	3.182 ± .021	5.358 ± .013	6.654 ± .095	17.28 ± .86	13.44 ± .94
		3	T-125	.0149 ± .0002	1.262 ± .001	.4323 ± .0009	98.291 ± .002	61.217 ± .012	21.074 ± .011	13.844 ± .007	3.864 ± .012	6.174 ± .051	6.742 ± .109	22.94 ± 1.15	15.45 ± 1.08
		4	T-126	.0135 ± .0003	1.273 ± .004	.4213 ± .0015	98.292 ± .004	61.69 ± .11	20.86 ± .10	13.70 ± .12	3.755 ± .041	5.933 ± .020	6.965 ± .142	23.38 ± 1.17	14.86 ± 1.04
		5	T-127	.0135 ± .0001	1.352 ± .010	.4055 ± .0014	98.229 ± .010	63.135 ± .032	20.439 ± .030	13.115 ± .031	3.310 ± .012	5.816 ± .021	6.733 ± .086	20.73 ± 1.04	14.68 ± 1.03
		6	T-128	.0160 ± .0001	1.979 ± .013	.2950 ± .0009	97.710 ± .013	74.033 ± .036	16.055 ± .039	8.732 ± .014	1.180 ± .004	5.954 ± .028	5.692 ± .077	12.20 ± .61	10.05 ± .70

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number ^a	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc.	Pu-239/U-238	Cs-137 ^c	Sr-90 ^c
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	(a/ml x 10 ⁻¹⁶)	(a/a x 10 ³)	(dpm/gmU x 10 ⁻¹⁰)	(dpm/gmU x 10 ⁻¹⁰)
PHASE I, PERTURBED NEUTRON SPECTRUM, CORNER SUBASSEMBLIES															
G4-NW-a1	1XY39	1	T	.02 ± .01	2.56 ± .01	.18 ± .01	97.24 ± .02	84.329 ± .044	11.285 ± .030	4.114 ± .022	0.2721 ± .0021	6.00 ± .05	3.77 ± .05 ^m	6.19 ± .37	5.13 ± .41
		3	T	.02 ± .01	1.87 ± .01	.31 ± .01	97.80 ± .01	71.913 ± .104	17.114 ± .060	9.486 ± .061	1.487 ± .015	6.40 ± .05	5.49 ± .10 ^m	12.91 ± .77	9.86 ± .79
		5	T	.02 ± .01	1.83 ± .01	.32 ± .01	97.84 ± .01	71.882 ± .048	17.099 ± .024	9.502 ± .043	1.517 ± .002	6.02 ± .05	5.49 ± .08 ^m	12.77 ± .77	12.31 ± .98
		6	T	.02 ± .01	2.42 ± .01	.23 ± .01	97.34 ± .01	80.952 ± .051	12.971 ± .035	5.579 ± .019	0.4977 ± .0026	5.79 ± .02	4.35 ± .06 ^m	7.78 ± .47	6.10 ± .49
G4-SE-f5	12760	1	G-34	.021 ± .004	2.69 ± .02	.189 ± .004	97.10 ± .20	85.24 ± .86	10.53 ± .10	3.99 ± .04	0.243 ± .008	22.00 ± .88	3.78 ± .08	5.95 ± .30	
		3	G-35	.018 ± .004	1.92 ± .02	.323 ± .006	97.73 ± .20	72.67 ± .71	16.50 ± .16	9.41 ± .09	1.41 ± .02	23.07 ± .92	5.60 ± .10	13.28 ± .66	
		5	G-36	.016 ± .002	1.86 ± .02	.321 ± .006	97.79 ± .20	72.27 ± .71	16.63 ± .16	9.58 ± .09	1.51 ± .02	20.64 ± .80	5.68 ± .10	13.37 ± .67	
		6	G-37	.020 ± .004	2.38 ± .02	.233 ± .004	97.36 ± .20	81.38 ± .80	12.58 ± .12	5.56 ± .04	0.482 ± .008	16.62 ± .66	4.27 ± .08	8.34 ± .42	
H3-SE-f5	220	Comp. ^{Ln}	G	.0165 ± .0004	2.29 ± .02	.231 ± .004	97.46 ± .44	78.886 ± .538	13.851 ± .100	6.549 ± .021	0.7130 ± .0080	4.94 ± .20	4.46 ± .08		
H3-SE-f5	220	1	G	.0182 ± .0002	2.77 ± .02	.144 ± .002	97.07 ± .44	87.93 ± .54	9.20 ± .06	2.73 ± .04	0.137 ± .006	6.99 ± .28	2.95 ± .06	4.67 ± .37	4.26 ± .38
		2	G	.0175 ± .0004	2.25 ± .02	.229 ± .002	97.50 ± .44	79.06 ± .44	13.88 ± .08	6.40 ± .06	0.651 ± .010	4.48 ± .18	4.22 ± .08	8.96 ± .72	7.47 ± .77
		3	G	.0162 ± .0004	2.137 ± .02	.259 ± .004	97.59 ± .32	76.97 ± .38	14.82 ± .08	7.37 ± .11	0.846 ± .032	6.59 ± .26	4.73 ± .18	10.40 ± .83	7.91 ± .72
		4	G	.0156 ± .0008	2.09 ± .02	.262 ± .002	97.64 ± .48	76.89 ± .30	14.76 ± .06	7.47 ± .04	0.879 ± .008	5.51 ± .22	4.76 ± .18	10.71 ± .86	8.22 ± .74
		5 ^g	G	.0179 ± .0002	2.21 ± .02	.257 ± .002	97.52 ± .68	77.94 ± .52	14.37 ± .06	6.95 ± .02	0.746 ± .006	5.15 ± .20	4.74 ± .04	9.72 ± .78	8.64 ± .78
		5 ^g	T-7	.0172 ± .0003	2.198 ± .008	.2555 ± .0012	97.530 ± .009	77.934 ± .053	14.422 ± .054	6.904 ± .026	0.740 ± .009	4.514 ± .023	4.81 ± .30		
		5 ^g	N-3	.016 ± .003	2.22 ± .03	.26 ± .02	97.50 ± .06	78.23 ± .06	14.14 ± .06	6.86 ± .04	0.76 ± .02	4.54	4.80		
6	G	.0170 ± .0006	2.60 ± .02	.162 ± .002	97.22 ± .38	86.08 ± .56	10.23 ± .06	3.49 ± .03	0.026 ± .004	5.44 ± .22	3.46 ± .18	5.52 ± .44	5.77 ± .52		
H5-NW-a1	9709	1	T-54	.0196 ± .0003	2.658 ± .007	.1745 ± .0013	97.148 ± .008	85.696 ± .036	10.403 ± .033	3.681 ± .019	0.219 ± .002	5.741 ± .020	3.561 ± .065	5.67 ± .28	
		3	T-55	.0171 ± .0001	1.941 ± .002	.3075 ± .0006	97.735 ± .003	73.162 ± .023	16.462 ± .025	9.058 ± .007	1.319 ± .003	5.566 ± .010	5.433 ± .131	12.65 ± .63	
		5	T-56	.0169 ± .0001	1.884 ± .004	.3143 ± .0020	97.785 ± .005	72.724 ± .025	16.604 ± .019	9.266 ± .023	1.406 ± .010	6.024 ± .020	5.356 ± .108	12.81 ± .64	
		6	T-57	.0184 ± .0001	2.357 ± .006	.2258 ± .0010	97.399 ± .007	81.299 ± .051	12.723 ± .048	5.505 ± .030	0.473 ± .005	6.621 ± .023	4.247 ± .095	8.12 ± .41	
J4-NW-a1	11183	1	T	.02 ± .01	2.77 ± .01	.14 ± .01	97.08 ± .01	87.838 ± .047	9.309 ± .045	2.715 ± .020	0.1385 ± .0030	6.60 ± .08	3.02 ± .05 ^m	4.38 ± .26	3.03 ± .24
		2	T	.02 ± .01	2.23 ± .02	.23 ± .01	97.52 ± .02	78.980 ± .055	13.984 ± .050	6.386 ± .035	0.6501 ± .0101	5.97 ± .05	4.50 ± .07 ^m	8.91 ± .53	6.58 ± .53
		3	T	.02 ± .01	2.09 ± .01	.25 ± .01	97.64 ± .01	76.898 ± .067	14.939 ± .060	7.314 ± .051	0.8495 ± .0120	6.07 ± .02	4.76 ± .06 ^m	10.12 ± .61	7.88 ± .63
		4	T	.02 ± .01	2.09 ± .01	.26 ± .01	97.64 ± .01	76.398 ± .049	15.129 ± .049	7.587 ± .027	0.8867 ± .0043	6.43 ± .02	4.82 ± .06 ^m	10.59 ± .64	8.23 ± .66
		5												9.48 ± .57	7.32 ± .58
		6												5.17 ± .31	4.69 ± .37
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY															
F4-SW-b4	17457	1	T-79	.0187 ± .0002	2.800 ± .019	.1472 ± .0011	97.034 ± .020	87.622 ± .071	9.230 ± .064	2.995 ± .044	.153 ± .006	7.196 ± .053	3.336 ± .048	4.77 ± .24	
		3	T-80	.0163 ± .0001	2.321 ± .007	.2514 ± .0014	97.412 ± .008	78.774 ± .045	13.654 ± .047	6.892 ± .017	.680 ± .005	7.004 ± .033	5.304 ± .089	9.57 ± .48	
		5	T-81	.0172 ± .0002	2.289 ± .004	.2578 ± .0009	97.436 ± .004	78.261 ± .031	13.955 ± .032	7.066 ± .014	.718 ± .005	5.845 ± .010	5.316 ± .075	9.74 ± .49	
		6	T-82	.0176 ± .0002	2.674 ± .004	.1696 ± .0006	97.139 ± .005	85.296 ± .074	10.522 ± .070	3.953 ± .036	.228 ± .005	4.618 ± .008	3.828 ± .047	5.86 ± .29	
F4-SW-d2	17460	1	T-83	.0196 ± .0001	2.864 ± .001	.1560 ± .0006	96.961 ± .001	88.281 ± .082	8.610 ± .081	2.980 ± .028	.129 ± .003	6.255 ± .005	3.472 ± .043	4.57 ± .23	
		3	T-84	.0162 ± .0001	2.342 ± .002	.2458 ± .0004	97.396 ± .003	80.039 ± .030	12.546 ± .032	6.801 ± .008	.613 ± .002	5.450 ± .010	5.677 ± .080	9.20 ± .46	
		5	T-85	.0180 ± .0005	2.334 ± .003	.2558 ± .0007	97.393 ± .003	79.750 ± .014	12.758 ± .012	6.853 ± .011	.639 ± .003	5.162 ± .008	5.612 ± .090	9.42 ± .47	
		6	T-86	.0180 ± .0002	2.693 ± .004	.1681 ± .0005	97.121 ± .004	86.187 ± .028	9.782 ± .027	3.820 ± .012	.211 ± .002	6.037 ± .010	3.936 ± .048	5.62 ± .28	

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc.	Pu-239/U-238	Cs-137 ^c	Sr-90 ^c
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	(a/ml x 10 ⁻¹⁶)	(a/a x 10 ⁻³)	(dpm/gmU x 10 ⁻¹⁰)	(dpm/gmU x 10 ⁻¹⁰)
PHASE II, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY															
F5-NW-b2	8172	1	G-56	.018 ± .004	2.46 ± .02	.224 ± .004	97.29 ± .80	81.88 ± .80	11.79 ± .12	5.86 ± .06	0.475 ± .010	27.6 ± 1.0	5.02 ± .10	8.06 ± .40	6.44 ± .45
		3	G-57	.014 ± .002	1.52 ± .02	.380 ± .008	98.08 ± .80	67.13 ± .60	18.36 ± .18	12.04 ± .11	2.46 ± .04	24.8 ± 1.0	7.19 ± .14	18.50 ± .92	13.06 ± .91
		5	G-58	.017 ± .004	1.53 ± .02	.414 ± .008	98.04 ± .80	69.95 ± .60	14.27 ± .14	13.09 ± .13	2.77 ± .06	21.7 ± 0.8	7.04 ± .14	19.03 ± .95	13.09 ± .92
		6	G-59	.018 ± .004	2.07 ± .02	.298 ± .006	97.61 ± .80	75.45 ± .60	15.01 ± .14	8.49 ± .08	1.05 ± .02	24.9 ± 1.0	5.67 ± .10	11.26 ± .56	9.69 ± .68
F5-NW-d4	8163	1	G-52	.020 ± .004	2.42 ± .02	.215 ± .004	97.34 ± .80	81.75 ± .60	11.57 ± .12	5.88 ± .06	0.802 ± .016	31.2 ± 1.2	5.12 ± .10	7.82 ± .39	7.02 ± .49
		3	G-53	.014 ± .002	1.66 ± .02	.363 ± .006	97.96 ± .80	69.87 ± .60	16.64 ± .16	11.50 ± .11	1.99 ± .04	24.5 ± 1.0	7.64 ± .14	16.73 ± .84	10.93 ± .76
		5	G-54	.020 ± .004	1.66 ± .02	.388 ± .008	97.92 ± .80	69.37 ± .60	16.93 ± .18	11.65 ± .11	2.05 ± .04	25.5 ± 1.0	7.44 ± .14	16.93 ± .85	11.15 ± .78
		6	G-55	.013 ± .002	2.17 ± .02	.291 ± .006	97.52 ± .80	77.42 ± .60	13.82 ± .14	7.89 ± .15	0.867 ± .018	28.0 ± 1.2	5.74 ± .12	10.18 ± .51	9.01 ± .63
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY															
H3-SE-b1	204	1	G	.0201 ± .0004	3.03 ± .02	.109 ± .004	96.84 ± .88	90.68 ± 1.56	7.23 ± .12	2.03 ± .04	0.068 ± .008	6.33 ± .04	2.76 ± .02 ^m	3.30 ± .20	2.85 ± .23
		3	G	.0174 ± .0006	2.50 ± .02	.195 ± .002	97.29 ± 1.10	83.05 ± 1.66	11.25 ± .08	5.30 ± .04	0.395 ± .014	5.98 ± .24	4.82 ± .04 ^m	7.23 ± .43	5.84 ± .47
		5 ^g	T	.02 ± .01	2.51 ± .01	.21 ± .01	97.26 ± .01	83.373 ± .054	11.087 ± .034	5.167 ± .030	0.3735 ± .0034	6.15 ± .02	4.78 ± .06 ^m	6.97 ± .42	5.40 ± .43
		5	G	.0174 ± .0004	2.56 ± .02	.207 ± .004	97.22 ± 1.08	83.41 ± 1.48	11.05 ± .06	5.17 ± .02	0.377 ± .006	5.90 ± .24	4.73 ± .04 ^m	6.97 ± .42	5.40 ± .43
		6	G	.0176 ± .0004	2.84 ± .02	.132 ± .004	97.01 ± .60	89.52 ± 1.80	7.90 ± .04	2.48 ± .02	0.101 ± .004	5.94 ± .22	3.08 ± .04 ^m	4.05 ± .24	3.63 ± .29
H3-SE-c2	116	1	G-26	.019 ± .004	3.01 ± .02	.112 ± .002	96.86 ± .20	91.45 ± .40	6.79 ± .06	1.71 ± .02	0.054 ± .010	19.9 ± .8	2.48 ± .04	3.59 ± .18	
		3	G-27	.016 ± .004	2.47 ± .02	.219 ± .004	97.29 ± .20	82.50 ± .40	11.62 ± .08	5.46 ± .04	0.416 ± .010	24.9 ± 1.0	4.67 ± .10	7.59 ± .38	
		5 ^o	G-28	.017 ± .004	2.79 ± .02	.144 ± .002	97.05 ± .20	88.58 ± .40	8.46 ± .08	2.82 ± .04	0.133 ± .002	15.5 ± .6	3.34 ± .06	7.69 ± .38	
		6	G-29	.018 ± .004	2.87 ± .02	.127 ± .002	96.98 ± .20	90.02 ± .40	7.54 ± .08	2.35 ± .04	0.092 ± .018	23.5 ± .8	2.98 ± .06	4.05 ± .20	
H3-SE-d3	103	1	G	.0178 ± .0004	2.94 ± .02	.123 ± .0018	96.93 ± .80	90.17 ± 1.34	7.63 ± .16	2.12 ± .04	0.079 ± .010	6.22 ± .24	2.87 ± .04 ^m	3.29 ± .20	2.71 ± .22
		3	G	.0166 ± .0010	2.43 ± .02	.220 ± .004	97.33 ± .88	81.74 ± 1.65	12.07 ± .20	5.71 ± .12	0.477 ± .018	5.69 ± .20	4.85 ± .04 ^m	7.79 ± .47	5.67 ± .45
		5	G	.0131 ± .0006	2.51 ± .06	.214 ± .004	97.26 ± 1.14	82.19 ± .84	11.92 ± .12	5.45 ± .04	0.433 ± .010	5.48 ± .20	4.75 ± .04 ^m	7.72 ± .46	5.48 ± .44
		6	G	.0178 ± .0006	2.86 ± .02	.1400 ± .0012	96.98 ± .62	88.89 ± .64	8.39 ± .04	2.61 ± .02	0.111 ± .002	5.82 ± .18	3.18 ± .04 ^m	4.54 ± .27	3.34 ± .27
H3-SE-e4	111	1	G-30	.019 ± .004	2.90 ± .02	.123 ± .002	96.96 ± .20	90.06 ± .90	7.74 ± .08	2.11 ± .02	0.0824 ± .0032	34.16 ± 1.36	2.58 ± .06	4.10 ± .20	
		3	G-31	.017 ± .004	2.32 ± .02	.236 ± .004	97.42 ± .20	80.09 ± .80	13.06 ± .12	6.26 ± .06	0.590 ± .024	20.41 ± .82	4.55 ± .10	9.09 ± .45	
		5	G-32	.017 ± .002	2.37 ± .02	.231 ± .004	97.38 ± .20	80.72 ± .82	12.76 ± .12	5.98 ± .06	0.534 ± .022	30.65 ± 1.24	4.48 ± .10	9.63 ± .58	
		6	G-33	.018 ± .004	2.76 ± .02	.156 ± .004	97.06 ± .20	87.33 ± .88	9.36 ± .10	3.15 ± .02	0.161 ± .006	30.01 ± 1.20	3.17 ± .06	5.52 ± .33	
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY ADJACENT REFLECTOR															
J3-NE-b5	22871	1	T-92	.0208 ± .0002	3.257 ± .008	.0618 ± .0001	96.661 ± .009	96.419 ± .010	3.240 ± .009	0.331 ± .003	0.0098 ± .0043	6.389 ± .023	1.030 ± .018	1.20 ± .06	
		3	T-93	.0200 ± .0001	3.061 ± .005	.1065 ± .0007	96.812 ± .005	92.715 ± .011	5.964 ± .009	1.285 ± .008	0.0356 ± .0012	5.468 ± .023	2.119 ± .030	2.65 ± .16	
		5	T-94	.0199 ± .0001	3.063 ± .005	.1027 ± .0005	96.814 ± .005	92.968 ± .005	5.813 ± .004	1.189 ± .004	0.0299 ± .0004	5.802 ± .013	2.024 ± .032	2.63 ± .13	
		6	T-95	.0217 ± .0022	3.207 ± .030	.0693 ± .0020	96.702 ± .031	95.848 ± .010	3.705 ± .009	0.438 ± .004	0.0094 ± .0004	5.754 ± .063	1.173 ± .022	1.41 ± .07	
J3-NE-d3	1425	1	T-109	.0205 ± .0002	3.268 ± .005	.0495 ± .0003	96.663 ± .005	96.933 ± .039	2.775 ± .037	0.261 ± .016	0.031 ± .008	7.183 ± .015	0.758 ± .021	0.95 ± .05	
		3	T-110	.0206 ± .0001	3.094 ± .003	.0873 ± .0004	96.798 ± .003	93.835 ± .019	5.274 ± .018	0.871 ± .007	0.0202 ± .0006	6.022 ± .015	1.62 ± .030	2.20 ± .11	
		5	T-111	.0207 ± .0005	3.107 ± .010	.0856 ± .0014	96.787 ± .010	93.921 ± .042	5.216 ± .042	0.839 ± .007	0.0230 ± .0016	6.333 ± .028	1.52 ± .021	2.13 ± .10	
		6	T-112	.0215 ± .0003	3.258 ± .008	.0555 ± .0007	96.665 ± .008	96.475 ± .009	3.231 ± .008	0.285 ± .003	0.0093 ± .0088	5.657 ± .023	0.908 ± .017	1.19 ± .06	

TABLE A1.1 (continued)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number ^a	ATOM PERCENT URANIUM ^b				ATOM PERCENT PLUTONIUM ^c				U-238 Conc. (a/ml x 10 ⁻¹⁶)	Pu-239/U-238 (a/a x 10 ³)	Cs-137 ^c (dpm/gmU x 10 ⁻¹⁰)	Sr-90 ^c (dpm/gmU x 10 ⁻¹⁰)
				U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242				
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY ADJACENT REFLECTOR															
J3-NE-e2	31	3	T-130	.0200 ± .0004	3.062 ± .006	.0895 ± .0008	96.827 ± .006	93.756 ± .027	5.414 ± .026	0.809 ± .008	0.0205 ± .0027	7.128 ± .015	1.403 ± .018	2.15 ± .10	
		5	T-131	.0202 ± .0002	3.136 ± .018	.0796 ± .0006	96.764 ± .018	94.552 ± .035	4.825 ± .033	0.607 ± .017	0.0163 ± .0032	7.355 ± .048	1.238 ± .016	1.83 ± .09	
J3-NE-f1	23018	1	T-96	.0204 ± .0002	3.287 ± .005	.0551 ± .0004	96.637 ± .005	96.935 ± .016	2.823 ± .015	0.226 ± .003	0.0160 ± .0005	5.668 ± .018	0.602 ± .012	1.02 ± .05	
		3	T-97	.0201 ± .0001	3.089 ± .008	.0847 ± .0004	96.807 ± .008	93.918 ± .034	5.382 ± .034	0.681 ± .006	0.0187 ± .0002	6.029 ± .020	1.263 ± .016	2.23 ± .13	
		5	T-98	.0202 ± .0003	3.078 ± .005	.0866 ± .0004	96.815 ± .005	93.908 ± .016	5.422 ± .016	0.651 ± .003	0.0181 ± .0012	5.200 ± .010	1.241 ± .015	2.20 ± .13	
		6	T-99	.0206 ± .0001	3.242 ± .004	.0607 ± .0002	96.677 ± .004	95.789 ± .013	3.689 ± .012	0.485 ± .007	0.0366 ± .0016	6.070 ± .010	0.718 ± .010	1.68 ± .08	

- Notes: a) The letters T, G, and N indicate that the mass spectrometric analyses were performed by the TRACERLAB (TL), General Electric - Vallecitos (GE), and the AEC New Brunswick (NB) analytical laboratories, respectively. The radiochemical analyses for Cs-137 and Sr-90 were performed by the Technical Service Laboratories, Atomic Power Division, Westinghouse.
- b) The atom percent of U-234, U-235, U-236, and U-238 are relative to the total uranium atoms. The atom percent of Pu-239, Pu-240, Pu-241, and Pu-242 are relative to the total plutonium atoms, not including Pu-238.
- c) All Phase I data are referenced to the Yankee Core I end-of-life on May 18, 1962; all Phase II data are referenced to the Yankee Core II end-of-life on September 2, 1963. To account for radioactive decay, half-lives of 13.2, 29.4, and 28.0 years were used for Pu-241, Cs-137, and Sr-90, respectively.
- d) The letters A and B denote the pellet samples immediately above and below the nominal zone 3 pellet, respectively.
- e) The uncertainty listed for all TL and GE mass spectrometric data is ± 2-standard deviation precision. The limits of uncertainty for the NB uranium composition are believed to be within that value listed, based on NBS U-030. There is a recognized bias of approximately + 1%, relative, in the NB U-235 a/o (see Ref. 8, p. 3509). The NB plutonium composition data is ± 2-standard deviation precision only. No estimate of uncertainty is given for the NB U-238 a/ml and Pu-239/U-238 data.
- f) The uncertainty listed for all radiochemical data is ± 2-standard deviation accuracy.
- g) Interlaboratory crosscheck sample.
- h) The numerals 1 and 2 in the NB sample code numbers indicate that the results are for the first and second runs of a mass spectrometric analysis performed in duplicate.
- i) Original sample dilution.
- j) New dilution of original sample.
- k) Seven solutions of the monitor sample 0994-3 were submitted to TRACERLAB over the time interval January 1964 to November 1964; four solutions of the monitor were submitted to GE-Vallecitos over the time interval April 1964 to October 1964.
- l) The analytical laboratories require monitor samples of different concentration.
- m) Dilute acid sample. Samples submitted for isotopic analysis prior to June 7, 1963 had nitric acid concentrations of about 0.01 M, and are referred to as dilute acid samples. All other samples, submitted after this date, were in approximately 2 M HNO₃, and are referred to as concentrated acid samples. The sample history of shipping, spiking, and data reporting dates can be found in the Yankee Core Evaluation quarterly progress reports.
- The Pu-239/U-238 atom ratios shown for the dilute acid samples in this table have been adjusted with the use of plutonium data obtained for concentrated acid samples. The U and Pu atom percent data are not affected by the acidity of the sample solution.
- n) A fuel rod section of 1.9 inches (approximately 3 pellets), was cut from a position immediately above the nominal 1-pellet sample location in each of the six axial sample zones. The 6 sections were then dissolved and diluted to make one composite sample solution.
- o) All mass spectrometric data for this sample are consistent. However, the relatively low burnup inferred from the data indicates a high probability that the sample solution was incorrectly identified.

Generally, samples were taken from Axial Sample Zones 1, 3, 5, and 6 of the fuel rods shown in Figure 3.2.2. In several spent fuel rods, samples were taken from all six zones. For the reasons given in Section 3.0, some fuel rods were sampled at just 1 or 2 positions, while a few rods were not sampled at all. The axial sample zones are described in Figure 3.3.2 and Table 3.3.1.

Most of the sample solutions sent to the analytical laboratories for isotopic analyses were coded. The code number for each sample contains the letter T, G, or N to indicate that the isotopic analysis was performed by TRACERLAB, General Electric-Vallecitos, or the AEC New Brunswick Laboratories, respectively. Approximately 50 samples submitted for isotopic analysis early in the EYC Program were not coded; for these samples, only the letter T, G, or N is shown to indicate which laboratory performed the analysis. All the radiochemical analyses for Cs-137 and Sr-90 were performed by the Technical Services Laboratories of WAPD.

The uranium composition is given in terms of the percent of U-234, U-235, U-236 and U-238 atoms, each relative to the total atoms of these four uranium isotopes present in the spent fuel sample. Similarly, the plutonium composition is given in terms of the percent of Pu-239, Pu-240, Pu-241, and Pu-242 atoms, each relative to the total atoms of the four plutonium isotopes present in the spent fuel sample. The uranium composition is given exactly as reported by the analytical laboratories. However, the plutonium composition given in Table A1.1 is not exactly as reported by the analytical laboratories; the Pu-241 concentration reported by the laboratories at the time of analysis has been corrected for radioactive decay between the time of Core I shutdown and the time of analysis for the Phase I samples, and for radioactive decay between the time of Core II shutdown and the time of analysis (including the time between Core I shutdown and Core II startup) for the Phase II samples. Using the decay corrected value of Pu-241 for each sample, the plutonium isotopic atom percents were recomputed, and it is these values, referenced to the core end-of-life (EOL) which are listed in Table A1.1.

The Pu-238 present in the spent fuel samples has not been measured under this program; consequently Pu-238 is not included among the plutonium isotopes in Table A1.1.

Both the GE-Vallecitos and the TRACERLAB laboratories report the fuel composition data in terms of U and Pu isotopic atom percents. The GE-Vallecitos laboratory reports the U and Pu concentration data in terms of U-238 atoms/ml and Pu-239/U-238 atom ratio, as shown in Table A1.1. TRACERLAB reports the U and Pu concentration in terms of ppm of each of the 4 uranium and 4 plutonium isotopes present in the sample solution. For simplicity of presentation, the ppm data have been converted to U-238 atoms/ml and the Pu-239/U-238 atom ratio with the equations described in Appendix A2, and it is these values which are listed for the TRACERLAB concentration data in Table A1.1.

Forty-six samples were submitted for analysis in dilute acid. While the acidity of the sample solution does not affect the U and Pu composition data, the acidity does affect the Pu concentration data as demonstrated in Figure A1.1. For this reason all the Pu concentration data reported for the dilute acid samples have been adjusted, or corrected, with the method described on pages 350.5 through 350.16 of Reference 21. Briefly, the method involves the use of spectrum-dependent correlations which relate the Pu-239/U-238 atom ratio to the Pu-239 atom percent as determined from the isotopic results obtained for the many samples submitted for analysis in concentrated acid solution. Using these correlations, and the Pu-239 atom percent measured for the dilute acid samples, the adjusted Pu-239/U-238 atom ratio is then determined for the dilute acid samples.

The dilute acid data plotted on Figure A1.1 are shown in Table A1.2 along with the corrected, or adjusted data. The Pu-239/U-238 atom ratios reported for the 29 dilute acid samples submitted to TRACERLAB are approximately 15% low, relative to the corrected values. The same discrepancy, but to a smaller degree, can be observed in 6 of the 17 values of Pu-239/U-238 atom ratio reported by GE-Vallecitos. Most of the discrepancies noted in the plutonium concentrations reported for the dilute acid samples have been attributed to plutonium plateout on the sides of the sample containers. The TRACERLAB data are consistent with this explanation. However, the GE-Vallecitos data indicate that the discrepancies may be due to some other cause (e.g., the methods of chemical separation used in the analysis may be dependent upon the acidity of the sample solution), since the reported and adjusted values of Pu-239/U-238 for 11 of 17 dilute acid samples are essentially the same.

Pu-239/U-238 Atom Ratio Versus Burnup for Yankee Spent Fuel Samples Submitted for Analysis in Concentrated and Dilute Acid Solutions.

Figure A1.1

TABLE A1.2
Pu-239/U-238 Atom Ratio in Dilute Acid Samples

					Pu-239/U-238 Atom Ratio					Ratio of
Fuel Rod Core Location	Fuel Rod Number	Axial Sample Zone	Neutron Spectrum	Analytical Laboratory	Reported ^{/a}	Adjusted ^{/b}	Adjusted Pu-239/U-238 Reported	Inferred Burnup (MWD/MTU)		Burnup Values (Adjusted/ Reported)
					Dilute Acid Sample	With Use of Concentrated Acid Data		Using Reported	Using Adjusted	
G4-C-a6	0985	1	Asymptotic	TL	.00339	.00356	1.06	6,780	6,830	1.01
		3			.00536	.00589	1.10	13,360	13,680	1.02
		5			.00479	.00593	1.24	13,270	13,970	1.05
		6			.00403	.00397	0.98	8,520	8,500	1.00
G4-C-f1	0979	1	Asymptotic	TL	.00323	.00350	1.08	6,390	6,470	1.01
		3			.00509	.00582	1.14	13,340	13,780	1.03
		5			.00457	.00585	1.28	12,860	13,630	1.06
		6			.00374	.00419	1.12	8,510	8,680	1.02
H3-C-a6	8510	Composite ^{/c}		TL	.00242	.00511	2.11	8,360	9,450	1.13
H3-C-a6	8510	1	Asymptotic	TL	.00267	.00302	1.13	5,350	5,440	1.02
		2			.00408	.00479	1.17	9,640	9,960	1.03
		3			.00445	.00517	1.16	10,760	11,120	1.03
		4			.00458	.00530	1.16	11,330	11,690	1.03
		5			.00455	.00514	1.13	11,000	11,300	1.03
		6			.00296	.00351	1.18	6,830	7,000	1.02
H3-C-f1	8528	5 ^{/d}	Asymptotic	TL	.00343	.00456	1.33	8,200	8,680	1.06
J4-C-a1	305	1	Asymptotic	TL	.00232	.00263	1.13	4,660	4,730	1.02
		3			.00412	.00469	1.14	9,400	9,650	1.03
		5			.00410	.00467	1.14	9,690	9,940	1.02
		6			.00273	.00306	1.12	6,110	6,200	1.01
H3-SE-b1	204	5 ^{/d}	Intermediate	TL	.00417	.00478	1.15	9,680	9,950	1.03
G4-NW-a1	1XY39 ^{/f}	1	Perturbed	TL	.00324	.00377	1.16	8,770	8,990	1.02
		3			.00475	.00549	1.16	17,620	18,310	1.04
		5			.00452	.00549	1.21	17,730	18,640	1.05
		6			.00377	.00435	1.15	10,340	10,640	1.03
J4-NW-a1	11183	1	Perturbed	TL	.00257	.00302	1.18	6,490	6,630	1.02
		2			.00362	.00450	1.24	12,480	13,010	1.04
		3			.00395	.00476	1.20	14,250	14,810	1.04
		4			.00366	.00482	1.32	14,030	14,850	1.06
					Average ^{/e}		1.16			1.03
G4-C-a6	0985	5 ^{/d}	Asymptotic	GE	.00557	.00597	1.07	14,080	14,330	1.02
G4-C-f1	0979	5 ^{/d}	Asymptotic	GE	.00550	.00582	1.06	13,440	13,630	1.01
H3-C-a1	8537	5	Asymptotic	GE	.00468	.00490	1.05	10,780	10,880	1.01
H3-C-a6	8510	4 ^{/d}	Asymptotic	GE	.00515	.00525	1.02	10,660	10,710	1.00
H3-C-f1	8528	1	Asymptotic	GE	.00258	.00255	0.99	4,320	4,320	1.00
		3			.00459	.00462	1.01	9,340	9,350	1.00
		5			.00457	.00454	0.99	9,070	9,050	1.00
		6			.00300	.00296	0.99	5,460	5,450	1.00
J4-C-a1	305	5 ^{/d}	Asymptotic	GE	.00419	.00458	1.09	9,680	9,850	1.02
H3-SE-b1	204	1	Intermediate	GE	.00274	.00276	1.01	4,050	4,060	1.00
		3			.00472	.00482	1.02	10,250	10,290	1.00
		5			.00472	.00473	1.00	9,430	9,440	1.00
		6			.00302	.00308	1.02	5,970	5,980	1.00
H3-SE-d3	103	1	Intermediate	GE	.00271	.00287	1.06	4,860	4,900	1.01
		3			.00453	.00485	1.07	10,780	10,940	1.01
		5			.00472	.00475	1.01	10,020	10,040	1.00
		6			.00322	.00318	0.99	5,750	5,740	1.00
					Average		1.03			1.00

NOTES:

- In most cases, the GE plutonium concentration data is reported as Pu-239/U-238 a/a, and the TL data is reported as isotopic ppm.
- Determined from spectrum dependent correlations of Pu-239/U-238 atom ratio versus Pu-239 atom percent.
- A fuel rod section of 1.9 inches (approximately 3 pellets) was cut from a position immediately above the nominal 1- pellet sample location in each of the six axial sample zones. The six sections were then dissolved and diluted to make one composite sample solution.
- Cross check sample.
- Results for composite sample (H3-C-a6) are not included.
- The letters X and Y indicate intergers not discernible during hot cell visual examination.

The Cs-137 and Sr-90 data are given in units of disintegrations per minute per gram of uranium. All the radiochemical data are referenced to the end of Core I life for the Phase I samples, and to the end of Core II life for the Phase II samples. No corrections or adjustments have been made to the radiochemical data obtained for those samples analyzed in dilute acid.

The uncertainties listed for the quantities in Table A1.1 are $\pm$ 2-standard deviation precision for the isotopic data, and $\pm$ 2-standard deviation accuracy for the radiochemical data.

A description of the analytical treatment afforded the data in Table A1.1 is given in Appendix A2.

APPENDIX A2

Treatment of the EYC Data

A material balance, performed with pre-irradiated and post-irradiated fuel composition data, can be used to infer the various fuel characteristics and nuclear parameters of interest (e.g., fuel burnup, specific production of U and Pu isotopes, and U-235 capture-to-fission ratio). However, the required computations are somewhat long and laborious, and involve several correction factors which are dependent upon the parameters to be calculated. For these reasons, and due to the great number of spent fuel analyses (≈ 270) performed in the EYC Program, the REBUF-2 computer code was written to facilitate the reduction of the mass spectrometric data. Provisions have been included in the code to process Cs-137, Sr-90, and X-ray fluorescence Pu/U data. Typical machine time for stacked cases on the IBM-7094 is 2 to 3 seconds per sample.

The data reduction methods used to infer fuel burnup are similar to those outlined in Reference 13.

A complete set of measured data obtained for the physics portion of the EYC Program is given in Appendix A1 of this report. Typical REBUF-2 printouts are shown on Figures A2.1 through A2.4 and a description of the code is given in the following sections. The burnup and fuel characteristics inferred from the measured data, with the use of the REBUF code, are given in Appendix A3.

1) Input to the REBUF-2 Code

For each spent fuel sample to be analyzed, the code requires the following input. (The superscript used to indicate a particular isotope consists of the last designator of the atomic number and the mass number of the isotope. Thus, the superscripts 24, 25, 26, and 28 refer to uranium-234, 235, 236 and 238 respectively; the superscripts 49, 40, 41 and 42 refer to plutonium 239, 240, 241, and 242 respectively.)

- A. Assumed beginning-of-life isotopic composition expressed as atom percent.

$$\begin{aligned}(a/o)_{24}^{24} &= 0.02 \\(a/o)_{25}^{25} &= 3.44 \\(a/o)_{26}^{26} &= 0.02 \\(a/o)_{28}^{28} &= 96.52\end{aligned}$$

K4-C-A1 2907 ZONE 6 (TLAB 5/5/64 T-35, CS RE-ANAL ON 4/8/65) 4/28/65

REBUF - 2A CODE

ISOTOPE	I S O T O P I C		C O N T E N T		F I S S I O N S	
	ATOM PERCENT A/O	ERROR	PARTS PER MILLION PPM	NO. DENSITY N(EOL)/N28(BOL)	NET PRODUCTION KG/MTU(BOL)	F/N28(BOL) F/FTOT
U-234	1.99000E-02	2.000E-04	5.41000E-03	5.000E-05	2.05257E-04	-1.8550E-03
235	3.14300	3.000E-03	0.85700	10.000E-04	3.24182E-02	-3.0720
236	8.28000E-02	7.000E-04	2.27000E-02	2.000E-04	8.54034E-04	0.6193
238	96.7550	3.000E-03	26.7200	4.000E-02	0.99797	-1.9583
TOTAL U			27.6051	4.001E-02	1.03145	-4.4128
PU239	94.6161	1.499E-02	4.20100E-02	7.100E-04	1.56247E-03	1.5151
240	4.62696	1.399E-02	2.06300E-03	3.000E-05	7.64085E-05	7.4403E-02
241	0.74047	5.542E-03	3.31528E-04	5.544E-06	1.22279E-05	1.1957E-02
242	1.64372E-02	7.016E-04	7.39000E-06	4.600E-07	2.71441E-07	2.6652E-04
TOTAL PU			4.44119E-02	7.107E-04	1.65138E-03	1.6017
						FTOT 2.9397E-03
						BURNUP(MWD/MTU)
						U-235 2361.1922
						U-238 193.83175
						PU239 154.96407
						PU241 0.6757839
						TOTAL BU 2710.6638

U-238 ATOMS PER ML = 6.76065E 16 1.0121E 14
 PU-239/U-238 ATOM RATIO = 1.56564E-03 2.6564E-05
 TOTAL PU TO U MASS RATIO = 1.60883E-03 2.5849E-05

U-235 DEPLETION = 9.0405E-02
 NET NO. FISSILE GM DESTROYED/MWD = 0.5699
 ALPHA-25 = 0.2569

EOL CS-137 DPM/GU = 1.81997E 10 9.0999E 08
 MWD/MTU FROM CS-137 DATA = 2519.83 125.99
 CS BURNUP/ISOTOPIC BURNUP = 0.92960

SIGMA A25 X THETA=LN(N025/N25) = 9.4756E-02
 SIGMA A28 X THETA=LN(N028/N28) = 2.0300E-03
 (U235 SIGMA A)/(U238 SIGMA A) = 46.677

EOL PU/U FROM X-RAY FLUORESCENCE = 1.6313E-03
 X-RAY FLUORESCENCE PU/U / ISOTOPIC PU/U = 1.0140

	D E C A Y		D A T A		TOTAL DECAY
	OUTOFCORE DECAY TIME(Y)	INCORE DECAY TIME(Y)	OUTOFCORE DECAY	INCORE DECAY	
PU-241(ISOTOPIC)	1.9670	1.4170	0.9019	0.9637	0.8692
PU-241(XRAY FL.)	1.9700	1.4170	0.9017	0.9637	0.8690
CS-137	2.8930	1.4170	0.9341	0.9835	0.9187

A S S U M P T I O N S
 BOL U234 A/O= 0.020 DELTA-28= 0.0782 U-236 LOSS= 0.98656 AVG MEV/FISS= 203.5 T1/2 PU241(Y)= 13.2
 BOL U235 A/O= 3.440 ALPHA-49= 0.5551 PU-242 LOSS= 1.00406 AVG CS YIELD= 0.0618 T1/2 CS137(Y)= 29.4
 BOL U236 A/O= 0.020 ALPHA-41= 0.3860 U238 CONVGENCE= 0.00001
 BOL U238 A/O= 96.520 NOTE. ERROR IS 2 SIGMA. IT INCLUDES 2 SIGMA UNCERTAINTY IN MEASURED DATA ONLY.

SUMMARY OF REBUF RESULTS FOR A TYPICAL ANALYSIS OF A LOW BURNUP
 SAMPLE FROM THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

FIGURE A2.1

R E B U F - 2 A C O D E

ISOTOPE	ATOM PERCENT A/O	I S O T O P I C		C O N T E N T		NET PRODUCTION KG/MTU(BOL)	F I S S I O N S	
		ERROR	PPM	NO. DENSITY N(EOL)/N28(BOL)	ERROR		F/N28(BOL)	F/FTOT
U-234	1.80000E-02	4.000E-03	1.26265E-02	2.806E-03	1.82314E-04	-2.3636E-02	U-235	1.052E-02 0.690
235	2.22000	2.000E-02	1.56399	5.524E-02	2.24854E-02	-12.542	U-238	1.191E-03 7.804E-02
236	0.26700	6.000E-03	0.18890	7.723E-03	2.70432E-03	2.3910	PU239	3.379E-03 0.221
238	97.4900	0.200	69.5602	2.371	0.98743	-12.136	PU241	1.675E-04 1.098E-02
TOTAL U			71.3257	2.432	1.01280	-22.311	FTOT 1.5256E-02	
							BURNUP(MWD/MTU)	
PU239	78.5671	0.397	0.43375	1.700E-02	6.13196E-03	5.9461	U-235	9688.2872
240	13.1011	7.934E-02	7.26373E-02	2.903E-03	1.02251E-03	0.9957	U-238	1082.5868
241	7.56409	6.674E-02	4.21132E-02	1.705E-03	5.90357E-04	0.5773	PU239	3226.7424
242	0.76762	1.388E-02	4.29151E-03	1.136E-04	5.99108E-05	5.8825E-02	PU241	161.64770
TOTAL PU			0.55283	2.168E-02	7.80473E-03	7.5778	TOTAL BU 14159.264	

U-238 ATOMS PER ML = 1.76000E 17 6.0000E 15
 PU-239/U-238 ATOM RATIO = 6.21000E-03 1.2000E-04
 TOTAL PU TO U MASS RATIO = 7.75075E-03 1.5024E-04

U-235 DEPLETION = 0.3691
 NET NO. FISSILE GM DESTROYED/MWD = 0.4251
 ALPHA-25 = 0.2506

EOL CS-137 DPM/GU = 1.04130E 11 6.2520E 09
 MWD/MTU FROM CS-137 DATA = 14314.5 859.45
 CS BURNUP/ISOTOPIC BURNUP = 1.01096

SIGMA A25 X THETA=LN(N025/N25) = 0.4606
 SIGMA A28 X THETA=LN(N028/N28) = 1.2647E-02
 (U235 SIGMA A)/(U238 SIGMA A) = 36.420

EOL PU/U FROM X-RAY FLUORESCENCE = 7.3196E-03
 X-RAY FLUORESCENCE PU/U / ISOTOPIC PU/U = 0.9444

	OUTOFCORE DECAY TIME(Y)	INCORE DECAY TIME(Y)	CUTOFCORE DECAY	INCORE DECAY	TOTAL DECAY
PU-241(ISOTOPIC)	2.1860	1.4170	0.8916	0.9637	0.8592
PU-241(XRAY FL.)	2.2600	1.4170	0.8881	0.9637	0.8559
CS-137	2.3150	1.4170	0.9469	0.9835	0.9313

A S S U M P T I O N S
 BOL U234 A/O= 0.020 DELTA-28= 0.0847 U-236 LOSS= 0.95110 AVG MEV/FISS= 204.9 T1/2 PU241(Y)= 13.2
 BOL U235 A/O= 3.440 ALPHA-49= 0.5524 PU-242 LOSS= 0.93992 AVG CS YIELD= 0.0627 T1/2 CS137(Y)= 29.4
 BOL U236 A/O= 0.020 ALPHA-41= 0.3E05 U238 CONVGENCE= 0.00001
 BOL U238 A/O= 96.520 NOTE. ERROR IS 2 SIGMA. IT INCLUDES 2 SIGMA UNCERTAINTY IN MEASURED DATA ONLY.

SUMMARY OF REBUF RESULTS FOR A TYPICAL ANALYSIS OF AN INTERMEDIATE
 BURNUP SAMPLE FROM THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

F5-C-A1 553 ZONE 3 (GE 10/21/64 G-44) PHASE 2 11/4/64 34

R E B U F - 2 A C G D E

I S O T O P I C		C O N T E N T		F I S S I O N S	
ISOTOPE	ATOM PERCENT	PARTS PER MILLION	NO. DENSITY	NET PRODUCTION	F/N28(BOL) F/FTOT
A/O	ERRGR	PPM	ERROR	N(EOL)/N28(BOL) KG/MTU(BOL)	
U-234	1.6000E-02	4.000E-03	1.61822E-02	4.048E-03	1.59832E-04 -4.4980E-02
235	1.65000	2.000E-02	1.67594	7.014E-02	1.64826E-02 -18.265
236	0.36900	8.000E-03	0.37640	1.714E-02	3.68612E-03 3.3310
238	97.9700	0.800	100.783	3.952	0.97867 -20.597
TOTAL U		102.852	4.033	0.99900	-35.577
PU239	70.5326	0.595	0.81067	3.568E-02	7.83914E-03 7.6015
240	16.1288	0.119	0.18615	8.454E-03	1.79259E-03 1.7455
241	11.4154	0.107	0.13230	6.057E-03	1.26873E-03 1.2406
242	1.92317	1.583E-02	2.23820E-02	1.012E-03	2.13745E-04 0.2099
TOTAL PU		1.15151	5.078E-02	1.11142E-02	10.798
U-238	ATOMS PER ML = 2.55000E 17	1.0000E 16			U-235 DEPLETION = 0.5375
PU-239/U-238	ATOM RATIO = 8.01000E-03	1.6000E-04			NET NO. FISSILE GM DESTROYED/MWD = 0.3979
TOTAL PU TO U	MASS RATIO = 1.11958E-02	2.2613E-04			ALPHA-25 = 0.2492
EOL CS-137	DPM/GU = 1.71377E 11	8.5740E 09			SIGMA A25 X THETA=LN(N025/N25) = 0.7712
MWD/MTU FROM CS-137	DATA = 23743.0	1187.9			SIGMA A28 X THETA=LN(N028/N28) = 2.1561E-02
CS BURNUP/ISOTOPIC	BURNUP = 1.00253				(U235 SIGMA A)/(U238 SIGMA A) = 35.766
EOL SR-90	DPM/GU = 1.43957E 11	1.0081E 10			EOL PU/U FROM X-RAY FLUORESCENCE = 1.1551E-02
MWD/MTU FROM SR-90	DATA = 27043.9	1893.9			X-RAY FLUORESCENCE PU/U / ISOTOPIC PU/U = 1.0317
SR BURNUP/ISOTOPIC	BURNUP = 1.14191				
O U T G C O R E		D E C A Y		D A T A	
DECAY TIME(Y)		DECAY TIME(Y)		DECAY	TOTAL DECAY
PU-241(ISOTOPIC)	1.4680	2.3570		0.9249	0.9406 0.8699
PU-241(XRAY FL.)	1.4080	2.3570		0.9287	0.9406 0.8736
CS-137	1.3780	2.3570		0.9680	0.9727 0.9416
SR-90	1.4740	2.3570		0.9642	0.9714 0.9366
A S S U M P T I O N S					
BOL U234 A/O=	0.020	DELTA-28= 0.0896	U-236 LOSS= 0.91496	AVG MEV/FISS= 205.5	T1/2 PU241(Y)= 13.2
BOL U235 A/O=	3.440	ALPHA-49= 0.5510	PU-242 LOSS= 0.85877	AVG CS YIELD= 0.0631	T1/2 CS137(Y)= 29.4
BOL U236 A/O=	0.020	ALPHA-41= 0.3776	U238 CONVGNC= 0.00001	AVG SR YIELD= 0.0443	T1/2 SR-90(Y)= 28.0
BOL U238 A/O=	96.520	NOTE. ERROR IS 2 SIGMA. IT INCLUDES 2 SIGMA UNCERTAINTY IN MEASURED DATA ONLY.			

SUMMARY OF REBUF RESULTS FOR A TYPICAL ANALYSIS OF A HIGH BURNUP
SAMPLE FROM THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

FIGURE A2.3

F5-SE-F5 8717 ZONE 3 (TLAB 9/2/64 T-125) PHASE 2 RUN ON 11/13/64

REBUF - 2A CODE

ISOTOPE	ISOTOPIC		CONTENT		F I S S I O N S	
	ATOM PERCENT	PARTS PER MILLION	NO. DENSITY	NET PRODUCTION	F/N28(BOL)	F/FTOT
	A/O	ERROR	PPM	ERROR	N(EOL)/N28(BOL)	KG/MTU(BOL)
U-234	1.45000E-02	2.000E-04	3.63900E-03	5.300E-05	1.47801E-04	-5.6401E-02
235	1.26200	1.300E-03	0.30930	2.600E-03	1.25185E-02	-22.045
236	0.43230	9.000E-04	0.10640	5.100E-04	4.28822E-03	3.9075
238	98.2907	1.600E-03	24.4000	0.200	0.97500	-24.141
TOTAL U			24.8193	0.200	0.99195	-42.335
PU239	61.2168	1.189E-02	0.16520	2.300E-03	6.57357E-03	6.3743
240	21.0745	1.090E-02	5.71100E-02	8.000E-04	2.26301E-03	2.2036
241	13.8442	7.126E-03	3.76732E-02	5.261E-04	1.48661E-03	1.4536
242	3.86453	1.188E-02	1.05600E-02	1.500E-04	4.14980E-04	0.4075
TOTAL PU			0.27054	2.496E-03	1.07382E-02	10.439
						TOTAL BU 30349.695

U-238 ATOMS PER ML = 6.17364E 16 5.0604E 14
 PU-239/U-238 ATOM RATIO = 6.74213E-03 1.0893E-04
 TOTAL PU TO U MASS RATIO = 1.09005E-02 1.3353E-04

U-235 DEPLETION = 0.6488
 NET NO. FISSILE GM DESTROYED/MWD = 0.4684
 ALPHA-25 = 0.2522

EOL CS-137 DPM/GU = 2.29433E 11 1.1466E 10
 MWD/MTU FROM CS-137 DATA = 31739.3 1586.2
 CS BURNUP/ISOTOPIC BURNUP = 1.04579
 EOL SR-90 DPM/GU = 1.54536E 11 1.0818E 10
 MWD/MTU FROM SR-90 DATA = 30039.4 2102.8
 SR BURNUP/ISOTOPIC BURNUP = 0.98978

SIGMA A25 X THETA=LN(N025/N25) = 1.0463
 SIGMA A28 X THETA=LN(N028/N28) = 2.5318E-02
 (U235 SIGMA A)/(U238 SIGMA A) = 41.325

EOL PU/U FROM X-RAY FLUORESCENCE = 1.1279E-02
 X-RAY FLUORESCENCE PU/U / ISOTOPIC PU/U = 1.0347

	OUTOFCORE DECAY TIME(Y)	INCORE DECAY TIME(Y)	OUTOFCORE DECAY	INCORE DECAY	TOTAL DECAY
PU-241(ISOTOPIC)	1.3530	2.3570	0.9314	0.9406	0.8761
PU-241(XRAY FL.)	1.4080	2.3570	0.9287	0.9406	0.8736
CS-137	1.3780	2.3570	0.9680	0.9727	0.9416
SR-90	1.4740	2.3570	0.9642	0.9714	0.9366

A S S U M P T I O N S

BOL U234 A/O= 0.020 DELTA-28= 0.0734 U-236 LOSS= 0.88172 AVG MEV/FISS= 205.9 T1/2 PU241(Y)= 13.2
 BOL U235 A/O= 3.440 ALPHA-49= 0.5502 PU-242 LOSS= 0.79655 AVG CS YIELD= 0.0633 T1/2 CS137(Y)= 29.4
 BOL U236 A/O= 0.020 ALPHA-41= 0.3758 U238 CONVGENCE= 0.00001 AVG SR YIELD= 0.0429 T1/2 SR-90(Y)= 28.0
 BOL U238 A/O= 96.520 NUTE. ERROR IS 2 SIGMA. IT INCLUDES 2 SIGMA UNCERTAINTY IN MEASURED DATA ONLY.

SUMMARY OF REBUF RESULTS FOR A TYPICAL ANALYSIS OF A HIGH BURNUP
 SAMPLE FROM THE YANKEE PERTURBED NEUTRON SPECTRUM

FIGURE A2.4

- B. The experimental isotopic composition received from mass spectrometric analysis and expressed as atom percent.

(a/o) ²⁴	(a/o) ⁴⁹
(a/o) ²⁵	(a/o) ⁴⁰
(a/o) ²⁶	(a/o) ⁴¹
(a/o) ²⁸	(a/o) ⁴²

- C. The experimental isotopic concentration received from mass spectrometric analysis and expressed as parts per million, by weight of the above isotopes in the final sample solution.

(ppm) ²⁴	(ppm) ⁴⁹
(ppm) ²⁵	(ppm) ⁴⁰
(ppm) ²⁶	(ppm) ⁴¹
(ppm) ²⁸	(ppm) ⁴²

or via an input option, the ppm data listed above may be replaced with the following input data.

Pu-239/U-238 atom ratio, and U-238 atoms/ml of sample solution

- D. Radiochemical data and associated constants for Cs-137 and Sr-90 burnup indicators (optional)

Half-life, years

U-235 fission yield, fission⁻¹

U-238 fission yield, fission⁻¹

Pu-239 fission yield, fission⁻¹

Pu-241 fission yield, fission⁻¹

Time from core shutdown to radiochemical analysis, years.

- E. Pu/U data from X-ray fluorescence analysis (optional).

Pu/U mass ratio at time of analysis

Time from core shutdown to analysis, years.

- F. Energy conversion constants (See Reference 3).

U-235	203.3	Mev/fission
U-238	200.7	Mev/fission
U-239	210.8	Mev/fission
U-241	213.0	Mev/fission

G. Atomic Weights

U-234	234.1141	Pu-239	239.1270
U-235	235.1175	Pu-240	240.1291
U-236	236.1196	Pu-241	241.1322
U-238	238.1252	Pu-242	242.1344

H. Assumed values of constants; δ^{28} , α^{49} , and α^{41} . The values of δ^{28} , α^{49} and α^{41} are defined as follows:

$$\begin{aligned}\delta^{28} &= \text{U-238 fissions} / (\text{U-235 fissions} + \text{Pu-239 fissions} + \text{Pu-241 fissions}) \\ \alpha^{49} &= \text{Pu-239 captures} / \text{Pu-239 fissions} \\ \alpha^{41} &= \text{Pu-241 captures} / \text{Pu-241 fissions}\end{aligned}$$

Or, to obtain more accurate results, the above constants can be replaced with burnup dependent values which have been programmed into the code. These are shown in Figure A2.5 as a function of the U-235 atom depletion.

The parameters α^{49} and α^{41} are assumed to be independent of the neutron spectrum; that is, the values of α^{49} and α^{41} obtained from the correlations on Figure A2.5 are used for samples from the intermediate and perturbed neutron spectra, as well as for samples from the asymptotic spectrum. However, the values of δ^{28} obtained from the correlation on Figure A2.5 are reduced by 0.01 and 0.02 for samples from the intermediate and perturbed neutron spectra, respectively.

I. The time between end-of-life and isotopic analysis expressed in years, and the in-core irradiation time expressed in years.

Variation of α^{49} , α^{41} , and δ^{28} with U-235 Depletion

Figure A2.5

2) Output of the REBUF-2 Code

Using the input parameters, REBUF-2 calculates all of the following quantities and, with the exception of item (D), prints them out.

- A. Isotopic atom percent
- B. The Pu-241 in-core and out-of-core decay correction factors to be used with isotopic data
- C. Isotopic parts per million, by weight of the sample solution
- D. Isotopic atom density relative to the end-of-life U-238 atom density
- E. Isotopic end-of-life atom density relative to the beginning-of-life U-238 atom density
- F. Net production of U and Pu isotopes
- G. U-238 atoms per ml of sample solution
- H. Pu-239/U-238 atom ratio
- I. Plutonium-to-uranium mass ratio, determined from isotopic data
- J. Number of fissions in U-235, U-238, Pu-239, and Pu-241 per unit volume of fuel relative to the beginning-of-life U-238 atom density
- K. The fraction of the total accumulated fissions due to accumulated fissions in U-235, U-238, Pu-239, and Pu-241
- L. The total burnup and the burnup due to each of the fissionable* isotopes expressed in MWD/MTU
(1 MTU = 1×10^6 grams of initial uranium loading)
- M. U-235 atom depletion
- N. Correction factor for U-236 loss due to neutron absorption
- O. Correction factor for Pu-242 loss due to neutron absorption
- P. Net number of grams of fissile materials destroyed per mega-watt day of energy release (NGD/MWD)
- Q. Alpha-25, where
 $\alpha^{25} \equiv \text{U-235 captures/U-235 fissions}$
- R. Pseudo flux variables, σ_a^{25} and σ_a^{28}

* As used herein, fissionable is defined as the property of a heavy nucleus such that it can be broken into two nearly equal fragments (all isotopes of Th, U, Pu) and fissile is defined as the property of a heavy nucleus such that it is fissionable by room temperature thermal neutrons (U-233, U-235, Pu-239, Pu-241).

- S. The U-235 to U-238 effective absorption cross section ratio
- T. The Cs-137 dpm/gm U at end-of-core life, the burnup inferred from the Cs-137 data, and the ratio of Cs-137 burnup to isotopic burnup
- U. The Sr-90 dpm/gm U at end of core life, the burnup inferred from the Sr-90 data, and the ratio of the Sr-90 burnup to isotopic burnup
- V. The X-ray fluorescence Pu/U mass ratio at end of core life, and the ratio of the X-ray Pu/U to the isotopic Pu/U
- W. In-core and out-of-core radioactive decay
- X. The assumed initial composition of fresh fuel
- Y. The assumed constants used in isotopic data reduction
- Z. The assumed constants used in the radiochemical data reduction

3) Calculations Performed by the REBUF-2 Code

Before presenting the equations which the code solves to obtain the quantities listed above, it is convenient to make the following definitions.

$N_o^i \equiv$ beginning-of-life atomic number density of isotope i , where i is the uranium isotopes 24, 25, 26, and 28 and the plutonium isotopes 49, 40, 41, and 42.

$N_f^i \equiv$ end-of-life atomic number density of isotope i

$$\begin{array}{ll} R_o^{25/28} = N_o^{25}/N_o^{28} & R^{28/28} = N_f^{28}/N_o^{28} \\ R_o^{26/28} = N_o^{26}/N_o^{28} & R^{49/28} = N_f^{49}/N_f^{28} \\ R_f^{24/28} = N_f^{24}/N_f^{28} & R^{40/28} = N_f^{40}/N_f^{28} \\ R_f^{25/28} = N_f^{25}/N_f^{28} & R^{41/28} = N_f^{41}/N_f^{28} \\ R_f^{26/28} = N_f^{26}/N_f^{28} & R^{42/28} = N_f^{42}/N_f^{28} \end{array}$$

$t_1 \equiv$ decay time, from core end-of-life time to the time of the isotopic analysis, in years

$t_2 \equiv$ in-core irradiation time, in years

$\lambda \equiv 0.0525 \text{ year}^{-1} =$ decay constant for Pu-241

$F^{25} \equiv$ Number of U-235 fissions per unit volume of fuel

$F^{28} \equiv$ Number of U-238 fissions per unit volume of fuel

$F^{49} \equiv$ Number of Pu-239 fissions per unit volume of fuel

$F^{41} \equiv$ Number of Pu-241 fissions per unit volume of fuel

$BU^{25} \equiv$ Burnup of U-235 in MWD/MTU

$BU^{28} \equiv$ Burnup of U-238 in MWD/MTU

$BU^{49} \equiv$ Burnup of Pu-239 in MWD/MTU

$BU^{41} \equiv$ Burnup of Pu-241 in MWD/MTU

- $(\text{COR})_1^{41} \equiv$ Pu-241 inventory correction for out-of-core decay
 $(\text{COR})_2^{41} \equiv$ Pu-241 inventory correction for in-core decay
 $(\text{COR})^{26} \equiv$ U-236 inventory correction for neutron absorption
 $(\text{COR})^{42} \equiv$ Pu-242 inventory correction for neutron absorption
 $A^i \equiv$ Atomic weight of isotope i
 $N_a \equiv$ The Avagadro number

With these definitions and the input parameters the code performs the following calculations:

A. Isotopic Atom Percent

The uranium isotopic atom percents remain the same as the input values. However, the plutonium atom percents are recalculated after the Pu-241 parts per million is corrected for out-of-core decay.

B. The Pu-241 In-Core and Out-of-Core Decay Correction Factors

$$\begin{aligned}
 (\text{COR})_1^{41} &= e^{-\lambda t_1} \\
 (\text{COR})_2^{41} &= \frac{1 - e^{-\lambda t_2}}{\lambda t_2}
 \end{aligned}$$

C. Isotopic Parts per Million

The parts per million of Pu-239, Pu-240, and Pu-242 remain unchanged. However, the value of Pu-241 ppm is corrected for out-of-core decay.

$$(\text{ppm})_c^{41} = (\text{ppm})^{41} / (\text{COR})_1^{41}$$

For those cases where the input isotopic ppm data are replaced with the Pu-239/U-238 atom ratio and the U-238 atom/ml, the following calculations are performed:

$$(\text{ppm})^{24} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{(a/o)^{24}}{(a/o)^{28}} \times \frac{A^{24} \times 10^6}{N_a}$$

$$(\text{ppm})^{25} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{(\text{a/o})^{25}}{(\text{a/o})^{28}} \times \frac{A^{25} \times 10^6}{N_a}$$

$$(\text{ppm})^{26} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{(\text{a/o})^{26}}{(\text{a/o})^{28}} \times \frac{A^{26} \times 10^6}{N_a}$$

$$(\text{ppm})^{28} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{A^{28} \times 10^6}{N_a}$$

$$(\text{ppm})^{49} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{N^{49}}{N^{28}} \times \frac{A^{49} \times 10^6}{N_a}$$

$$(\text{ppm})^{40} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{(\text{a/o})^{40}}{(\text{a/o})^{49}} \times \frac{N^{49}}{N^{28}} \times \frac{A^{40} \times 10^6}{N_a}$$

$$(\text{ppm})^{41} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{(\text{a/o})^{41}}{(\text{a/o})^{49}} \times \frac{N^{49}}{N^{28}} \times \frac{A^{41} \times 10^6}{N_a}$$

$$(\text{ppm})^{42} = \text{U-238} \frac{\text{atom}}{\text{ml}} \times \frac{(\text{a/o})^{42}}{(\text{a/o})^{49}} \times \frac{N^{49}}{N^{28}} \times \frac{A^{42} \times 10^6}{N_a}$$

D. Isotopic Atom Density Relative to the End-of-Life U-238 Atom Density

For the uranium isotopes, the atomic ratio of each isotope to the end-of-life U-238 is just the ratio of the atom percents.

$$R^{i/28} = (\text{a/o})^i / (\text{a/o})^{28}$$

For the plutonium isotopes,

$$R^{49/28} = A^{28}/A^{49} \times (\text{ppm})^{49}/(\text{ppm})^{28}$$

$$R^{40/28} = A^{28}/A^{40} \times (\text{ppm})^{40}/(\text{ppm})^{28}$$

$$R^{41/28} = A^{28}/A^{41} \times (\text{ppm})^{41}_c/(\text{ppm})^{28}$$

$$R^{42/28} = A^{28}/A^{42} \times (\text{ppm})^{42}/(\text{ppm})^{28}$$

where $(\text{ppm})^{41}_c$ is the decay corrected Pu-241 parts per million.

E. Isotopic End-of-Life Atom Density Relative to the Beginning-of-Life U-238 Atom Density

$$N_f^i / N_o^{28} = R^{i/28} R^{28/28}$$

where, by iteration,

$$R^{28/28} = 1 - R^{28/28} [R^{49/28} + (1 + \alpha^{49}) F^{49}/N_f^{28} + F^{28}/N_f^{28}]$$

F. Net Production of U and Pu Isotopes, Relative to the Initial Uranium Loading

Let the average atomic weight of the fresh fuel be:

$$\bar{A} = \left[A^{24}(a/o)_o^{24} + A^{25}(a/o)_o^{25} + A^{26}(a/o)_o^{26} + A^{28}(a/o)_o^{28} \right] \times 10^{-2}$$

Then, for the net production of uranium isotopes,

$$\frac{\text{kg U-234}}{\text{MTU}} = (R^{24/28} R^{28/28} - R_o^{24/28}) \times \frac{A^{24}}{\bar{A}} \times (a/o)_o^{28} \times 10$$

$$\frac{\text{kg U-235}}{\text{MTU}} = (R^{25/28} R^{28/28} - R_o^{25/28}) \times \frac{A^{25}}{\bar{A}} \times (a/o)_o^{28} \times 10$$

$$\frac{\text{kg U-236}}{\text{MTU}} = (R^{26/28} R^{28/28} - R_o^{26/28}) \times \frac{A^{26}}{\bar{A}} \times (a/o)_o^{28} \times 10$$

$$\frac{\text{kg U-238}}{\text{MTU}} = (R^{28/28} - 1.0) \times \frac{A^{28}}{\bar{A}} \times (a/o)_o^{28} \times 10$$

$$\frac{\text{kg U}}{\text{MTU}} = \frac{\text{kg U-234}}{\text{MTU}} + \frac{\text{kg U-235}}{\text{MTU}} + \frac{\text{kg U-236}}{\text{MTU}} + \frac{\text{kg U-238}}{\text{MTU}}$$

For the net production of plutonium isotopes,

$$\frac{\text{kg Pu-239}}{\text{MTU}} = R^{49/28} R^{28/28} \times \frac{A^{49}}{\bar{A}} \times (a/o)_o^{28} \times 10$$

$$\frac{\text{kg Pu-240}}{\text{MTU}} = R^{40/28} R^{28/28} \times \frac{A^{40}}{\bar{A}} \times (a/o)_o^{28} \times 10$$

$$\frac{\text{kg Pu-241}}{\text{MTU}} = R^{41/28} R^{28/28} \times \frac{A^{41}}{\bar{A}} \times (a/o)_O^{28} \times 10$$

$$\frac{\text{kg Pu-242}}{\text{MTU}} = R^{42/28} R^{28/28} \times \frac{A^{42}}{\bar{A}} \times (a/o)_O^{28} \times 10$$

$$\frac{\text{kg Pu}}{\text{MTU}} = \frac{\text{kg Pu-239}}{\text{MTU}} + \frac{\text{kg Pu-240}}{\text{MTU}} + \frac{\text{kg Pu-241}}{\text{MTU}} + \frac{\text{kg Pu-242}}{\text{MTU}}$$

G. U-238 Atoms Per Milliliter of Sample Solution

If U and Pu isotopic ppm data are input,

$$\text{U-238} \frac{\text{atom}}{\text{ml}} = \frac{(\text{ppm})^{28} \times N_a}{A^{28} \times 10^6}$$

H. Pu-239/U-238 Atom Ratio

If U and Pu isotopic ppm data are input,

$$\frac{N^{49}}{N^{28}} = \frac{(\text{ppm})^{49} \times A^{28}}{(\text{ppm})^{28} \times A^{49}}$$

I. Plutonium-to-Uranium Mass Ratio, Determined from Isotopic Data

$$\frac{\text{Pu mass}}{\text{U mass}} = \frac{(\text{ppm})^{49} + (\text{ppm})^{40} + (\text{ppm})^{41} + (\text{ppm})^{42}}{(\text{ppm})^{24} + (\text{ppm})^{25} + (\text{ppm})^{26} + (\text{ppm})^{28}}$$

J. Number of Fissions in U-235, Pu-239, Pu-241, and U-238 per Unit Volume of Fuel Relative to the Beginning-of-Life U-238 Atom Density

First relating the isotopic fissions to the end-of-life U-238 atomic density one obtains:

$$F^{25}/N_f^{28} = \frac{R_O^{25/28} + R_O^{26/28}}{R^{28/28}} - R^{25/28} - R^{26/28} (\text{COR})^{26}$$

$$F^{49}/N_f^{28} = \frac{1}{\alpha^{49}} \left[R^{40/28} + \frac{R^{41/28}}{(\text{COR})^{41}_2} + (\text{COR})^{42} R^{42/28} \left(\frac{1 + \alpha^{41}}{\alpha^{41}} \right) \right]$$

$$F^{41}/N_f^{28} = (\text{COR})^{42} R^{42/28}/\alpha^{41}$$

$$F^{28}/N_f^{28} = \delta^{28} (F^{25}/N_f^{28} + F^{49}/N_f^{28} + F^{41}/N_f^{28})$$

To relate the isotopic fissions to the beginning-of-life U-238 atomic density it is necessary to multiply the above quantities by $R^{28/28}$.

$$F^{25}/N_o^{28} = R^{28/28} F^{25}/N_f^{28}$$

$$F^{49}/N_o^{28} = R^{28/28} F^{49}/N_f^{28}$$

$$F^{41}/N_o^{28} = R^{28/28} F^{41}/N_f^{28}$$

$$F^{28}/N_o^{28} = R^{28/28} F^{28}/N_f^{28}$$

- K. The Fraction of the Total Accumulated Fissions Due to Fissions in U-235, Pu-239, Pu-241 and U-238

$$\text{Fraction of total fissions in isotope } i = \frac{F^i/N_o^{28}}{\text{Total Fissions}/N_o^{28}}$$

- L. The Total Burnup and the Burnup Due to Each of the Fissionable Isotopes Expressed in MWD/MTU

In calculating the burnup due to each of the fissionable isotopes, the following constants are obtained:

$$F^{25} \text{ fissions/MWD} = \frac{1 \text{ fission}}{\text{EPF5 Mev}} \times \frac{0.2247 \times 10^{17} \text{ Mev}}{\text{Watt-Hr}} \times \frac{10^6 \text{ Watts}}{\text{Mega-Watt}} \times \frac{24 \text{ Hr}}{\text{Day}}$$

$$F^{49} \text{ fissions/MWD} = \frac{0.2247 \times 10^{17} \times 10^6 \times 24}{\text{EPF9}}$$

$$F^{41} \text{ fissions/MWD} = \frac{0.2247 \times 10^{17} \times 10^6 \times 24}{\text{EPF1}}$$

$$F^{28} \text{ fissions/MWD} = \frac{0.2247 \times 10^{17} \times 10^6 \times 24}{\text{EPF8}}$$

Where, EPF5, EPF9, EPF1, and EPF8 are the Mev/fission in U-235, Pu-239, Pu-241, and U-238, respectively.

$$\text{Initial U-238 atoms/MTU} = 0.6025 \times 10^{24} \frac{\text{atoms}}{\text{gm.at.wt.}} \times \frac{1 \text{ gm.at.wt.}}{\bar{A} \text{ gm}}$$

$$\times \frac{10^6 \text{ gm}}{\text{MTU}} \times \frac{(a/o)_0^{28}}{100}$$

REBUF then makes the following calculation for isotopes U-235, Pu-239, Pu-241, and U-238:

$$BU^i = \frac{F^i/N_O^{28} \frac{\text{fission}}{\text{atom}} \times \text{Initial U-238} \frac{\text{atom}}{\text{MTU}}}{F^i \frac{\text{fission}}{\text{MWD}}}$$

M. U-235 Atom Depletion

The depletion of U-235 is defined as that fraction of the original U-235 atoms which are destroyed.

$$D^{25} = (R_O^{25/28} - R^{28/28} R^{25/28})/R_O^{25/28}$$

N. Correction Factor for U-236 Loss Due to Neutron Absorption

The U-236 formed during irradiation is the sum of the U-236 remaining at the end-of-life (corrected for the beginning-of-life U-236) and the Np-237 formed during irradiation. One can define a U-236 correction factor as follows:

$$(\text{COR})^{26} = \frac{\text{Gross U-236 Production}}{\text{Net U-236 Production}}$$

Using data obtained from a LEOPARD calculation of the Yankee asymptotic unit cell, the following correlations were obtained, and programmed into the REBUF-2 code:

Over a burnup range of 0 to 2,200 MWD/MTU,

$$(\text{COR})^{26} = 1.0011 + 0.27593 (D^{25}) - 1.3529 (D^{25})^2$$

From 2,200 to 40,000 MWD/MTU,

$$(\text{COR})^{26} = 1.0034 + 0.11921 (D^{25}) - 0.099193 (D^{25})^2 + 0.34849 (D^{25})^3$$

O. Correction Factor for Pu-242 Loss Due to Neutron Absorption

To account for absorptions in Pu-242, an analytic expression which relates the gross/net Pu-242 production to the U-235 atom depletion (D^{25}) was programmed into the REBUF-2 code. The relationship, which is a least squares fit of Pu-242 data taken from a LEOPARD calculation of the Yankee unit cell to a burnup of 40,000 MWD/MTU is given below:

$$(COR)^{42} = \frac{N^{42}(\text{gross})}{N^{42}(\text{net})} = 1.0 - 0.1189 (D^{25}) + 0.7895 (D^{25})^2$$

P. Net Number of Grams of Fissile Materials Destroyed per MWD

The net number of grams of fissile material destroyed per MWD is defined as the mass of U-235 destroyed/MWD less the net mass of Pu-239 and Pu-241 formed/MWD.

Net number of grams of fissile material destroyed per MWD = $NGD/MWD =$

$$\frac{(w/f)_O^{28} \times 10^6}{(A^{28}) \times \text{total burnup}} \left[(A^{25}) (D^{25}) R_O^{25/28} - (A^{49}) R^{49/28} R^{28/28} - (A^{41}) R^{41/28} R^{28/28} \right]$$

where $(w/f)_O^{28}$ is the beginning-of-life U-238 weight fraction.

Q. Alpha-25

The effective alpha-25 is defined as the ratio of total captures-to-total fissions in U-235.

$$\begin{aligned} \text{Let } C^{25} &= \text{the total captures in U-235} \\ \text{then } C^{25}/N_O^{28} &= R^{26/28} R^{28/28} (COR)^{26} - R_O^{26/28} \\ \text{and } \alpha^{25} &= \frac{C^{25}/N_O^{28}}{F^{25}/N_O^{28}} \end{aligned}$$

R. Pseudo Flux Variables, $\sigma_a^{25}\theta$ and $\sigma_a^{28}\theta$

In certain curve-fitting procedures, it is convenient to use the quantities $\sigma_a^{25}\theta$ and/or $\sigma_a^{28}\theta$ as independent variables. These quantities are herein defined as:

$$\sigma_a^{25}\theta = \ln \left(\frac{N_o^{25}/N_o^{28}}{N^{25}/N_o^{28}} \right), \text{ and}$$

$$\sigma_a^{28}\theta = \ln \left(\frac{N_o^{28}}{N^{28}} \right).$$

The expressions are obtained by solving the depletion equations for U-235 and U-238, respectively.

$$\frac{dN^{25}}{dt} = \sigma_a^{24} N^{24}\phi - \sigma_a^{25} N^{25}\phi$$

$$\frac{dN^{28}}{dt} = - \sigma_a^{28} N^{28}\phi$$

The assumptions are:

$$\sigma_a^{24} N^{24} < \sigma_a^{25} N^{25}$$

$$d\theta = \phi dt$$

S. The U-235 to U-238 Effective Absorption Cross Section Ratio

$$\langle \sigma_a^{25}/\sigma_a^{28} \rangle = \frac{\sigma_a^{25}\theta}{\sigma_a^{28}\theta}$$

T. Cesium-137 Data

$$A^{EOL} = (1/e^{-\lambda t_1}) A^{TOA}$$

$$BU^{Cs} = \frac{(1.854 \times 10^{-24} \frac{MWD}{Mev}) (10^6 \frac{gm}{MTU}) (E_{eff} \frac{Mev}{fission}) (A^{TOA} \frac{1}{min.gm U})}{(e^{-\lambda t_1}) (\frac{1 - e^{-\lambda t_2}}{\lambda t_2}) (Y_{eff} \frac{1}{fission}) (\lambda \frac{1}{min})}$$

Where,

EOL	≡	end-of-life
TOA	≡	time of analysis
A	≡	Cs-137 activity (dpm/gm U)
t_1	≡	post irradiation decay time
t_2	≡	in-core decay time
E_{eff}	≡	effective energy per fission, weighted by isotope (Mev/fission)
Y_{eff}	≡	fission yield of Cs-137, weighted by isotope (fission ⁻¹)
λ	≡	decay constant of Cs-137

U. Strontium-90 Data

Same as in item (T), except Sr-90 constants used in place of Cs-137 constants.

V. X-ray Fluorescence Pu/U Mass Ratio

Let the Pu-241 contribution to the Pu/U mass ratio determined from the X-ray fluorescence analysis be approximated with:

$$\text{ppm}_x^{41} \approx \frac{\text{ppm}_c^{41}}{\text{ppm}^{49} + \text{ppm}^{40} + \text{ppm}_c^{41} + \text{ppm}^{42}} \left(\frac{\text{Pu}}{\text{U}} \right)_x^{\text{TOA}}$$

Then to decay correct the X-ray Pu/U from time-of-analysis to end-of-life

$$\left(\frac{\text{Pu}}{\text{U}} \right)_x^{\text{EOL}} = \left(\frac{\text{Pu}}{\text{U}} \right)_x^{\text{TOA}} - \text{ppm}_x^{41} (1 - e^{-\lambda t_1})$$

where,

λ = Pu-241 decay constant

t_1 = time between end-of-life and X-ray fluorescence analysis

W. In-Core and Out-of-Core Radioactive Decay

The out-of-core decay is just $e^{-\lambda t_1}$. The in-core decay is approximated

with: $\frac{1 - e^{-\lambda t_2}}{\lambda t_2}$

These factors were described in items (B), (T), (U), and (V).

X. Initial Composition of Fresh Fuel

These are input values from item (A), Section (1).

Y. The Assumed Constants Used in the Isotopic Data Reduction

Included among the assumed constants are the values of δ^{28} , α^{49} , α^{41} , the U-236 absorption loss, the weighted Mev/fission conversion constant, and the Pu-241 half-life used in the various calculations described earlier.

The convergence criteria of the iterative procedure used to determine N_f^{28}/N_o^{28} is:

$$\epsilon \geq \left| (N_f^{28}/N_o^{28})_n - (N_f^{28}/N_o^{28})_{n-1} \right|$$

where,

n is nth iteration, and

$$\epsilon = 0.00001$$

Z. The Assumed Constants Used in the Radiochemical Data Reduction

Included among the assumed constants are the values of Cs-137, and Sr-90 half-life and weighted fission yield used in the calculations described previously.

Along with the measured data described above, the measured 2-standard deviation precision in the isotopic data and the 2-standard deviation accuracy in the Cs-137 and Sr-90 dpm/gU are input to the code. Using these values of uncertainty, printed out under the column headed "Error" in the REBUF code, and appropriate error propagation techniques, the 2-standard deviation on some inferred quantities are computed and printed out.

APPENDIX A3

Burnup and Fuel Characteristics Inferred from Mass Spectrometric, Radiochemical, and X-Ray Spectrographic Data

Using the relationships outlined in Appendix A2, the burnup and various fuel characteristics were inferred from the data listed in Table A1.1. The results are given in Table A3.1, which is described briefly in the following paragraphs. The numerical examples make use of the data obtained for sample #G-24, which represents Axial Sample Zone 5 in Fuel Rod #0982, core location G4-C-f6. The summary of REBUF results obtained for this sample is given on Figure A2.2

1. U-235 Atomic Depletion

The U-235 atom depletion is a measure of the fuel burnup and is simply the fraction of the initial U-235 atoms which were destroyed by neutron absorption. The U-235 formed by neutron capture in U-234 has been neglected since the initial composition of the fresh fuel contained approximately 0.02 atom percent U-234 and 3.44 atom percent U-235. At a burnup of 14,200 MWD/MTU, the error in the inferred value of U-235 atom depletion, induced by the neglect of the U-234 captures, is about 0.2%, relative. Thus, a value of U-235 atom depletion of 0.369 reported should actually be 0.370 (after round off), if the effects of U-234 captures are included.

2. Fission Density

This quantity can be used to determine the total number of fissions per unit volume of fuel. For example, in the intermediate burnup sample #G-24, the fission density (i.e., the accumulated fissions per unit volume of fuel) was 1.5% of the initial U-238 atom density in the fresh fuel. Thus,

$$\frac{\text{Total No. of Fissions}}{\text{cm}^3 \text{ of fuel}} = 0.01526 \frac{\text{fissions/cm}^3 \text{ fuel}}{\text{initial U-238 atom/cm}^3 \text{ fresh fuel}}$$
$$\times 0.02194 \times 10^{24} \text{ initial atoms U-238/cm}^3 \text{ fresh fuel} = 3.35 \times 10^{20} \frac{\text{fissions}}{\text{cm}^3 \text{ of fuel}}$$

TABLE A3.1
BURNUP AND FUEL CHARACTERISTICS INFERRED FROM RESULTS OF EYC MASS SPECTROMETRIC, RADIOCHEMICAL, AND X-RAY SPECTROGRAPHIC ANALYSES

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	U-235 Atom Depletion	Total Accumulated Fissions Relative to N ₀ ²⁸ (fission/atom)	Accumulated Fission Fraction (total fissions)				Burnup (MWD/MTU) ^b			N _U ²⁸ N ₀ (atoms/atom)	N _{Pu} ²⁸ N ₀ (atoms/atom x 10 ³)	Pu/U Mass Ratio ^c (gm/gm x 10 ³)		Alpha-25 ^e	NGD/MWD ^f
						U-235	U-238	Pu-239	Pu-241	Isotopic	Cs-137	Sr-90			Isotopic	X-ray		
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES																		
C3-C-a1	9578	3A ^d	T-132	.270	.01011	.755	.0759	.164	.00476	9,360	9,140		1.0204	5.394	5.32	5.63	.259	.481
		3B	T-133	.272	.01027	.753	.0759	.166	.00484	9,510	9,150		1.0202	5.527	5.45	5.61	.253	.470
C3-C-f6	9595	1	T-100	.158	.00550	.806	.0737	.119	.00147	5,080	5,360		1.0271	3.436	3.36	3.73	.270	.452
		3	T-101	.304	.01185	.719	.0766	.197	.00740	10,990	11,100		1.0176	6.477	6.40	7.00	.269	.437
		5	T-102	.306	.01182	.729	.0766	.188	.00676	10,960	10,840		1.0178	6.273	6.20	6.77	.263	.459
		6	T-103	.192	.00679	.791	.0743	.132	.00255	6,280	6,430		1.0253	3.946	3.87	3.92	.276	.485
C8-C-a6	569	3A ^d	T-134	.263	.00994	.757	.0757	.163	.00451	9,200	9,110		1.0207	5.332	5.25	5.48	.245	.470
		3B	T-135	.263	.00990	.754	.0757	.166	.00470	9,160	9,090		1.0207	5.366	5.29	5.61	.256	.469
C8-C-f1	5551	1	T-105	.163	.00570	.808	.0738	.116	.00142	5,270	5,520		1.0269	3.386	3.31	3.91	.263	.480
		3	T-106	.308	.01201	.727	.0767	.189	.00702	11,130	11,230		1.0176	6.308	6.23	6.45	.258	.458
		5	T-107	.300	.01164	.729	.0765	.188	.00685	10,790	11,220		1.0181	6.167	6.09	6.11	.261	.458
		6	T-108	.203	.00725	.801	.0745	.122	.00207	6,700	6,580		1.0249	3.893	3.82	3.86	.245	.516
E5-C-a6	7992	1	T-11	.234	.00855	.792	.0751	.129	.00337	7,900	6,720		1.0234	4.076	4.00	3.72	.234	.558
		3 ^{/E}	T-12	.333	.01367	.700	.0772	.213	.00962	12,680	13,690		1.0150	7.232	7.16	7.28	.240	.412
		3 ^{/E}	N-10-1 ^{/h}	.357	.01450	.701	.0778	.211	.01009	13,450	13,690		1.0142	7.184	7.12	7.29	.252	.455
		3 ^{/E}	N-10-2	.357	.01449	.701	.0778	.211	.01012	13,440	13,690		1.0142	7.138	7.08	7.29	.252	.458
		5	T-13	.373	.01499	.718	.0781	.194	.00947	13,900	13,700		1.0139	6.925	6.87	7.33	.234	.494
		6	T-14	.243	.00894	.780	.0753	.142	.00328	8,270	8,020		1.0224	4.645	4.57	4.64	.242	.509
E5-C-f1	7980	1	G-15	.210	.00759	.801	.0746	.123	.00211	7,020	6,540		1.0244	4.045	3.97		.229	.506
		3 ^{/E}	G	.363	.01470	.707	.0779	.206	.00963	13,640	14,410	13,960	1.0139	7.249	7.19	7.15	.244	.458
		3 ^{/E}	T ^{/i}	.360	.01472	.699	.0778	.213	.01024	13,660	14,410	14,050	1.0137	7.400	7.34	7.15	.248	.443
		3 ^{/E}	T-2 ^{/j}	.354	.01434	.700	.0777	.212	.00964	13,300	14,410	14,030	1.0142	7.356	7.30	7.15	.255	.439
		5	G	.368	.01491	.712	.0780	.200	.00954	13,830	13,900	13,230	1.0138	7.088	7.03	6.83	.236	.476
		6	G-16	.243	.00894	.770	.0753	.151	.00377	8,270	8,220		1.0223	4.737	4.66		.256	.500
F4-C-a1	8982	3 ^{/E}	G-2	.363	.01486	.709	.0779	.203	.01039	13,780	12,680		1.0137	7.272	7.21	7.17	.228	.451
		3 ^{/E}	T-9	.351	.01427	.698	.0776	.214	.00996	13,240	12,680		1.0141	7.456	7.39	7.17	.254	.426
		3 ^{/E}	N-5 ^{/l}	.345	.01370	.711	.0775	.202	.00941	12,700	12,680		1.0151	7.106	7.04	7.17	.262	.449
F4-C-a6	8994	1	T-87	.184	.00660	.791	.0742	.133	.00219	6,100	6,380		1.0254	3.959	3.88	3.90	.261	.454
		3	T-88	.345	.01394	.705	.0775	.208	.00935	12,930	13,130		1.0148	7.186	7.12	7.01	.252	.439
		5	T-89	.348	.01417	.701	.0776	.211	.00966	13,140	13,100		1.0144	7.286	7.22	7.21	.250	.435
		6	T-90	.232	.00858	.773	.0751	.149	.00338	7,940	8,090		1.0226	4.738	4.66	4.47	.249	.474
F4-C-f1	8977	1	T-37	.201	.00724	.797	.0745	.127	.00215	6,700	6,190		1.0248	3.973	3.90	3.93	.243	.498
		2	T-38	.323	.01280	.715	.0770	.200	.00817	11,870	11,810		1.0164	6.721	6.65	6.88	.258	.445
		3	T-39	.353	.01414	.708	.0777	.205	.00947	13,120	12,250		1.0146	7.096	7.03	7.49	.256	.459
		3 ^{/E}	N-14-1 ^{/h}	.354	.01424	.706	.0777	.207	.00950	13,210	12,250		1.0145	7.098	7.04	7.49	.254	.459
		3 ^{/E}	N-14-2	.351	.01416	.703	.0776	.210	.00967	13,140	12,250		1.0145	7.199	7.14	7.49	.257	.447
		4	T-40	.364	.01489	.695	.0779	.216	.01071	13,820	13,230		1.0135	7.428	7.37	7.59	.254	.446
		5	T-41	.360	.01456	.703	.0778	.209	.01000	13,510	13,300		1.0141	7.178	7.12	7.42	.252	.460
		6	T-42	.246	.00910	.774	.0754	.147	.00356	8,420	8,350		1.0221	4.749	4.67	4.62	.242	.502
F4-C-f6	19579	3 ^{/E}	G-3	.363	.01466	.706	.0779	.206	.00994	13,600	14,780		1.0138	7.414	7.35	7.44	.250	.448
		3 ^{/E}	T-10	.348	.01414	.690	.0776	.223	.01035	13,120	14,770		1.0141	7.659	7.60	7.44	.270	.409
		3 ^{/E}	N-6 ^{/l}	.342	.01392	.693	.0774	.219	.01058	12,920	14,770		1.0146	7.350	7.28	7.44	.265	.422

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Table A3.1 (Cont'd)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	U-235 Atom Depletion	Total Accumulated Fissions Relative to N _O ²⁸ (fission/atom)	Accumulated Fission Fraction (total fissions)				Burnup (MWD/MTU) ^{/b}			N _O ^U 28 (atoms/atom)	N _O ^{Pu} 28 (atoms/atom x 10 ³)	Pu/U Mass Ratio ^{/c} (gm/gm x 10 ³)			Alpha-25 ^{/e}	NGD/MWD ^{/f}
						U-235	U-238	Pu-239	Pu-241	Isotopic	Cs-137	Sr-90			Isotopic	X-Ray			
PHASE II, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES																			
F5-C-a1	553	1	G-43	.315	.01229	.731	.0768	.186	.00675	11,390	11,390	15,370	1.0173	6.284	6.21	6.61	.250	.470	
		3	G-44	.538	.02544	.603	.0823	.289	.02591	23,680	23,740	27,040	.9990	11.111	11.20	11.55	.249	.398	
		5	G-45	.540	.02554	.607	.0823	.284	.02576	23,780	23,660	21,180	.9991	10.906	10.98	11.69	.241	.408	
		6	G-46	.398	.01639	.704	.0787	.207	.01070	15,210	14,780	13,390	1.0120	7.473	7.43	7.87	.230	.481	
F5-C-f6	541	1	T-117	.300	.01173	.721	.0765	.196	.00713	10,880	11,570	15,630	1.0179	6.309	6.23	6.31	.267	.445	
		2	T-118	.491	.02240	.622	.0810	.275	.02182	20,840	21,280	23,640	1.0031	10.136	10.17	10.75	.256	.409	
		3	T-119	.520	.02465	.595	.0818	.297	.02643	22,950	24,340	23,880	.9998	11.070	11.14	11.17	.265	.387	
		4	T-120	.540	.02607	.588	.0823	.302	.02822	24,280	25,620	22,400	.9980	11.477	11.57	11.50	.256	.381	
		5	T-121	.528	.02499	.601	.0820	.290	.02616	23,270	23,630	20,150	.9997	10.814	10.88	10.96	.253	.403	
		6	T-122	.374	.01548	.688	.0782	.222	.01133	14,370	14,570	11,550	1.0127	7.631	7.58	7.77	.251	.443	
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES																			
G4-C-a1	0994	1/k	T-43	.193	.00696	.791	.0743	.133	.00230	6,430	6,640		1.0250	4.038	3.96	3.91	.252	.467	
		3	T-18	.329	.01304	.726	.0771	.190	.00760	12,080	11,760	11,560	1.0162	6.627	6.56	6.37	.238	.459	
		3	T-36	.321	.01262	.725	.0770	.191	.00762	11,700	11,710		1.0168	6.424	6.35	6.58	.252	.467	
		3	T-58	.324	.01288	.719	.0770	.196	.00783	11,940	11,660		1.0164	6.657	6.59	6.55	.246	.449	
		3	T-91	.323	.01286	.718	.0770	.197	.00781	11,920	11,780		1.0163	6.777	6.71	6.56	.246	.438	
		3	T-104	.323	.01282	.720	.0770	.195	.00784	11,880	11,500		1.0165	6.618	6.55	6.62	.248	.452	
		3	T-144	.322	.01279	.720	.0770	.196	.00779	11,850	11,460		1.0164	6.707	6.64	6.47	.246	.442	
		3	T-150	.328	.01296	.726	.0771	.190	.00761	12,010	11,730		1.0164	6.495	6.43	6.56	.242	.469	
		3	G-6	.333	.01320	.728	.0772	.188	.00756	12,240	11,630		1.0161	6.596	6.53	6.55	.235	.467	
		3	G-17	.324	.01286	.725	.0770	.191	.00759	11,920	11,780		1.0165	6.605	6.53	6.58	.240	.453	
		3	G-42	.324	.01293	.719	.0770	.196	.00775	11,990	11,570		1.0163	6.724	6.65	6.54	.242	.443	
		3	G-51	.321	.01271	.717	.0770	.198	.00782	11,780	11,620		1.0165	6.700	6.63	6.32	.257	.444	
		3	N-12-1/h	.318	.01268	.715	.0769	.197	.00796	11,760	11,350		1.0167	6.575	6.50	6.94	.245	.446	
		3	N-12-2	.324	.01275	.722	.0770	.193	.00772	11,820	11,600		1.0166	6.548	6.48	6.58	.254	.461	
		5	T-44	.363	.01501	.692	.0779	.220	.01095	13,930	13,930		1.0133	7.560	7.50	7.03	.247	.433	
		6	T-45	.244	.00916	.774	.0753	.147	.00360	8,480	8,200		1.0220	4.781	4.70	4.59	.228	.490	
		6/g	N-13-1/h	.243	.00898	.775	.0753	.146	.00368	8,310	8,200		1.0223	4.728	4.65	4.59	.243	.497	
		6/g	N-13-2	.243	.00899	.775	.0753	.146	.00353	8,310	8,200		1.0222	4.736	4.66	4.59	.243	.497	
G4-C-a6	0985	1	T	.207	.00739	.796	.0746	.127	.00225	6,830	6,590	6,180	1.0246	4.057/m	3.98		.251	.503	
		3	T	.363	.01475	.703	.0779	.209	.00996	13,680	13,410	12,340	1.0138	7.352/m	7.29		.247	.450	
		5/g	G	.378	.01545	.706	.0782	.206	.00974	14,330	13,500	12,340	1.0129	7.477/m	7.42		.233	.458	
		5	T	.369	.01506	.703	.0780	.209	.01054	13,970	13,500	12,380	1.0134	7.414/m	7.36		.242	.452	
		6	T	.251	.00919	.790	.0755	.130	.00359	8,500	9,590	7,560	1.0222	4.567/m	4.49		.233	.532	
G4-C-f1	0979	1	T	.195	.00700	.796	.0744	.127	.00233	6,470	6,020	5,660	1.0251	3.954/m	3.88		.246	.482	
		3	T	.366	.01486	.712	.0780	.201	.00945	13,780	12,650	14,000	1.0138	7.239/m	7.18		.232	.460	
		5/g	G	.363	.01470	.710	.0779	.203	.00946	13,630	13,180	13,320	1.0139	7.239/m	7.18		.239	.458	
		5	T	.363	.01470	.706	.0779	.207	.01000	13,630	13,180	13,360	1.0139	7.289/m	7.23		.247	.456	
		6	T	.252	.00938	.776	.0755	.144	.00407	8,680	8,120	8,730	1.0217	4.864/m	4.78		.232	.498	
G4-C-f6	0982	1	G-22	.198	.00713	.797	.0744	.126	.00213	6,600	6,790		1.0248	4.029	3.95	3.62	.239	.478	
		3	G-23	.324	.01287	.719	.0770	.196	.00806	11,930	12,880		1.0163	6.768	6.70	6.57	.249	.442	
		5	G-24	.369	.01526	.690	.0780	.221	.01098	14,160	14,310		1.0128	7.805	7.75	7.32	.251	.425	
		6	G-25	.246	.00916	.774	.0754	.147	.00351	8,470	8,110		1.0220	4.834	4.76	4.97	.236	.490	

TABLE A3.1 (CONT'D)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	U-235 Atom Depletion	Total Accumulated Fissions Relative to N ²⁸ _O (fission/atom)	Accumulated Fission Fraction				Burnup (MWD/MTU)/ ^b			N U 28 N _O (atoms) atom	N Pu 28 N _O (atoms) x 10 ³	Pu/U Mass Ratio/ ^c			Alpha-25/ ^e	NGD/MWD/ ^f
						(total fissions)				Isotopic	Cs-137	Sr-90			(g/g x 10 ³) Isotopic	X-Ray	Alpha-25/ ^e		
						U-235	U-238	Pu-239	Pu-241										
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER ASSEMBLIES																			
H3-C-a1	8537	5	G	.303	.01175	.753	.0766	.166	.00534	10,880	10,460	8,050	1.0183	5.857/m	5.78		.221	.485	
H3-C-a6	8510	Comp./n	T	.262	.01020	.729	.0757	.189	.00638	9,450			1.0197	6.056/m	5.97		.256	.390	
H3-C-a6	8510	1	T	.167	.00589	.816	.0739	.109	.00150	5,440	5,070	3,730	1.0268	3.349/m	3.28		.241	.495	
		2	T	.279	.01075	.748	.0760	.171	.00516	9,960	9,650	7,160	1.0195	5.698/m	5.62		.237	.461	
		3	T	.306	.01200	.732	.0766	.185	.00657	11,120	10,880	8,520	1.0177	6.252/m	6.18		.242	.456	
		4/g	G	.294	.01155	.716	.0764	.199	.00769	10,710	10,830	8,850	1.0180	6.369/m	6.29		.266	.426	
		4	T	.318	.01262	.730	.0769	.186	.00679	11,690	10,830	8,760	1.0169	6.444/m	6.37		.231	.456	
		5	T	.309	.01219	.738	.0767	.179	.00636	11,300	10,880	9,080	1.0176	6.206/m	6.13		.224	.461	
		6	T	.207	.00757	.805	.0746	.118	.00208	7,000	6,230	3,990	1.0245	3.966/m	3.89		.208	.503	
H3-C-f1	8528	1	G	.137	.00467	.832	.0733	.0942	.000846	4,320	4,030	4,690	1.0286	2.779/m	2.72		.257	.497	
		3A/d	T-142	.262	.00992	.750	.0757	.169	.00490	9,180	9,290		1.0206	5.459/m	5.38	5.26	.256	.457	
		3	G	.267	.01010	.752	.0758	.167	.00500	9,350	8,630	9,130	1.0204	5.457/m	5.38		.254	.468	
		3B	T-143	.214	.00791	.756	.0748	.165	.00426	7,330	8,060		1.0233	4.743/m	4.66	4.20	.278	.430	
		5/g	G	.261	.00978	.755	.0757	.165	.00453	9,050	7,990	7,060	1.0208	5.345/m	5.26		.261	.470	
		5/g	T	.249	.00937	.745	.0754	.175	.00491	8,680	7,980	7,110	1.0212	5.374/m	5.29		.272	.441	
		6	G	.167	.00590	.822	.0739	.103	.00122	5,450	5,230	4,260	1.0268	3.275/m	3.21		.230	.506	
H3-C-f6	8574	1	T-138	.148	.00507	.826	.0735	.100	.00101	4,680	4,610		1.0280	2.949	2.88	3.04	.257	.505	
		3	T-139	.295	.01128	.749	.0764	.169	.00554	10,440	9,910		1.0189	5.728	5.65	5.83	.244	.489	
		5	T-140	.284	.01095	.746	.0762	.172	.00550	10,140	9,980		1.0192	5.771	5.69	5.71	.238	.462	
		6	T-141	.180	.00642	.812	.0741	.113	.00158	5,930	5,750		1.0260	3.546	3.47	3.75	.234	.501	
H4-C-a1	2021	1	G-11	.188	.00658	.808	.0742	.116	.00174	6,080	5,490		1.0258	3.650	3.58	3.59	.261	.519	
		3	G-4	.333	.01305	.722	.0772	.193	.00764	12,100	14,100		1.0162	6.657	6.59	6.85	.258	.468	
		5	G-5	.327	.01292	.711	.0771	.204	.00859	11,980	12,720		1.0162	6.767	6.70	7.00	.270	.450	
		6	G-12	.224	.00805	.792	.0750	.131	.00248	7,450	6,620		1.0236	4.316	4.24		.255	.515	
H5-C-a1	1000	1	T-25	.194	.00697	.806	.0744	.118	.00178	6,440	6,200		1.0252	3.784	3.71	3.56	.234	.506	
		3/g	G-1	.345	.01409	.707	.0775	.205	.00972	13,070	12,840		1.0146	7.215	7.15	6.72	.235	.433	
		3	T-8	.335	.01348	.703	.0773	.211	.00906	12,500	12,840		1.0152	7.189	7.12	6.72	.261	.426	
		3/g	N-4/j	.322	.01290	.699	.0770	.215	.00935	11,970	12,830		1.0160	7.037	6.96	6.72	.271	.416	
		5	T-26	.358	.01440	.717	.0778	.196	.00886	13,350	13,280		1.0146	6.901	6.84	7.35	.236	.477	
		5/g	N-11-1/h	.345	.01394	.706	.0775	.207	.00930	12,930	13,280		1.0149	7.040	6.98	7.35	.249	.449	
		5/g	N-11-2	.342	.01380	.706	.0774	.207	.00936	12,800	13,280		1.0152	6.921	6.86	7.36	.251	.454	
		6	T-27	.236	.00870	.781	.0752	.140	.00309	8,050	6,820		1.0226	4.662	4.58	4.59	.235	.488	
H8-C-a1	21305	1	T-113	.168	.00586	.819	.0739	.106	.00124	5,420	5,340		1.0268	3.340	3.27	3.18	.245	.500	
		3	T-114	.306	.01192	.731	.0766	.186	.00674	11,050	11,270		1.0177	6.287	6.21	6.04	.253	.456	
		5	T-115	.304	.01182	.732	.0766	.185	.00666	10,960	11,220		1.0179	6.233	6.16	5.98	.253	.458	
		6	T-116	.208	.00741	.814	.0746	.110	.00172	6,840	6,260		1.0248	3.740	3.67	4.00	.232	.550	
H8-C-f6	1306	3A/a	T-136	.268	.00994	.769	.0758	.151	.00422	9,200	8,920		1.0210	4.997	4.92	5.46	.251	.521	
		3B	T-137	.270	.01014	.758	.0759	.162	.00446	9,390	9,030		1.0204	5.396	5.32	5.48	.253	.482	

TABLE A3.1 (CONT'D)

TABLE A3.1 (CONT'D)																				
Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number ^{/a}	U-235 Atom Depletion	Total Accumulated Fissions Relative to N ₀ ²⁸ (fission/atom)	Accumulated Fission Fraction				Burnup (MWD/MTU) ^{/b}			N _U ²⁸ N _O (atoms/atom)	N _{Pu} ²⁸ N _O (atoms/atom x 10 ³)	Pu/U Mass Ratio ^{/c} ($\frac{gm}{gm}$ x 10 ³)				Alpha-25 ^{/e}	NGD/MWD ^{/f}
						U-235	U-238	Pu-239	Pu-241	Isotopic	Cs-137	Sr-90	Isotopic	X-Ray	Isotopic	X-Ray				
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES																				
J3-C-a1	3477	3 <u>/g</u>	G	.159	.00555	.783	.0737	.142	.00203	5,130	5,680		1.0267	3.804	3.72	3.71	.305	.393		
			T	.191	.00674	.808	.0743	.116	.00173	6,220	5,680		1.0255	3.752	3.68	3.71	.252	.508		
			T-1 <u>/i</u>	.184	.00644	.803	.0742	.121	.00165	5,950	5,680		1.0258	3.782	3.70	3.71	.266	.484		
			N-1 <u>/i</u>	.177	.00622	.793	.0740	.131	.00193	5,760	5,680		1.0258	3.946	3.87	3.71	.277	.434		
			T-3 <u>/j</u>	.184	.00644	.803	.0742	.121	.00180	5,960	5,680		1.0258	3.786	3.71	3.71	.263	.481		
			N-2 <u>/j</u>	.156	.00526	.776	.0736	.148	.00220	4,870	5,680		1.0270	3.786	3.70	3.71	.360	.396		
			5	G	.159	.00552	.791	.0737	.134	.00184	5,100	5,590	6,590	1.0268	3.689	3.61	3.60	.296	.413	
J3-C-f6	3480	1	T-19	.100	.00330	.875	.0727	.0522	.00023	3,040	2,650		1.0310	1.740	1.70	1.72	.234	.588		
			T-20	.191	.00671	.815	.0743	.109	.00156	6,200	4,390		1.0257	3.635	3.56	3.48	.243	.523		
			T-21	.181	.00638	.816	.0741	.108	.00142	5,890	4,860		1.0262	3.468	3.40	3.58	.241	.521		
			T-22	.109	.00359	.861	.0728	.0655	.00039	3,310	3,230		1.0305	1.981	1.93	1.76	.252	.566		
J4-C-a1	305	3 <u>/g</u>	T	.146	.00512	.833	.0735	.0923	.000930	4,730	4,090	3,710	1.0280	2.876 <u>/m</u>	2.81	2.31	.222	.503		
			T	.273	.01042	.751	.0759	.168	.00510	9,650	9,090	8,380	1.0200	5.556 <u>/m</u>	5.48	5.26	.243	.466		
			G	.279	.01064	.764	.0760	.156	.00463	9,850	9,200	8,550	1.0199	5.401 <u>/m</u>	5.32	5.01	.223	.489		
			T	.282	.01074	.758	.0761	.161	.00468	9,940	9,200	8,590	1.0197	5.527 <u>/m</u>	5.45	5.01	.234	.485		
			T	.188	.00671	.827	.0742	.0978	.00133	6,200	5,230	4,080	1.0259	3.398 <u>/m</u>	3.33	3.23	.209	.543		
J4-C-f6	0918	3 <u>/g</u>	T-73	.125	.00426	.830	.0731	.0960	.000782	3,940	4,310		1.0291	2.659	2.60		.262	.471		
			G	.258	.00960	.746	.0756	.173	.00562	8,900	9,590		1.0209	5.469	5.39	5.21	.285	.455		
			T-5 <u>/j</u>	.253	.00947	.754	.0755	.166	.00436	8,770	9,590		1.0212	5.298	5.22	5.21	.261	.456		
			N-7 <u>/j</u>	.241	.00916	.729	.0753	.190	.00520	8,490	9,580		1.0212	5.624	5.54	5.21	.284	.390		
			G	.211	.00791	.730	.0747	.190	.00535	7,330	8,760	9,860	1.0229	5.150	5.06	4.72	.301	.368		
			T-6 <u>/j</u>	.257	.00973	.756	.0756	.164	.00419	9,000	8,770	9,660	1.0208	5.440	5.36	4.72	.245	.447		
			N-8 <u>/j</u>	.249	.00931	.750	.0754	.169	.00539	8,620	8,770	9,710	1.0214	5.248	5.17	4.72	.272	.456		
			T-74	.158	.00551	.821	.0737	.104	.00119	5,090	5,230		1.0274	3.134	3.07		.243	.502		
J5-C-a1	3422	1	T-69	.161	.00553	.825	.0737	.0999	.00108	5,110	4,350		1.0274	3.120	3.05	3.23	.255	.522		
			T-70	.289	.01110	.743	.0763	.175	.00561	10,280	10,240		1.0189	5.914	5.84	5.93	.249	.461		
			T-71	.286	.01088	.744	.0762	.174	.00568	10,080	10,000		1.0193	5.753	5.68	5.62	.261	.476		
			T-72	.181	.00632	.808	.0741	.117	.00163	5,840	5,320		1.0261	3.608	3.53	3.74	.264	.503		
K4-C-a1	2907	2 <u>/g</u>	T-32	.0777	.00248	.876	.0723	.0519	.000190	2,290	2,160		1.0321	1.489	1.45	1.57	.273	.548		
			T-33	.149	.00506	.845	.0735	.0810	.000690	4,670	4,160		1.0283	2.677	2.62	2.80	.243	.566		
			N-15-1 <u>/h</u>	.134	.00447	.833	.0733	.0933	.000806	4,130	4,000		1.0288	2.732	2.67	2.80	.282	.503		
			N-15-2	.134	.00436	.830	.0733	.0959	.000822	4,030	4,000		1.0290	2.716	2.65	2.80	.319	.520		
			G	.170	.00591	.835	.0739	.0904	.000985	5,460	4,820	4,930	1.0270	3.126	3.06	2.90	.228	.543		
			T-4 <u>/j</u>	.152	.00518	.823	.0736	.102	.00116	4,790	4,820	4,970	1.0278	3.078	3.01	2.90	.267	.498		
			N-9 <u>/j</u>	.131	.00440	.802	.0732	.123	.00128	4,060	4,810	5,050	1.0285	3.155	3.08	2.90	.327	.400		
			T-34	.153	.00522	.817	.0736	.108	.00123	4,820	4,860		1.0277	3.135	3.06	2.93	.280	.496		
			G	.149	.00511	.831	.0735	.0949	.000909	4,720	4,650	4,280	1.0279	2.982	2.92	2.88	.253	.504		
			T-24	.148	.00505	.828	.0735	.0970	.000995	4,660	4,840	4,290	1.0280	2.969	2.90	2.68	.265	.507		
			T-35	.0904	.00294	.872	.0725	.0552	.000238	2,710	2,520		1.0314	1.651	1.61	1.63	.257	.570		
			N-16-1 <u>/h</u>	.0703	.00215	.851	.0722	.0759	.000725	1,990	2,060		1.0322	1.684	1.64	1.63	.367	.417		
			N-16-2	.0794	.00248	.861	.0723	.0660	.000633	2,290	2,060		1.0319	1.694	1.65	1.63	.323	.493		

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TABLE A3.1 (CONT'D)

TABLE A3.1 (CONT'D)																		
Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number ^{/a}	U-235 Atom Depletion	Total Accumulated Fissions Relative to N ₀ ²⁸ (fission/atom)	Accumulated Fission Fraction (total fissions)				Burnup (MWD/MTU) ^{/b}			N _U N ₀ ²⁸ (atoms/atom)	N _{Pu} N ₀ ²⁸ (atoms/atom x 10 ³)	Pu/U Mass Ratio ^{/c}			
						U-235	U-238	Pu-239	Pu-241	Isotopic	Cs-137	Sr-90			Isotopic	X-Ray	Alpha-25 ^{/e}	NGD/MWD ^{/f}
PHASE I, ASYMPTOTIC NEUTRON SPECTRUM, CENTER SUBASSEMBLIES																		
K4-C-f6	2905	1	T-15	.0721	.00234	.887	.0722	.0405	.000228	2,160	1,900		1.0325	1.253	1.22	1.16	.239	.594
		3	T-16	.143	.00485	.847	.0734	.0793	.000651	4,480	4,500		1.0286	2.572	2.51	2.57	.242	.566
		5	T-17	.140	.00477	.849	.0734	.0771	.000651	4,400	3,670		1.0287	2.534	2.48	2.41	.236	.561
		6	T-23	.0743	.00235	.877	.0722	.0503		2,170	2,120		1.0322	1.449	1.41	1.42	.283	.542
K5-C-a1	0877	1	T-28	.111	.00382	.866	.0729	.0609	.000369	3,530	2,840		1.0303	1.942	1.89	2.00	.195	.566
		3	T-29	.216	.00782	.806	.0748	.117	.00204	7,230	6,450		1.0242	4.007	3.93	3.98	.218	.524
		5	T-30	.207	.00751	.806	.0746	.117	.00208	6,940	6,450		1.0246	3.912	3.84	3.93	.221	.517
		6	T-31	.126	.00437	.856	.0731	.0698	.000570	4,030	3,820		1.0294	2.210	2.16	2.04	.205	.568
K5-C-f6	011	1	G-18	.0914	.00310	.882	.0725	.0457	.000134	2,860	2,480		1.0313	1.615	1.57	1.53	.192	.562
		3	G-19	.170	.00600	.829	.0739	.0958	.00118	5,540	6,440		1.0268	3.206	3.14	3.29	.220	.524
		5	G-20	.170	.00601	.832	.0739	.0925	.00109	5,550	7,860		1.0268	3.157	3.09	3.16	.212	.531
		6	G-21	.0945	.00317	.866	.0726	.0607	.000357	2,930	3,000		1.0310	1.831	1.78	2.08	.225	.520
PHASE I, PERTURBED NEUTRON SPECTRUM, CORNER SUBASSEMBLIES																		
E5-NE-f1	2300	1	G-38	.255	.00941	.773	.0581	.165	.00426	8,720	9,490		1.0217	4.860	4.78	4.68	.247	.518
		3	G-39	.415	.0172	.699	.0619	.226	.0134	15,980	15,460		1.0113	7.307	7.27	7.45	.231	.516
		5	G-40	.368	.0149	.714	.0607	.215	.0104	13,830	13,380		1.0138	7.080	7.02	6.32	.234	.485
		6	G-41	.236	.00854	.803	.0578	.136	.00288	7,900	7,840		1.0232	4.251	4.18	3.75	.228	.550
E5-SW-a5	15025	1	T-46	.246	.00888	.790	.0580	.148	.00357	8,220	8,670		1.0227	4.405	4.33	4.23	.250	.557
		3	T-47	.436	.0182	.683	.0624	.239	.0157	16,910	17,680		1.0098	7.775	7.75	7.55	.250	.509
		5	T-48	.419	.0172	.698	.0619	.226	.0137	15,980	16,350		1.0113	7.348	7.31	7.31	.242	.520
		6	T-49	.270	.00996	.767	.0585	.169	.00491	9,230	9,850		1.0209	5.067	4.99	4.68	.256	.526
F4-NE-f1	20036	1	T-50	.257	.00924	.795	.0582	.144	.00357	8,560	8,820		1.0223	4.406	4.33	4.35	.248	.580
		3	T-51	.463	.0191	.691	.0631	.229	.0168	17,790	17,330		1.0092	7.383	7.36	7.70	.247	.556
		5	T-52	.468	.0194	.687	.0632	.232	.0175	18,070	18,340		1.0088	7.481	7.46	7.63	.248	.553
		6	T-53	.312	.0116	.764	.0594	.170	.00616	10,730	11,220		1.0191	5.258	5.19	5.29	.256	.577
F4-SW-a5	17383	1	T-75	.212	.00758	.804	.0573	.136	.00245	7,020	7,350		1.0244	4.009	3.93	3.96	.241	.532
		3	T-76	.373	.01509	.706	.0608	.222	.01112	14,010	14,190		1.0137	7.075	7.02	7.17	.248	.493
		5	T-77	.388	.01567	.712	.0612	.216	.01147	14,550	14,970		1.0132	6.929	6.88	7.07	.240	.519
		6	T-78	.253	.00925	.784	.0581	.154	.00377	8,560	8,640		1.0220	4.678	4.60	4.49	.241	.535
PHASE II, PERTURBED NEUTRON SPECTRUM, CORNER SUBASSEMBLIES																		
F5-NW-a1	8604	1	G-47	.324	.01254	.741	.0597	.192	.00704	11,620	11,760	13,280	1.0170	6.356	6.28	6.88	.242	.484
		3	G-48	.612	.03001	.580	.0673	.315	.03823	27,990	28,680	28,000	.9946	10.674	10.80	11.37	.254	.458
		5	G-49	.637	.03106	.586	.0680	.307	.03971	28,960	29,160	27,320	.9941	10.140	10.27	10.81	.249	.490
		6	G-50	.465	.01970	.682	.0632	.238	.01677	18,300	17,850	14,310	1.0081	7.900	7.88	8.27	.234	.521
F5-SE-f5	8717	1	T-123	.358	.01423	.716	.0605	.214	.01001	13,200	13,090	17,630	1.0147	6.904	6.84	7.29	.253	.487
		2	T-124	.607	.02918	.591	.0671	.306	.03609	27,200	28,630	25,620	.9960	10.215	10.32	10.24	.256	.479
		3	T-125	.649	.03253	.568	.0684	.321	.04262	30,350	31,740	30,040	.9920	10.738	10.90	11.28	.252	.468
		4	T-126	.646	.03263	.567	.0683	.323	.04222	30,440	32,340	28,910	.9916	11.005	11.17	11.56	.245	.456
		5	T-127	.623	.03039	.588	.0676	.307	.03717	28,330	28,700	28,030	.9945	10.417	10.55	11.00	.242	.475
		6	T-128	.439	.01822	.696	.0625	.227	.01426	16,920	16,960	17,590	1.0100	7.587	7.56	7.83	.234	.522

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TABLE A3.1 (CONT'D)

TABLE A3.1 (CONT'D)																		
Core Location	Fuel Rod Number	Axial Sample Zone	Sample ^{/a} Code Number	U-235 Atom Depletion	Total Accumulated Fissions Relative to N ²⁸ (fission/atom)	Accumulated Fission Fraction				Burnup (MWD/MTU) ^{/b}			N ^U N ²⁸ O (^{atoms} / _{atom})	N ^{Pu} N ²⁸ O (^{atoms} / _{atom} x 10 ³)	Pu/U Mass Ratio ^{/c}		Alpha-25 ^{/e} NGD/MWD ^{/f}	
						(total fissions)				Isotopic	Cs-137	Sr-90			(gm/gm x 10 ³)	Isotopic		X-Ray
						U-235	U-238	Pu-239	Pu-241									
PHASE I, PERTURBED NEUTRON SPECTRUM, CORNER SUBASSEMBLIES																		
G4-NW-a1	1XY39	1	T	.266	.00971	.801	.0584	.137	.00333	8,990	8,540	8,200	1.0218	4.442 ^{/m}	4.37		.218	.582
		3	T	.471	.0197	.691	.0633	.229	.0168	18,310	17,750	17,120	1.0085	7.530 ^{/m}	7.51		.231	.549
		5	T	.482	.0201	.694	.0636	.226	.0169	18,640	17,550	21,330	1.0082	7.533 ^{/m}	7.52		.235	.560
		6	T	.308	.0115	.761	.0593	.173	.00628	10,640	10,720	10,040	1.0191	5.330 ^{/m}	5.26		.255	.563
G4-SE-f5	12760	1	G-34	.228	.00805	.785	.0576	.154	.00352	7,460	8,210		1.0235	4.407	4.33	4.82	.282	.525
		3	G-35	.456	.01889	.687	.0629	.233	.01668	17,550	18,240		1.0093	7.601	7.58	7.49	.253	.538
		5	G-36	.474	.01982	.687	.0634	.232	.01752	18,410	18,370		1.0082	7.749	7.73	8.16	.240	.540
		6	G-37	.320	.01179	.774	.0596	.161	.00582	10,920	11,500		1.0189	5.205	5.14	5.06	.249	.592
H3-SE-f5	220	Comp. ^{/n}	G	.346	.0130	.778	.057	.158	.007	12,070			1.0175	5.391	5.32		.220	.605
H3-SE-f5	220	1	G	.203	.00710	.835	.0571	.106	.00169	6,560	6,450	6,650	1.0256	3.340	3.27	3.18	.221	.604
		2	G	.358	.01365	.771	.0604	.162	.00702	12,640	12,360	12,210	1.0170	5.292	5.23	5.33	.213	.615
		3	G	.392	.01532	.744	.0613	.185	.00950	14,200	14,330	13,190	1.0144	6.084	6.03	5.81	.225	.587
		4	G	.406	.01586	.747	.0616	.182	.00967	14,700	14,760	13,670	1.0138	6.128	6.08	6.18	.220	.596
		5 ^{/g}	G	.370	.01432	.744	.0608	.186	.00877	13,280	13,390	14,410	1.0155	6.023	5.96	5.80	.238	.574
		5 ^{/g}	T-7	.374	.01452	.744	.0608	.186	.00873	13,460	13,390	14,420	1.0152	6.106	6.05	5.80	.234	.571
		5 ^{/g}	N-3 ^{/l}	.367	.0142	.742	.0607	.188	.00916	13,140	13,390	14,440	1.0156	6.071	6.01	5.80	.244	.568
6	G	.254	.00915	.824	.0581	.115	.00241	8,460	7,630	9,080	1.0228	3.997	3.93	3.76	.200	.609		
H5-NW-a1	9709	1	T-54	.237	.00844	.806	.0578	.133	.00285	7,800	7,820		1.0234	4.131	4.06	4.07	.240	.572
		3	T-55	.450	.01851	.698	.0627	.224	.01525	17,190	17,400		1.0100	7.330	7.30	7.53	.241	.549
		5	T-56	.466	.01915	.701	.0632	.220	.01573	17,780	17,620		1.0093	7.269	7.24	7.44	.238	.566
		6	T-57	.326	.01213	.778	.0597	.156	.00555	11,230	11,200		1.0186	5.183	5.12	5.26	.232	.599
J4-NW-a1	11183	1	T	.203	.00718	.832	.0571	.109	.00174	6,630	6,060	4,730	1.0254	3.422 ^{/m}	3.35	2.97	.212	.588
		2	T	.364	.01405	.764	.0606	.168	.00729	13,010	12,280	10,820	1.0162	5.646 ^{/m}	5.59	5.37	.210	.592
		3	T	.406	.01598	.749	.0616	.180	.00928	14,810	13,940	13,080	1.0138	6.127 ^{/m}	6.08	6.56	.207	.592
		4	T	.406	.01602	.742	.0616	.187	.00985	14,850	14,590	13,750	1.0136	6.243 ^{/m}	6.20	6.06	.218	.586
		5									13,050	12,200						
		6									7,140	7,370						
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY																		
F4-SW-b4	17457	1	T-79	.194	.00694	.805	.0657	.127	.00218	6,420	6,580		1.0253	3.788	3.71	3.73	.240	.511
		3	T-80	.338	.0135	.713	.0688	.209	.00926	12,500	13,160		1.0157	6.662	6.60	6.66	.257	.477
		5	T-81	.348	.0139	.711	.0690	.211	.00962	12,880	13,400		1.0153	6.719	6.66	6.34	.257	.486
		6	T-82	.232	.00855	.785	.0665	.146	.00316	7,910	8,090		1.0230	4.459	4.38	3.81	.236	.511
F4-SW-d2	17460	1	T-83	.176	.00615	.784	.0740	.140	.00214	5,690	6,300		1.0259	3.912	3.83	3.86	.296	.440
		3	T-84	.333	.01339	.706	.0772	.208	.00883	12,420	12,660		1.0155	7.015	6.95	6.95	.254	.434
		5	T-85	.335	.01340	.704	.0773	.210	.00914	12,430	12,960		1.0155	6.960	6.89	6.68	.266	.445
		6	T-86	.227	.00837	.780	.0750	.142	.00303	7,740	7,750		1.0231	4.538	4.46	4.28	.240	.484

TABLE A3.1 (CONT'D)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	U-235 Atom Depletion	Total Accumulated Fissions Relative to ²³⁸ N ₀ (fission/atom)	Accumulated Fission Fraction (total fissions)				Burnup (MWD/MTU) ^{/b}			^N ₂₈ / ^N ₀	^N ₂₈ / ^N ₀	Pu/U Mass Ratio ^{/c}			
						U-235	U-238	Pu-239	Pu-241	Isotopic	Cs-137	Sr-90	(atoms/atom)	(atoms/atom x 10 ³)	(gm/gm x 10 ³)			
															Isotopic	X-Ray	Alpha-25 ^{/e}	NGD/MWD ^{/f}
PHASE II, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY																		
F5-NW-b2	8172	1	G-56	.297	.01145	.735	.0678	.190	.00683	10,620	11,220	10,920	1.0184	6.075	6.00	6.86	.257	.464
		3	G-57	.574	.02732	.604	.0748	.292	.02983	25,440	25,630	24,560	0.9977	10.480	10.57	11.67	.242	.451
		5	G-58	.571	.02598	.615	.0747	.277	.03322	24,190	26,370	24,390	0.9996	9.863	9.93	11.26	.273	.474
		6	G-59	.412	.01692	.692	.0705	.224	.01320	15,710	15,650	16,970	1.0115	7.420	7.38	8.00	.255	.508
F5-NW-d4	8163	1	G-52	.309	.01237	.723	.0767	.189	.01095	11,470	10,880	11,990	1.0174	6.202	6.13	6.78	.231	.456
		3	G-53	.534	.02528	.605	.0822	.287	.02598	23,530	23,190	20,500	0.9996	10.707	10.78	11.24	.246	.412
		5	G-54	.534	.02497	.601	.0822	.290	.02656	23,250	23,460	20,980	1.0001	10.505	10.57	11.00	.269	.425
		6	G-55	.383	.01555	.692	.0784	.218	.01150	14,430	14,140	15,760	1.0129	7.324	7.27	7.50	.270	.482
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY																		
H3-SE-b1	204	1	G	.126	.00439	.806	.0731	.119	.00122	4,060	4,550	4,520	1.0286	3.032 ^{/m}	2.96		.264	.379
		3	G	.285	.01111	.748	.0762	.171	.00551	10,290	9,960	9,670	1.0191	5.754 ^{/m}	5.68		.223	.462
		5 ^{/g}	T	.282	.01075	.748	.0761	.171	.00532	9,950	9,610	8,940	1.0195	5.685 ^{/m}	5.61		.250	.472
		5	G	.267	.01019	.741	.0758	.178	.00558	9,440	9,600	8,980	1.0201	5.624 ^{/m}	5.54		.262	.450
		6	G	.182	.00648	.821	.0741	.103	.00139	5,980	5,600	5,700	1.0261	3.425 ^{/m}	3.35		.222	.525
H3-SE-c2	116	1	G-26	.131	.00446	.830	.0732	.0956	.00084	4,120	4,960		1.0288	2.702	2.64	2.50	.261	.488
		3	G-27	.294	.01119	.747	.0764	.171	.00565	10,360	10,460		1.0191	5.613	5.54	5.58	.253	.501
		5 ^{/h}	G-28	.198	.00708	.810	.0744	.114	.00185	6,540	10,620		1.0252	3.751	3.68	3.36	.228	.518
		6	G-29	.173	.00612	.825	.0740	.100	.00129	5,660	5,600		1.0266	3.296	3.23	3.23	.223	.519
H3-SE-d3	103	1	G	.152	.00530	.820	.0693	.109	.00123	4,900	4,540	4,260	1.0275	3.170 ^{/m}	3.10		.249	.478
		3	G	.306	.01181	.743	.0723	.178	.00646	10,940	10,730	9,440	1.0182	5.880 ^{/m}	5.81		.242	.494
		5	G	.282	.01083	.738	.0718	.184	.00616	10,040	10,630	9,150	1.0194	5.730 ^{/m}	5.65		.257	.470
		6	G	.176	.00621	.810	.0697	.119	.00166	5,740	6,280	5,300	1.0262	3.560 ^{/m}	3.49		.251	.494
H3-SE-e4	111	1	G-30	.164	.00566	.841	.0651	.0930	.00108	5,230	5,660		1.0275	2.854	2.79	2.90	.228	.578
		3	G-31	.338	.01294	.752	.0687	.172	.00707	11,980	12,520		1.0173	5.631	5.57	5.50	.237	.564
		5	G-32	.323	.01227	.755	.0684	.170	.00656	11,370	13,270		1.0181	5.502	5.43	5.72	.242	.558
		6	G-33	.206	.00727	.813	.0659	.119	.00211	6,720	7,630		1.0251	3.612	3.54	4.01	.243	.571
PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY ADJACENT REFLECTOR																		
J3-NE-b5	22871	1	T-92	.0558	.00174	.888	.0720	.0398	.00016	1,610	1,660		1.0332	1.067	1.04	.89	.284	.557
		3	T-93	.115	.00379	.846	.0729	.0805	.00055	3,500	3,660		1.0300	2.279	2.22	1.54	.283	.527
		5	T-94	.115	.00378	.853	.0729	.0741	.00044	3,480	3,640		1.0301	2.171	2.12	2.17	.270	.550
		6	T-95	.0709	.00226	.887	.0722	.0408	.00013	2,080	1,960		1.0326	1.222	1.19	.93	.258	.607
J3-NE-d3	1425	1	T-109	.0523	.00171	.906	.0676	.0259	.00036	1,580	1,320		1.0336	.781	.76	.86	.200	.659
		3	T-110	.105	.00347	.875	.0684	.0560	.00026	3,200	3,050		1.0308	1.726	1.68	1.78	.232	.620
		5	T-111	.101	.00332	.877	.0684	.0542	.00029	3,060	2,950		1.0311	1.619	1.58	1.85	.236	.636
		6	T-112	.0554	.00178	.898	.0676	.0338	.00013	1,640	1,640		1.0333	.941	.91	1.14	.233	.609

TABLE A3.1 (CONT'D)

Core Location	Fuel Rod Number	Axial Sample Zone	Sample Code Number	U-235 Atom Depletion	Total Accumulated Fissions Relative to N ²⁸ _O (fission/atom)	Accumulated Fission Fraction ($\frac{\text{fissions}}{\text{total fissions}}$)				Burnup (MWD/MTU) ^{/b}			$\frac{N_U}{N_O^{28}}$ (atoms/atom)	$\frac{N_{Pu}}{N_O^{28}}$ (atoms/atom x 10 ³)	Pu/U Mass Ratio ^{/c} ($\frac{\text{gm}}{\text{gm}}$ x 10 ³)		Alpha-25 ^{/e}	NGD/MWD	/f
						U-235	U-238	Pu-239	Pu-241	Isotopic	Cs-137	Sr-90			Isotopic	X-Ray			
						PHASE I, INTERMEDIATE NEUTRON SPECTRUM, ON DIAGONAL OF CORNER SUBASSEMBLY ADJACENT REFLECTOR													
J3-NE-e2	31	3	T-130	.114	.00376	.890	.0642	.0453	.00021	3,460	2,980		1.0308	1.493	1.46	1.84	.218	.726	
		5	T-131	.0922	.00298	.892	.0639	.0436	.00018	2,750	2,540		1.0318	1.307	1.27	1.71	.235	.701	
J3-NE-f1	23018	1	T-96	.0463	.00139	.921	.0542	.0250	.00018	1,280	1,420		1.0340	.620	.60	.58	.288	.771	
		3	T-97	.106	.00344	.901	.0553	.0432	.00019	3,180	3,090		1.0312	1.342	1.31	1.50	.219	.748	
		5	T-98	.109	.00354	.903	.0553	.0414	.00017	3,270	3,040		1.0312	1.319	1.28	1.32	.219	.768	
		6	T-99	.0599	.00187	.914	.0545	.0312	.00038	1,720	1,720		1.0334	.749	.73	1.17	.251	.777	

- Notes: a) The letters T, G, and N indicate that the mass spectrometric analyses were performed by the TRACERLAB (TL), General Electric-Vallecitos (GE), and the New Brunswick (NB) analytical laboratories, respectively. The radiochemical analyses for Cs-137 and Sr-90, and the X-ray fluorescence analyses for Pu/U mass ratio were performed by the Technical Service Laboratories, Westinghouse Atomic Power Division.
- b) Units are megawatt days per metric tonne of initial uranium loading (MT = 10⁶ gm).
- c) All Phase I data are referenced to the Core I end-of-life(May 18,1962); all Phase II data are referenced to the Core II end-of-life (Sept. 2, 1963).
- d) The letters A and B denote the pellet samples immediately above and below the nominal zone 3 pellet, respectively.
- e) Alpha-25 is the ratio of total captures to total fissions in U-235.
- f) The parameter NGD/MWD is the net number of grams of fissile materials destroyed per megawatt day of energy release.
- g) Interlaboratory crosscheck sample.
- h) The numerals 1 and 2 in the NB sample code numbers indicate that the results are inferred from the first and second runs of a mass spectrometric analysis performed in duplicate.
- i) Original sample dilution.
- j) New dilution of original sample.
- k) Seven solutions of the monitor sample 0994-3 were submitted to TRACERLAB over the time interval Jan. 1964 to Nov. 1964; four solutions of the monitor were submitted to GE-Vallecitos over the time interval April 1964 to Oct. 1964.
- l) All mass spectrometric data for this sample are consistent. However, the relatively low burnup inferred from the data indicates a high probability that the sample solution was incorrectly labeled.
- m) Dilute acid sample. Samples submitted for mass spectrometric analysis prior to June 7, 1963 had nitric acid concentrations of about 0.01 M, and are referred to as dilute acid samples. All other samples, submitted after this date, were in approximately 2 M HNO₃, and are referred to as concentrated acid samples. The sample history of shipping, spiking, and data reporting dates can be found in the Yankee Core Evaluation quarterly progress reports.
- The values of Pu-239/U-238 atom ratio reported for the dilute acid samples have been adjusted, or corrected, with the use of plutonium data obtained for concentrated samples. The values of burnup and fuel characteristics listed for the dilute acid samples in this table were inferred from the mass spectrometric data, using the adjusted values of Pu-239/U-238 atom ratio.
- n) A fuel rod section of 1.9 inches (approximately 3 pellets), was cut from a position immediately above the nominal 1-pellet sample location in each of the six axial sample zones. The six sections were then dissolved and diluted to make one composite sample solution.

3. Isotopic Fractional Accumulated Fissions

This quantity can be used to determine the number of fissions which occurred in each of the four fissionable isotopes. Thus, for the example cited above for sample #G-24,

Accumulated fissions in U-235 per initial atom of U-238 =

$$0.01526 \frac{\text{fissions/cm}^3 \text{ of fuel}}{\text{initial U-238 atoms/cm}^3 \text{ of fresh fuel}} \times$$

$$0.690 \frac{\text{fissions in U-235/cm}^3 \text{ of fuel}}{\text{total fissions/cm}^3 \text{ of fuel}} = 0.0105$$

Accumulated fissions in U-235 per unit volume of fuel =

$$3.35 \times 10^{20} \frac{\text{total fissions}}{\text{cm}^3 \text{ of fuel}} \times 0.690 \frac{\text{fissions in U-235}}{\text{total fissions}} =$$

$$2.31 \times 10^{20} \text{ fissions in U-235/cm}^3 \text{ of fuel}$$

Similarly,

$$\frac{\text{fissions in U-238}}{\text{cm}^3 \text{ of fuel}} = 3.35 \times 10^{20} \times 0.0780 = 2.61 \times 10^{19}$$

$$\frac{\text{fissions in Pu-239}}{\text{cm}^3 \text{ of fuel}} = 3.35 \times 10^{20} \times 0.221 = 7.40 \times 10^{19}$$

$$\frac{\text{fissions in Pu-241}}{\text{cm}^3 \text{ of fuel}} = 3.35 \times 10^{20} \times 0.0110 = 3.68 \times 10^{18}$$

4. Fuel Burnup

For each sample, values of fuel burnup have been inferred from the mass spectrometric and Cs-137 data. For some samples, the value of burnup was also

inferred from the Sr-90 activity. For each pair or set of cross check samples, or those samples analyzed in duplicate, there generally is just one measurement of Cs-137 dpm/gmU; the inferred values of Cs-137 burnup listed for each such pair or set of samples are not always identical since the values of effective yield and Mev/fission used in the reduction of the radiochemical data are dependent upon the isotopic composition obtained from the mass spectrometric analysis.

5. Density of Uranium Atoms

The value of N^U/N_O^{28} , in conjunction with the measured uranium atom percent data given in Table A1.1, can be used to determine the uranium isotopic inventory of the fuel. For example, in sample #G-24, the uranium isotopic atom inventory, relative to the initial U-238 atom density in the fresh fuel, is:

$$\begin{aligned} N^{24}/N_O^{28} &= 0.000180 \frac{\text{atoms U-234/cm}^3 \text{ fuel}}{\text{total atoms U/cm}^3 \text{ fuel}} \times 1.0128 \frac{\text{atoms U/cm}^3 \text{ fuel}}{\text{initial atoms U-238/cm}^3 \text{ fresh fuel}} \\ &= 0.000182 \frac{\text{atoms U-234/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}} \end{aligned}$$

Similarly,

$$\begin{aligned} N^{25}/N_O^{28} &= 0.0222 \times 1.0128 = 0.0225 \frac{\text{atoms U-235/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}} \\ N^{26}/N_O^{28} &= 0.00267 \times 1.0128 = 0.00270 \frac{\text{atoms U-236/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}} \\ N^{28}/N_O^{28} &= 0.9749 \times 1.0128 = 0.9874 \frac{\text{atoms U-238/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}} \end{aligned}$$

To obtain the uranium isotopic atom inventory per unit volume of fuel:

$$\begin{aligned} N^{24} &= 0.000182 \times 0.02194 \times 10^{24} = 3.99 \times 10^{18} \text{ atoms U-234/cm}^3 \text{ fuel} \\ N^{25} &= 0.0225 \times 0.02194 \times 10^{24} = 4.94 \times 10^{20} \text{ atoms U-235/cm}^3 \text{ fuel} \end{aligned}$$

$$N^{26} = 0.00270 \times 0.02194 \times 10^{24} = 5.92 \times 10^{19} \text{ atoms U-236/cm}^3 \text{ fuel}$$

$$N^{28} = 0.9874 \times 0.02194 \times 10^{24} = 2.166 \times 10^{22} \text{ atoms U-238/cm}^3 \text{ fuel}$$

To obtain the uranium isotopic inventory in kg/MTU:

$$\begin{aligned} \frac{\text{kg U-234}}{\text{MTU}} &= 0.000180 \frac{\text{atoms U-234/cm}^3 \text{ fuel}}{\text{total atoms U/cm}^3 \text{ fuel}} \times 1.0128 \frac{\text{atoms U/cm}^3 \text{ fuel}}{\text{initial atoms U-238/cm}^3 \text{ fresh fuel}} \\ &\times \frac{1 \text{ gm. at. wt. U-234}}{0.6025 \times 10^{24} \text{ atoms U-234}} \times \frac{234.11 \text{ gm. U-234}}{1 \text{ gm. at. wt. U-234}} \\ &\times 0.9652 \frac{\text{initial atoms U-238}}{\text{initial atom U}} \times \frac{0.6025 \times 10^{24} \text{ initial atoms U}}{1 \text{ gm. at. wt. of U}} \\ &\times \frac{1 \text{ gm. at. wt. U}}{(0.002)(234.11) + (0.0344)(235.12) + (0.0002)(236.12) + (0.9652)(238.12) \text{ gm U}} \\ &\times \frac{1 \text{ kg U-234}}{10^3 \text{ gm U-234}} \times \frac{10^6 \text{ gm U}}{\text{MTU}} \\ &= 0.000180 \times 1.0128 \times \frac{234.11}{238.02} \times 0.9652 \times 10^3 \end{aligned}$$

$$\frac{\text{kg U-234}}{\text{MTU}} = 0.173$$

$$\frac{\text{kg U-235}}{\text{MTU}} = 0.0222 \times 1.0128 \times \frac{235.12}{238.02} \times 0.9652 \times 10^3 = 21.4$$

$$\frac{\text{kg U-236}}{\text{MTU}} = 0.00267 \times 1.0128 \times \frac{236.12}{238.02} \times 0.9652 \times 10^3 = 2.59$$

$$\frac{\text{kg U-238}}{\text{MTU}} = 0.9749 \times 1.0128 \times \frac{238.12}{238.02} \times 0.9652 \times 10^3 = 953.4$$

6. Density of Plutonium Atoms

The value of $N^{\text{Pu}}/N_{\text{O}}^{28}$, in conjunction with the measured plutonium atom percent data given in Table A1.1, can be used to determine the plutonium isotopic inventory of the fuel. Again, using the results obtained for sample #G-24, the plutonium isotopic inventory can be determined in a manner completely analogous to that used for uranium in the previous section:

$$N^{49}/N_{\text{O}}^{28} = 0.7857 \times 0.007805 = 0.00613 \frac{\text{atoms Pu-239/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}}$$

$$N^{40}/N_{\text{O}}^{28} = 0.1310 \times 0.007805 = 0.00102 \frac{\text{atoms Pu-240/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}}$$

$$N^{41}/N_{\text{O}}^{28} = 0.0756 \times 0.007805 = 0.000590 \frac{\text{atoms Pu-241/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}}$$

$$N^{42}/N_{\text{O}}^{28} = 0.00768 \times 0.007805 = 0.0000599 \frac{\text{atoms Pu-242/cm}^3 \text{ fuel}}{\text{initial atom U-238/cm}^3 \text{ fresh fuel}}$$

$$N^{49} = 0.00613 \times 0.02194 \times 10^{24} = 1.34 \times 10^{20} \text{ atoms Pu-239/cm}^3 \text{ fuel}$$

$$N^{40} = 0.00102 \times 0.02194 \times 10^{24} = 2.24 \times 10^{19} \text{ atoms Pu-240/cm}^3 \text{ fuel}$$

$$N^{41} = 0.000590 \times 0.02194 \times 10^{24} = 1.29 \times 10^{19} \text{ atoms Pu-241/cm}^3 \text{ fuel}$$

$$N^{42} = 0.0000599 \times 0.02194 \times 10^{24} = 1.31 \times 10^{18} \text{ atoms Pu-242/cm}^3 \text{ fuel}$$

$$\frac{\text{kg Pu-239}}{\text{MTU}} = 0.7857 \times 0.007805 \times \frac{239.13}{238.02} \times 0.9652 \times 10^3 = 5.95$$

$$\frac{\text{kg Pu-240}}{\text{MTU}} = 0.1310 \times 0.007805 \times \frac{240.13}{238.02} \times 0.9652 \times 10^3 = 0.996$$

$$\frac{\text{kg Pu-241}}{\text{MTU}} = 0.0756 \times 0.007805 \times \frac{241.13}{238.02} \times 0.9652 \times 10^3 = 0.577$$

$$\frac{\text{kg Pu-242}}{\text{MTU}} = 0.00768 \times 0.007805 \times \frac{242.13}{238.02} \times 0.9652 \times 10^3 = 0.0588$$

7. Pu/U Mass Ratio

The Pu/U mass ratio determined from the mass spectrometric data is the sum of the Pu-239, Pu-240, Pu-241, and Pu-242 mass divided by the sum of the U-234, U-235, U-236, and U-238 mass. The Pu/U mass ratio determined from x-ray fluorescence analysis is the total plutonium mass relative to the total uranium mass. All Phase I data are referenced to the Core I end-of-life (May 18, 1962); all Phase II data are referenced the Core II end-of-life (September 2, 1963).

8. Alpha-25

Alpha-25 is the effective capture-to-fission ratio in U-235:

$$\alpha^{25} = \frac{\text{total captures in U-235}}{\text{total fissions in U-235}}$$

For the example used above, G-24,

$$\begin{aligned} \text{Total captures in U-235} &= \text{U-236 at core end-of-life plus the U-236 loss} \\ &\quad \text{through neutron capture minus the U-236 initially} \\ &\quad \text{present in the fresh fuel.} \\ &= 5.92 \times 10^{19} \text{ atoms U-236/cm}^3 \text{ fuel} \\ &\quad + (5.92 \times 10^{19})(1.0 - 0.9511) \text{ atoms U-236 destroyed/} \\ &\quad \text{cm}^3 \text{ fuel} \\ &\quad - (0.0002/0.9652)(0.02194 \times 10^{24}) \text{ initial atoms U-236/} \\ &\quad \text{cm}^3 \text{ fuel} \\ &= (5.92 + 0.29 - 0.45) \times 10^{19} \\ &= 5.76 \times 10^{19} \text{ captures/cm}^3 \text{ fuel} \end{aligned}$$

$$\text{Total fissions in U-235} = 2.31 \times 10^{20} \text{ fissions/cm}^3 \text{ fuel}$$

$$\alpha^{25} = \frac{5.76 \times 10^{19}}{2.31 \times 10^{20}} = 0.249$$

9. Net Number of Grams of Fissile Material Destroyed Per Megawatt Day

The net number of grams of fissile material destroyed per megawatt day of energy release in the fuel is:

NGD/MWD $\equiv$ mass of U-235 destroyed less the sum of Pu-239 and Pu-241 mass remaining in the spent fuel, divided by the total energy released.

Thus, to achieve a burnup of 14,200 MWD/MTU in sample #G-24, it was necessary to destroy a net 0.425 grams of fissile materials for each megawatt day of energy released.

$$\begin{aligned} \text{NGD/MWD} &= (34.00 - 21.44) \frac{\text{kg U-235}}{\text{MTU}} - 5.95 \frac{\text{kg Pu-239}}{\text{MTU}} - 0.577 \frac{\text{kg Pu-241}}{\text{MTU}} \\ &\times \frac{10^3 \text{ gm fissile material}}{\text{kg fissile material}} \div 14,200 \text{ MWD/MTU} \end{aligned}$$

$$\text{NGD/MWD} = 0.425$$

The consistency in the results inferred from the mass spectrometric data can be demonstrated by performing a materials balance to account for the initial loading of fresh fuel. Using the spent fuel data obtained for sample #G-24, such a materials balance is summarized in Table A3.2.

TABLE A3.2

Materials Balance for a Yankee Spent Fuel Pellet
After 14,200 MWD/MTU Burnup

<u>Quantity</u>	<u>Atoms</u> <u>cm³ fuel</u> $\times 10^{-20}$	<u>Accountability</u>
U-234	0.0399	Present in spent fuel
U-235	4.94	Present in spent fuel
U-236	0.592	Present in spent fuel
U-238	216.6	Present in spent fuel
Pu-239	1.34	Present in spent fuel
Pu-240	0.224	Present in spent fuel
Pu-241	0.129	Present in spent fuel
Pu-242	0.0131	Present in spent fuel
U-235	2.31	Accumulated fissions
U-238	0.261	Accumulated fissions
Pu-239	0.740	Accumulated fissions
Pu-241	0.0368	Accumulated fissions
U-236 Loss	0.0289	Neutron capture
Pu-242 Loss	0.00079	Neutron capture
Pu-241 In-Core Decay	0.00468	Radioactive Decay
Total	227.3	

$$\text{Initial uranium loading} = (0.02194 \times 10^{24}) / 0.9652 = 227.3 \times 10^{20} \frac{\text{atoms}}{\text{cm}^3 \text{ fuel}}$$