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Measurements on Spent-Fuel Assemblies at Arkansas Nuclear One Using the Fork System

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Measurements On Spent-Fuel Assemblies at Arkansas Nuclear One Using the Fork System[#]

Final Report, January 1995

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ABSTRACT

The Fork measurement system has been used to examine spent-fuel assemblies at the two reactors of Arkansas Nuclear One, operated by Entergy Operations, Inc. The Unit 1 reactor is a Babcock and Wilcox (B&W) design, and the Unit 2 reactor is a Combustion Engineering (CE) design. The neutron and gamma-ray emissions from individual spent-fuel assemblies were measured in the storage pools by raising each assembly partway out of the storage rack and performing a measurement near the center of the assembly. The overall accuracy of the measurements after corrections is about 2%. Thirty-four assemblies were examined at Unit 1, and forty-one assemblies at Unit 2. The average deviation of the burnup measurements from the calibration was 3.0% at Unit 1 and 3.5% at Unit 2, indicating 2 to 3% random variation among the reactor records. There was no indication of clearly anomalous assemblies. Axial Scans of the variation in neutron and gamma ray emission were obtained by collecting data at several locations along the length of three assemblies at Unit 2. Two of these assemblies were non-standard in that each contained a small neutron source. The sources were detected by the axial scans. The test program was a cooperative effort involving Sandia National Laboratories, Los Alamos National Laboratory, Entergy Operations, Inc., the Electric Power Research Institute, and the Office of Civilian Radioactive Waste Management of the U.S. Department of Energy.

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INTRODUCTION

The nuclear properties of spent reactor fuel are determined by the initial composition of the fuel and the burnup that the fuel experienced in the reactor. The nuclear reactivity of a spent-fuel assembly and the criticality (the ability to sustain a fission chain reaction) of an array of spent-fuel assemblies can be calculated from the initial enrichment, burnup, and cooling time of the assemblies.¹ An average burnup value is assigned to each fuel assembly at the time of discharge from the reactor based on the operating history of the reactor and the distribution of the neutron flux as monitored by in-core measurements during operation. In this report, burnup will refer to the average burnup value assigned to a spent-fuel assembly. Burnup is commonly expressed in units of gigawatt days (GWd, a time integral of the thermal power) per metric ton of uranium metal originally contained in the assembly (MTU).

The purposes of the operation described here are to enlarge the database of measurements with the Fork detector system at operating nuclear utilities, to conduct axial profile measurements along the length of assemblies, to measure non-standard assemblies that might produce anomalous results, and to investigate the impact of differing assembly designs on the results of the measurements. A similar exercise was conducted in 1993 at Oconee Nuclear Station of Duke Power Company.²

Role of Measurements

A need for verification of assembly burnup arises from concerns of nuclear criticality safety in the design of storage and transport systems for spent reactor fuel. A verification measurement can provide a check of reactor records that could affect nuclear criticality safety. Studies have concluded that the utility-supplied data on burnup are of greater accuracy and reliability than could be provided by additional radiation measurements on spent-fuel assemblies.³ The primary reason for this conclusion is the nature of the measurements from which the burnup is derived. The thermal output of a reactor is measured very accurately (less than one percent uncertainty) using thermocouples and flow meters at the inlet and outlet of the cooling water circulating in the reactor. The time integral of the power is the energy (gigawatt days, GWd) produced by the reactor and is the basis of the burnup assignment to individual assemblies. In-core radiation measurements located throughout the reactor

are used to distribute the GWd to each assembly through a distribution function provided by the manufacturer of the reactor. The distribution function assigns a fraction of the total GWd to each assembly based on the in-core measurements and the history of the assembly. Since the sum of the GWd of all assemblies must equal the total GWd of the reactor over any time span, if the burnup for one assembly is "high," another must be "low." The distribution function could possibly generate errors that can be characterized as random, because of the "zero-sum" aspect of the errors. The most likely source of a possible systematic error, one that applies to all assemblies, is the integral of the power output of the reactor, since the burnup for each assembly is determined by multiplying the distribution fraction by that number. Since the integral of the power output of the reactor is measured very accurately, a relative burnup measurement among the assemblies can determine the extent of the random error among the assemblies and indicate any assemblies whose radiation output is inconsistent with the record for burnup. The Fork system is designed to determine the extent of the random variation among assembly burnups, and to identify any anomalous values.

Since assembly burnup is defined and measured only while the reactor is in operation, once an assembly has been removed from the reactor as spent fuel, its burnup can no longer be directly measured. Radiation measurements on the spent fuel can be used to verify the burnup assigned to the assembly if the emitted radiation can be correlated with some aspect of the burnup experienced by the assembly while it was in the operating reactor.

Radiation From Spent Fuel

In the application of nuclear criticality safety to the transport and storage of spent fuel from commercial nuclear reactors, the fuel assemblies of interest have been cooled for over five years, which greatly simplifies the analysis of the emitted radiation. For shorter cooling times, many isotopes are significant emitters, but most have decayed to insignificance after five years because of the predominance of short half-lives in the fission and activation products. After five years of cooling time, the predominant neutron emitter is curium-244, which is formed by successive neutron capture beginning with uranium-238. The production of curium-244 is found to increase with about the fourth power of the burnup. The neutron emission is therefore very sensitive to variations in burnup. Cesium-137, the major gamma emitter after five years of cooling, is produced as a fission product. Its production is essentially a linear function of burnup. The attenuation factor in the assembly is greater for gamma rays than it is for neutrons, so that neutrons emitted from the assembly can originate from rods deeper inside the assembly than can gamma rays. In the case of neutrons from curium-244 and gamma rays from cesium-137, the neutrons can sample about 80% of the rods of an assembly, while the gamma rays can sample the outer 30% of the rods.

Fork System

The Fork system measures the passive neutron and gamma-ray emissions from individual spent-fuel assemblies while the assembly is in the storage pool. The effectiveness of the Fork system for verification of reactor records is due to the sensitivity of the neutron yield to burnup, the self-calibration generated by a series of measurements, and the redundancy provided by three independent detection systems. The Fork system was designed at Los Alamos National Laboratory and has been used for many years by the International Atomic Energy Agency (IAEA) to verify reactor records for safeguard applications. The results of those measurements are summarized, and publications cited, in Reference 4.

The Fork detector and its associated electronics are shown in Figure 1-1. Figure 1-2 is a disassembled view of the detector head. Each of the two arms of the Fork detector contains two fission chambers (the outer steel cylinders in Figure 1-2) to measure the yield of neutrons, and one ion chamber (the inner brass cylinders shown between the fission chambers in Figure 1-2) to measure gross gamma-ray emission. One fission chamber (the epithermal detector) in each arm is embedded in a polyethylene cylinder that is surrounded by a thin sheet of cadmium which serves to absorb thermal energy neutrons. The other fission chamber, outside the cadmium cover, is sensitive to thermal neutrons and the boron content of the water in the spent-fuel pool. The polyethylene cylinders containing the epithermal detectors are inserted into the polyethylene outer cover shown in Figure 1-2. The epithermal detectors provide the primary data used in the Fork technique. In the original (IAEA) application, the thermal neutron detectors were used to check the variation of the boron content among the spent-fuel pools at different sites. In the present use, the thermal detectors serve as a backup measurement to the epithermal data. The system is diagrammed in an operational arrangement in Figure 1-3. The detector is moved in the storage pool to the location of the spent-fuel assembly that is to be examined. The assembly is raised until the measuring point is located at the detector head, the detector is moved into contact with the assembly, and the neutron and gamma-ray data are collected for approximately 60 seconds. A battery-powered electronics unit and microprocessor are used to supply all power to the detectors, collect and analyze the detector outputs, and perform necessary calculations and documentation.

Figure 1-1
Fork Detector and Control Electronics

Figure 1-2
Disassembled Fork Detector Head

Figure 1-3
Fork System Arrangement in Spent-Fuel Pool

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VERIFICATION TESTING AT UNIT 1

Unit 1

Arkansas Nuclear One is a two-unit generating site located at Russellville, Arkansas. Commercial operation at the site began in 1974. Unit 1 utilizes a Babcock and Wilcox 2568 MW (thermal) pressurized water reactor (PWR) currently fueled with 177 Mark B8 fuel assemblies. The assembly design is a 15 x 15 array that contains 208 fuel rods and 16 guide tubes. The cross section of the assembly is 8.54 X 8.54 inches, and overall length is 165.6 inches. The nominal uranium weight is 464 kilograms per assembly.

Procedure

The Fork detector was suspended from the fuel-handling bridge over the spent-fuel pool. The demineralized water in the pool contained approximately 2500 parts per million boron. The top of the storage rack is approximately 25 feet below the water level. The Fork measurements were performed underwater between the inner fuel mast of the fuel-handling bridge and the top of the storage rack. The detector head was fixed at a location just below the inner fuel mast approximately 14 inches above the top of the storage rack. The shielding provided by the pool water and top fittings of the assemblies was adequate to produce an acceptable background reading in the detectors, typically less than one percent of an assembly measurement. Each selected assembly was raised in its rack using the fuel-handling hoist until the detector was 1 foot above the center of the assembly. A steel band located at the center of each assembly precluded a measurement at that level. Other measurements have shown the neutron and gamma-ray yield to be essentially constant along the central section of similar PWR assemblies, extending several feet either side of the center. For the purpose of verifying the reactor records, a measurement at 1 foot above the center of the assembly (halfway between bands) is equivalent to a measurement at the center of the assembly. The detector was placed in contact with the assembly, and data were accumulated for approximately 60 seconds to ensure that more than 10,000 counts were obtained in the epithermal neutron detectors. The ion chamber (gamma) current reaches its maximum value in approximately 1 second. After the measurement the assembly was lowered back into its rest position in the rack. Background data (no raised assembly) were taken each time the location of the detector was changed appreciably, and were found to vary insignificantly.

Results and Analysis

Thirty-four assemblies were measured in one and a half working days of operation. The initial enrichment of the assemblies ranged from 2.016 to 3.209 weight percent uranium-235. The range in assembly average burnup was from 19.9 to 57.3 GWd/MTU. The cooling times varied from 6.1 to 17.6 years. Background counts were generally less than 1% of the signal from the assembly. Appendix A lists the data and analysis values for all assemblies.

The analysis of the Fork data makes use of the reactor records for cooling time, burnup, and initial enrichment in such a way that errors in any of these parameters increases the average deviation, and the data from each assembly is compared with the average calibration. The approach used in the analysis described here is to accumulate data from a number of assemblies and generate an internal calibration by comparing each assembly to the best derived fit to all the data. This self-calibration eliminates the uncertainties and complications that are introduced by external calibration techniques, while retaining the sensitivity to detect measurements that are inconsistent with the reactor records.

To correct the observed data for the variation in cooling times among the assemblies, the neutron data (after background subtraction) were extrapolated back to the date of discharge of each assembly using an exponential factor of half-life equal to 18 years, the half-life of the principal neutron emitter, curium-244. A factor to adjust the observed count rates for the variation in initial enrichment among the assemblies was calculated using the reactor records for the initial enrichment and burnup for each assembly as described in Appendix B. This correction is required because the curium-244, which produces the neutrons, is produced by activation of uranium-238 rather than by fission reactions. The enrichment correction factor is normalized (=1) for an arbitrarily chosen enrichment of 3.0 weight percent uranium-235. For thirty-one assemblies, the correction factor varies from 0.8 to 1.1. For the three assemblies with the lowest enrichment (2.06 weight percent), the correction factor is 0.35.

The neutron count rates in the epithermal neutron detector (corrected by the cooling time and enrichment corrections described above) are shown in Figure 2-1, a log-log plot of neutron signal versus burnup (reactor record) for each assembly. The calibration curve shown in Figure 2-1 is the power law best fit (least squares) to the data, and is given by

$$N = C \cdot B^{3.83} \quad (\text{eq. 1})$$

where N is the neutron count rate in counts per second, B is the burnup in GWd/MTU, and C is a fitted constant whose value is 0.00100. The neutron signal is proportional to the 3.83 power of the burnup. This value closely matches the value observed in earlier operations on B&W assemblies with the Fork system.² The observed average absolute

deviation in burnup from the calibration curve shown in Figure 2-1 is 3.0%. The overall statistical uncertainty of the neutron measurement after corrections is about 2%. Combining the variations by the square root of the sum of the squares indicates a random variation in the records of about 3%. The maximum deviation observed for a single assembly was 8.6%.

Figure 2-1
Neutron Data and Calibration

The thermal neutron data is used to back up the epithermal data analyzed above. When the thermal neutron data are analyzed in the same manner as the epithermal data, the average deviation and the functional dependence on the burnup are essentially unchanged. The results for Unit 1 have been reported in an earlier publication.⁵

Figure 2-2 is a plot of the gamma-ray signal for each assembly divided by its burnup versus the cooling time. If the gamma-ray signal were due solely to fission products (like cesium-137), there ideally would be a single value for each cooling time. The average deviation from the mean value of the data at each cooling time is approximately 5%. Since the neutron data is a far more sensitive indicator of variations in burnup, the gamma-ray data is used only as a general confirmation of cooling time and burnup. The batch discharge of spent-fuel assemblies is evident in this display from the clustering of data around certain cooling times.

Figure 2-2
Gamma-Ray Data

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VERIFICATION TESTING AT UNIT 2

Unit 2

Unit 2 of Arkansas Nuclear One utilizes a Combustion Engineering pressurized water reactor rated at 2815 MW (thermal) that is currently fueled with 177 fuel assemblies. The assembly design is a 16 x 16 array that contains 5 guide tubes. The assembly cross section is slightly less than 8 X 8 inches, and overall length is approximately 177 inches. The active portion of each fuel rod is 150 inches overall, and contains 2114 grams of uranium metal. An additional feature of the assemblies of Unit 2 is the use of rods of two different initial enrichments in assemblies received after 1983. Eight rods of slightly lower enrichment surrounded each of the five guide tubes, and three such rods were located at each corner of the assembly. For the analysis of this data, the average enrichment for each assembly was used.

Procedure

The measurement procedure for Unit 2 was essentially the same as that for Unit 1 described on page 2-1, except for the differences described here. The fuel-handling bridge of Unit 2 does not use an inner fuel mast, so that the Fork detector had unobstructed access to the raised fuel assembly. The detector head was located approximately one foot above the storage rack. For the single point measurements, each assembly was raised until the detector head was located at the center of the assembly, about halfway between bands five and six counting from the top. There was no band at the center level of the Unit 2 assemblies that might interfere with the measurements. For the axial scans, data were taken halfway between successive bands on the assembly by raising the assembly and visually estimating the halfway point. The bands were separated by about 15 inches. The depth tape readings (ZZ tape) were used to specify the data and to locate data points closer together than 15 inches.

Results and Analysis

Measurements were performed on thirty-nine standard assemblies that had cooled longer than five years, and two non-standard assemblies of shorter cooling time that contained small neutron sources (plutonium-beryllium). Axial scans were performed on one of the standard assemblies and on both of the non-standard assemblies. The non-standard assemblies were not included in the burnup curve fitting analysis. The average initial enrichment of the assemblies ranged from 1.9 to 3.9 weight percent uranium-235. The range in assembly average burnup was 12.3 to 50.7 GWd/MTU. The cooling times varied from 3.8 to 13.7 years.

The analysis of the data followed the approach described for Unit 1 on page 2-2. The enrichment correction factor varied from 0.29 to 1.60.

The neutron count rates in the epithermal neutron detector (corrected by the cooling time and enrichment corrections) are shown in Figure 3-1, a log-log plot of neutron signal versus burnup (reactor record) for each assembly. The calibration curve shown in Figure 3-1 is the power law best fit (least squares) to the data, and is given by

$$N = C \cdot B^{4.35} \quad (\text{eq. 2})$$

where N is the neutron count rate in counts per second, B is the burnup in GWd/MTU, and C is a fitted constant whose value is 0.000145. The neutron signal is proportional to the 4.35 power of the burnup, which differs from the value of about 3.8 that has been observed for the B&W assembly design at several reactors. The observed average absolute deviation in burnup from the calibration curve shown in Figure 3-1 is 3.5%. The overall statistical uncertainty of the neutron measurement after corrections is about 2%. Combining the variations by the square root of the sum of the squares indicates a random variation in the records of about 3%. The maximum deviation observed for a single assembly was 8.6%.

Figure 3-2 is a plot of the gamma-ray signal for each assembly divided by its burnup versus the cooling time. If the gamma-ray signal were due solely to fission products (like cesium-137), there ideally would be a single value for each cooling time. The average deviation from the mean value of the data at each cooling time is approximately 5%. Since the neutron data is a far more sensitive indicator of variations in burnup, the gamma-ray data is used only as a general confirmation of cooling time and burnup.

Axial Scans

The three axial scans are shown in Figures 3-3, 3-4, and 3-5, in which the observed neutron and gamma-ray signals (without any corrections) are plotted as a function of

Figure 3-1
Neutron Count Rates in the Epithermal Neutron Detector

Figure 3-2
Gamma Signal Divided by Burnup vs. Cooling Time

Figure 3-3
Axial Scan of Standard Assembly

Figure 3-4
Axial Scan of Assembly Containing Neutron Source

Figure 3-5
Axial Scan of Assembly Containing Neutron Source

location along the assembly. The x-axis is the height reading of the fuel-handling bridge depth gauge in inches, and provides an approximate location. The center of the assemblies is located around 200 inches, and the top end of the active portion of the rod is located at about 280 inches. Figure 3-3 is the standard assembly, and exhibits the expected flat profile near the center of the assembly, with decreasing signals near the ends of the assemblies due to the reduced burnup. The nominal spatial resolution of the uncollimated detectors is about 15 inches on the axis of the assembly, so that finer detail of the axial variation of burnup is not resolved. Figures 3-4 and 3-5 show similar measurements for the two non-standard assemblies that incorporated the neutron sources. The sources were detected by a rise in signal of about 25 to 40% near the mid-point of the assemblies for the two neutron detectors. The neutron sources do not produce gamma rays, so the gamma-ray scan did not detect the sources.

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4

CONCLUSIONS

The Fork measurements correlated well with the records of both reactors at Arkansas Nuclear One. The correlation of the reactor records with the emitted neutrons indicated random variations of less than 3% in the records of assembly burnup. No obvious anomalous assemblies were detected, and the maximum deviation for a single assembly was less than 10%. The Fork system proved to be compatible with utility operations and equipment at both reactors. The effect of the different assembly designs in the two reactors is shown by the functional dependence of the neutron emission on burnup. For the Unit 1 (B&W) assemblies the neutron signal increased as the 3.83 power of the burnup, in agreement with earlier measurements on the B&W assemblies at Oconee Nuclear Station.² For the CE assemblies of Unit 2 the neutron signal increased as the 4.35 power of the burnup. This result suggests that the burnup dependence of the neutron signal may be specific to each assembly design. A variation in the burnup exponent has been noted in IAEA measurements with the Fork system.⁴ This variation is probably due to the varying reactor operating parameters among the different reactor designs. The axial scan measurements on the three assemblies indicate the resolution that can be obtained with uncollimated detectors, about 15 inches along the axis of the assembly. This resolution is adequate to locate the neutron sources in the two non-standard assemblies, which produced a rise in the neutron count rate of 25 to 40%. The two non-standard assemblies had been cooling for about 2 years. The relative increase in neutron count rate would be expected to increase with increased cooling time, since the curium-244 signal will decay with an 18-year half-life, while the neutron output of the plutonium-beryllium sources will remain essentially constant. This effect may account for the anomalous results observed in two assemblies at Oconee Nuclear Station.² Those two assemblies had been cooling for about 15 years, and are known to have contained neutron sources at one time.

Acknowledgments

The authors appreciate the essential help of the staff at Arkansas Nuclear One in the training, planning, and execution of this operation. Sandia National Laboratories managed the Fork application project with the sponsorship of Sandia's Laboratory Directed Research and Development Program. Los Alamos National Laboratory made available a Fork system and the experienced personnel to operate it. The Electric Power

Research Institute sponsored the proposal to the utility and supported the measurement campaign. Entergy Operations, Inc., made its facilities and personnel available in support of the measurements. The work was supported in part by the Office of Civilian Radioactive Waste Management of the U.S. Department of Energy.

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APPENDIX A: ASSEMBLY AND FORK DATA

Unit 1 data are on page A-2; Unit 2 data are on page A-3.

The first four columns were obtained from the reactor records for the assemblies tested. Column ID is the assembly identification. Column E_i is the initial enrichment (for Unit 2, the average enrichment). Column B is the average burnup in gigawatt days per metric ton of uranium. Column T is the time interval between discharge from the reactor and the date of the measurement in days. Column n is the observed epithermal neutron count rate in counts per second, with the background subtracted. Column n_{th} is the observed thermal neutron count rate in counts per second, with the background subtracted. Column E_i factor is the correction factor for initial enrichment described in Appendix B. Column N is the epithermal neutron count rate of column n multiplied by the factor of column E_i factor, and extrapolated to the date of discharge (by $e^{\lambda T}$, $\lambda = 0.0001048 \text{ d}^{-1}$, $T = \text{data of column T}$) using the 18-year half-life of curium-244. Column N contains the data plotted in Figure 2-1 (for Unit 2, Figure 3-1) as a function of the burnup, B. Column B_{cal} is the burnup value determined from the calibration line at the neutron count rate for that assembly. The calibration line is described in the text, and is derived from the best power law fit (least squares) to the data of column N. Column $\Delta B/B$ is the absolute deviation (in percent) of the burnup determined from the calibration (column B_{cal}) and the reactor record burnup (column B). Column G is the observed gamma (ion chamber) signal in milliamperes. Column G/B is the gamma signal divided by the burnup of column B, and is plotted in Figure 2-2 (Figure 3-2 for Unit 2).

Unit 1											
ID	E _i (wt%)	B (GWD/MTU)	T (d)	n (cps)	n _{th} (cps)	E _i factor	N (cps)	B _{cal} (GWD/MTU)	ΔB/B (%)	G (ma)	G/B
NJO23P	2.96	57.32	2210	3292.32	2604.11	0.98	4072	53.10	7.94	625.41	10.91
NJO35S	3.21	36.67	2210	750.18	578.45	1.11	1053	37.31	1.71	387.29	10.56
NJO0WS	3.01	31.90	4330	412.61	337.85	1.01	654.2	32.95	3.20	216.88	6.80
NJO22J	2.96	23.88	3626	122.64	100.47	0.97	175.0	23.36	2.14	158.57	6.64
NJO35R	3.21	36.68	2210	733.12	584.43	1.11	1029	37.08	1.09	383.54	10.46
NJO34Q	3.20	42.28	2210	1218.33	966.33	1.10	1690	42.21	0.159	492.91	11.66
NJO0XR	3.01	26.67	4330	230.89	178.00	1.00	365.1	28.30	5.76	183.07	6.39
NJO21Y	2.96	26.76	2934	210.66	205.71	0.97	279.1	26.39	1.42	193.31	7.22
NJO234	2.95	37.25	3526	624.01	691.35	0.97	1159	38.26	2.64	331.04	8.89
NJO23S	2.96	47.55	2934	1683.04	1371.11	0.98	2239	45.43	4.66	412.29	8.67
NJO23T	2.96	45.01	2934	1739.30	1341.18	0.98	2314	45.82	1.77	416.04	9.24
NJO349	3.21	28.39	2210	230.31	200.69	1.13	327.6	27.51	3.20	259.68	9.15
NJO35H	3.12	30.78	2934	348.33	270.15	1.12	530.2	31.20	1.33	271.78	8.83
NJO23V	2.96	44.82	2934	1725.56	1331.26	0.98	2298	45.74	2.01	411.66	9.19
NJO233	2.95	37.46	2934	893.91	648.20	0.97	1182	38.45	2.59	330.41	8.82
NJO23W	2.96	44.96	2935	1736.24	1340.97	0.98	2310	45.80	1.84	416.88	9.27
NJO22X	2.95	35.28	2935	582.17	517.70	0.97	770.7	34.39	2.58	303.95	8.61
NJO357	3.20	32.05	2935	455.22	371.22	1.12	690.6	33.42	1.41	292.17	8.87
NJO35G	3.20	30.78	2935	345.37	261.00	1.12	525.8	31.13	1.11	270.77	8.80
NJO34E	3.21	40.10	2211	1065.40	847.60	1.10	1479	40.77	1.65	471.88	11.77
NJO34F	3.21	40.18	2211	1006.60	854.80	1.10	1398	40.17	0.0194	471.25	11.73
NJO23Q	2.96	47.63	2935	1698.43	1357.86	0.98	2260	45.54	4.59	400.00	8.40
NJ1B14	2.72	23.00	6071	128.53	109.52	0.80	195.4	24.04	4.33	130.27	5.66
NJ1A29	2.06	19.91	6442	112.83	93.97	0.35	78.07	18.92	5.22	116.31	5.84
NJO0WY	3.01	25.00	5006	150.65	118.95	1.01	256.7	25.82	3.16	153.53	6.14
NJ1B18	2.72	28.34	6071	225.53	175.87	0.82	348.4	27.96	1.37	150.43	6.31
NJ1A35	2.07	19.90	5650	121.52	87.42	0.36	78.03	18.92	5.18	115.53	6.81
NJ1C28	3.05	32.60	5650	287.69	251.83	1.03	534.4	31.26	4.29	174.00	5.34
NJ1B46	2.73	28.29	6071	239.83	194.57	0.82	371.3	28.43	0.479	153.53	6.43
NJ1C01	3.05	28.88	5650	162.40	151.98	1.03	302.6	26.95	7.17	150.12	6.20
NJ1C14	3.04	30.87	5650	258.79	199.00	1.02	479.4	30.38	1.60	170.90	5.54
NJODX4	3.00	24.96	5006	149.28	129.12	1.00	253.1	25.72	2.96	155.70	6.24
NJO22C	2.95	35.33	2935	787.90	679.90	0.97	1043	37.22	5.08	322.80	8.32
NJ1A39	2.07	19.90	5650	182.20	93.50	0.36	117.0	21.03	5.37	119.00	6.04

Unit 2

ID	Ei (wt%)	B (GWd/MTU)	T (d)	n (cps)	nth (cps)	Ei factor	N (cps)	Bcal (GWd/MTU)	ΔB/B (%)	G (ma)	G/B
A002	1.93	12.31	5002	18.00	7.340	0.2915	8.863	12.56	1.96	74.00	6.01
A027	1.93	12.35	5002	16.48	7.360	0.2922	8.134	12.31	0.319	68.43	5.54
A009	1.93	14.16	5002	29.02	12.08	0.2892	14.18	13.99	1.25	82.40	5.82
A021	1.93	18.98	4091	135.2	55.59	0.2924	60.69	19.53	2.82	126.9	6.69
A026	1.93	19.15	4089	132.4	56.80	0.2879	58.51	19.37	1.12	126.0	6.58
B005	2.26	22.44	4492	116.1	49.66	0.4745	88.21	21.28	5.44	124.9	5.57
B006	2.26	22.89	4492	125.9	52.28	0.4772	96.2	21.71	5.44	129.1	5.64
BT05	2.26	26.13	4491	206.0	87.00	0.4861	160.3	24.41	7.04	147.0	5.63
B030	2.26	25.63	4492	194.7	85.48	0.4841	150.9	24.07	6.46	141.4	5.52
A028	1.93	24.22	3099	279.5	118.2	0.3018	116.7	22.70	6.72	175.6	7.25
C411	2.94	25.39	4090	173.5	72.91	0.9622	256.3	27.19	6.61	157.0	6.18
C413	2.94	27.26	4091	235.5	95.40	0.9615	347.6	29.16	6.51	173.1	6.35
C212	2.94	30.07	4091	289.5	121.6	0.9613	427.3	30.57	1.65	185.6	6.17
D113	2.97	32.82	4091	491.9	204.0	0.9844	743.4	34.72	5.47	241.0	7.34
D022	3.39	28.88	3554	273.7	118.2	1.239	492.2	31.58	8.56	212.2	7.35
D026	3.39	29.31	3554	273.2	117.7	1.238	490.9	31.56	7.14	212.0	7.23
D006	3.38	32.21	3554	374.8	162.1	1.229	668.5	33.88	4.94	230.8	7.17
E110	2.75	34.90	3100	658.7	272.3	0.8495	774.4	35.05	0.417	279.9	8.02
E021	3.33	31.23	3099	318.9	137.9	1.198	528.6	32.10	2.73	245.2	7.85
E030	3.33	30.97	3099	310.3	137.3	1.198	514.4	31.90	2.93	240.3	7.76
E020	3.33	33.37	3099	408.3	176.5	1.193	674	33.95	1.70	264.0	7.91
E040	3.33	35.48	3099	499.3	209.3	1.186	819.4	35.50	0.0680	276.6	7.80
E006	3.33	35.56	3099	542.9	227.6	1.184	889.4	36.18	1.71	288.5	8.11
F102	3.27	43.93	2490	1244	526.4	1.130	1826	42.67	2.94	436.0	9.92
F112	3.27	43.28	2490	1211	503.0	1.134	1783	42.44	1.98	431.0	9.96
F033	3.87	40.18	2490	710.4	294.9	1.525	1406	40.19	0.0328	393.5	9.79
F012	3.88	42.33	2490	933.1	380.6	1.525	1847	42.79	1.08	438.5	10.4
F014	3.88	43.90	2491	1023	430.0	1.511	2007	43.61	0.659	455.0	10.4
G311	3.37	43.84	1898	1069	451.0	1.178	1536	41.02	6.88	523.0	11.9
G310	3.37	39.65	1899	835.2	364.4	1.839	1874	42.93	7.65	447.3	11.3
G007	3.90	37.01	1898	542.0	232.1	1.605	1061	37.68	1.77	428.0	11.6
G119	3.90	40.26	1898	671.0	278.0	1.551	1270	39.26	2.54	454.0	11.3
G120	3.90	40.48	1899	718.0	297.0	1.547	1355	39.85	1.57	489.0	12.1
G103	3.90	45.75	1898	1097	452.0	1.520	2034	43.75	4.57	584.0	12.8
G104	3.90	45.79	1898	1079	472.0	1.520	2001	43.58	5.06	589.0	12.9
X003	3.90	37.51	1898	528.0	216.0	1.598	1029	37.41	0.256	430.0	11.5
H106	3.89	37.32	1898	447.0	191.0	1.588	866.1	35.96	3.79	440.0	11.8
H107	3.89	37.32	1898	454.0	189.0	1.589	880.2	36.09	3.40	443.0	11.9
F025	3.88	50.66	1383	1914	785.0	1.460	3230	48.65	4.13	834.0	16.5

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APPENDIX B: ENRICHMENT CORRECTION FACTOR

In the analysis of the neutron data, the assumption is made that the neutron source is the curium-244 in the assemblies. To adjust for the decay of curium-244 after discharge from the reactor, the observed neutron counting rates are extrapolated to the date of discharge using the 18-year half-life of curium-244, as described in Appendix A. An additional correction factor is needed because the neutron emission rate at a given burnup is dependent upon the initial enrichment of the assemblies. This is due to the fact that the production of curium is dependent upon the neutron flux in the reactor, rather than solely on the fission rate. The relationship between the production of curium and the burnup is dependent upon the initial enrichment of the assemblies and the burnup. To adjust the observed neutron data for this dependence, an enrichment correction factor is applied to the extrapolated neutron count rates. The factor is determined from calculations of the production of curium-244 corresponding to the initial enrichment and burnup of the assembly. Curium production calculations were obtained using a combination of the EPRI-CELL⁶ and EPRI-CINDER⁷ codes. The calculations were validated by comparison to destructive chemical analysis of spent fuel from the H. B. Robinson reactor.⁸ The curium-244 production calculations that were used in determining the initial enrichment correction are plotted in Figure B-1. The relative production rate for curium-244 is shown as a function of burnup for a family of curves of varying initial enrichments. Additional values were generated to cover the assemblies with lower enrichment. Since the neutron data for each assembly are multiplied by a correction factor, only relative values are required for the correction factor. For convenience the factors are normalized (=1) at an enrichment of 3.0 weight percent uranium-235. The correction factor for an assembly of a given burnup, taken from the reactor records, is defined as the ratio of the curium production at an enrichment of 3.0 weight percent divided by the curium production at the initial enrichment of the assembly. Values between the curves are interpolated using standard Lagrangian routines. The enrichment correction factors are listed for each of the assemblies in Appendix A, column **E_i factor**.

Figure B-1
Relative Cm-244 Production

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