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Burnup Verification Measurements on Spent-Fuel Assemblies at Oconee Nuclear Station

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Albuquerque, New Mexico
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REPORT SUMMARY

Burnup Verification Measurements on Spent-Fuel Assemblies at Oconee Nuclear Station

The application of burnup credit to the design of spent-fuel casks results in significantly reduced costs and risks in the transport and storage of spent-fuel assemblies. A measurement system to verify reactor records of spent-fuel burnup can permit the burnup credit savings to be realized. This report describes the demonstration of a practical, accurate method of verifying reactor records for the exposure of spent fuel.

INTEREST CATEGORIES

Light water reactor fuel
Radioactive waste management

KEYWORDS

Spent fuels
Waste transport
Criticality
Burnup

BACKGROUND NRC regulations require a substantial margin of safety to ensure that spent nuclear fuel cannot reach criticality (support a sustained nuclear reaction) as a result of unforeseen accident or abnormal shipping conditions. A practical measurement can permit use of realistic properties instead of conservative fresh-fuel properties. Storage, transportation, and disposal designs based on realistic spent-fuel composition (burnup credit calculations) can result in significantly more-efficient arrays of assemblies, reduce the need for expensive neutron absorbers, and decrease the risk in transporting a given quantity of fuel by permitting higher payload in each shipment and fewer total shipments. EPRI cosponsored this work with Sandia National Laboratories and Los Alamos National Laboratory to perform burnup verification measurements applying the fork detector system, used by the International Atomic Energy Agency to verify reactor records for safeguard applications. Duke Power Company's Oconee Nuclear Station served as host utility.

OBJECTIVES To establish a database using the fork detector system at an operating nuclear utility; to determine measurement compatibility with utility operating procedures; to develop an operational plan for implementing verification measurements with utility input.

APPROACH Investigators used the fork detector system to examine spent-fuel assemblies. First, they measured neutron and gamma-ray emissions from individual spent-fuel assemblies in the storage pool. Next, they performed tests that demonstrated the ability of the system to verify reactor records for burnup and cooling times and detect deviations from those records. Finally, they examined 93 assemblies, measuring burnup variation in two assemblies.

RESULTS The fork detector system measures the passive neutron and gamma-ray emissions from individual spent-fuel assemblies while in the storage pool. After five years of decay, the predominant neutron emitter in spent fuel is curium-244, formed by successive neutron capture beginning with uranium-238. The major gamma emitter after several years of cooling is cesium-137, produced as a fission product. The shorter-lived isotopes of curium and cesium are activation products, which are insignificant after a few years of cooling.

In testing, the fork detector system measurements correlated well with the Oconee Nuclear Station records. The average deviation of the reactor burnup records from

the calibration was 10% without corrections for initial enrichment and 2.2% with corrections. The derived calibration indicated that the neutron signal was proportional to the burnup raised to the 3.81 power. The gamma-ray signals were also in general agreement (15%) with the burnup records. Two of the 93 assemblies measured proved anomalous, producing much higher neutron signals than the burnup would explain. In a verification campaign, these two assemblies would require further study or be excluded from the acceptable fuel for a burnup credit cask.

EPRI PERSPECTIVE The fork detector system performed quite well and proved relatively easy to set up and operate. It could provide an acceptable means for verifying burnup of fuel assemblies before loading into a burnup credit cask or canister. Detector system measurements should be used to screen for gross errors in reactor records, such as inadvertent assignment of the burnup of one assembly to another. Ultimate qualification of a fuel assembly for loading should be based on the verified reactor records for burnup, since the records are likely to have less uncertainty in isotopic composition.

PROJECT

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Project Manager: R. F. Williams

Nuclear Power Division

Contractors: Sandia National Laboratories; Los Alamos National Laboratory

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ABSTRACT

The FORK measurement system has been used to examine spent fuel assemblies at the Oconee Nuclear Station of Duke Power Company. The neutron and gamma-ray emissions from individual spent fuel assemblies were measured in the storage pool after the assemblies were partially raised out of the storage rack. The tests were designed to demonstrate the ability of the FORK system to verify reactor records for burnup and cooling time, to detect deviations from those records, and to develop procedures for the use of the system that are compatible with utility operations. Ninety-three assemblies were examined in 3 1/2 days of operation. The variation in burnup along the length of the assembly was measured for two assemblies. The FORK measurements correlated satisfactorily with the Oconee reactor records. The average deviation of the burnup measurements from the calibration was 10% without corrections for initial enrichment, and 2.2% with corrections. Two anomalous assemblies were detected well outside these values. The system proved to be compatible with storage pool operations, and could be used most effectively to verify reactor records in a campaign involving a large number of assemblies. The test program was a cooperative effort involving Sandia National Laboratories, Los Alamos National Laboratory, Duke Power Company, and the Electric Power Research Institute.

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INTRODUCTION

The nuclear properties of spent reactor fuel depend on 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 an assembly. Burnup is commonly expressed as the time integral of the thermal power (e.g., gigawatt-days) per metric ton of uranium (GWD/MTU) metal originally contained in the assembly. A need for verification measurements arises from the incorporation of burnup credit concepts in the design of storage and transport systems for spent reactor fuel. A verification measurement can contribute to the acceptibility of burnup credit by preventing criticality problems due to miscalculation or misidentification of assemblies. The purposes of the measurement operation described here were to establish a database with the FORK detector system at an operating nuclear utility, to determine compatibility with utility operating procedures, and to develop an operational plan for implementing verification measurements with utility input.

Burnup Credit

Spent fuel assemblies must be stored and transported so that criticality is not possible, even under theoretically optimized conditions. Calculations of criticality have traditionally assumed that the assemblies are immersed in pure water, and that the composition of the fuel is unchanged from its original (fresh) state. Calculations using realistic spent fuel composition (burnup credit calculations) can result in significantly more efficient arrays of assemblies, and can reduce the need for expensive neutron absorbers. Burnup credit calculations make use of the fact that the nuclear reactivity of the spent assembly is reduced by the depletion of fissile material and the production of neutron absorbers by activation and fission reactions. The use of burnup credit calculations to replace "fresh fuel" calculations in the design of casks for transporting spent fuel can increase the number of assemblies that can be safely loaded into a cask by as much as a factor of four. The application of burnup credit to the design of spent fuel casks results in significantly reduced costs and risks in the transport and storage of spent fuel assemblies[2].

Spent fuel casks designed using burnup credit are restricted to accept only assemblies that meet certain minimum burnup restrictions, to limit maximum theoretical criticality to less than 95%. The characteristics of fuel acceptable for loading into a burnup credit cask can be specified by a loading curve, an example of which is shown in Figure 1-1. This loading curve is for illustrative purposes only. The design of each cask or storage arrangement would generate its own specific loading curve. The curve delineates the minimum burnup credit required for a particular initial enrichment and separates the assemblies with acceptable characteristics from those that are unacceptable. If unacceptable assemblies are present in the spent fuel pool, the possibility exists that some unacceptable fuel could be misloaded due to misapplied reactor records or an error in assembly identification.

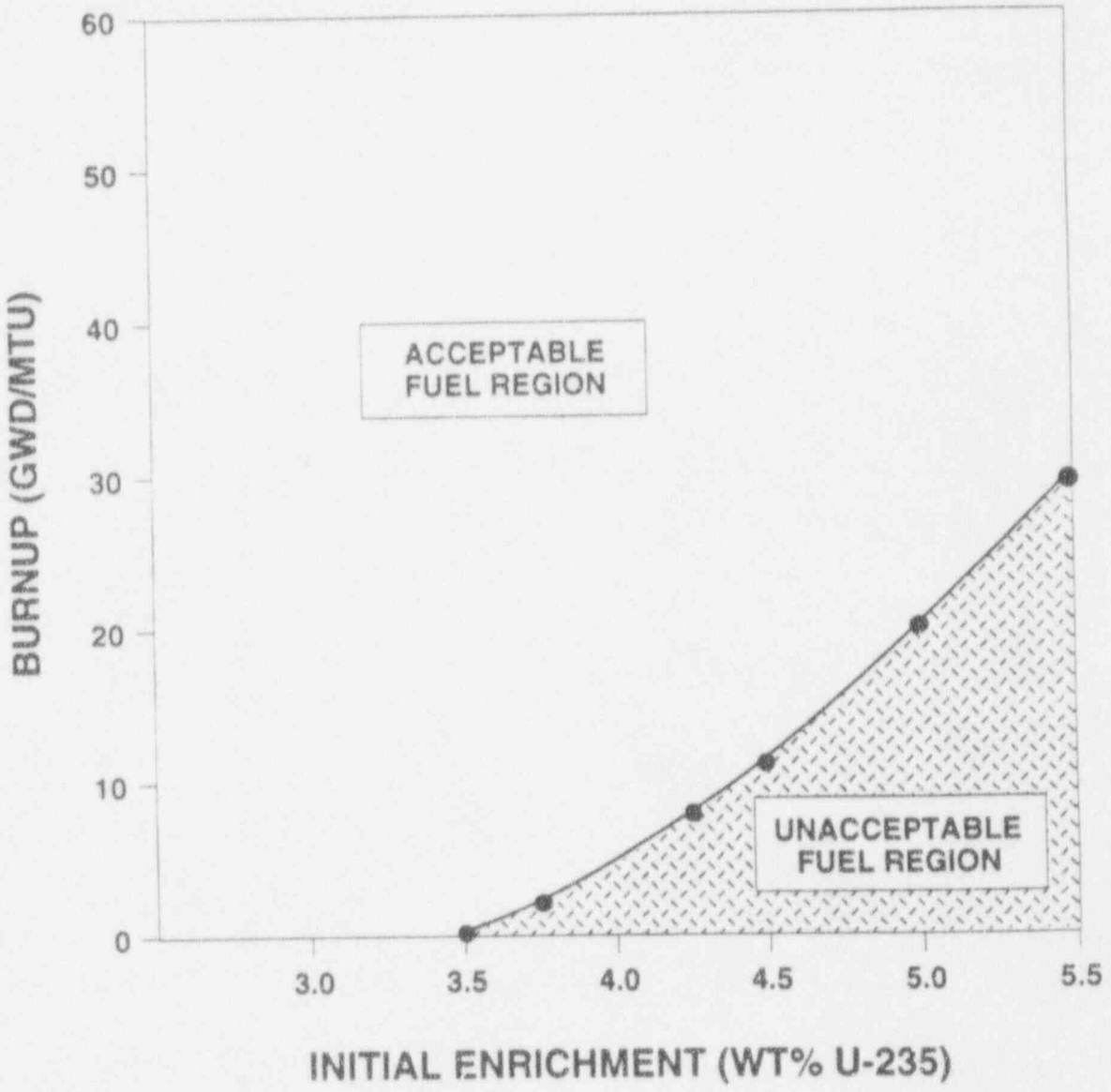

Figure 1-1
Burnup Credit Loading Curve

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 role of a burnup measurement after discharge is to demonstrate the consistency of the reactor records, detect possible misidentification of assemblies, and detect anomalous assemblies that might affect criticality.

Radiation From Spent Fuel

The emission rate of neutrons and gamma-rays produced by natural radioactive decay of the radioisotopes in the spent fuel can be related to the burnup of the assembly. Radioisotopes are produced in the fuel elements during operation of the reactor by activation and fission reactions, and decay with a wide range of half-lives after the assembly is discharged from the reactor. Some important gamma-ray and neutron emitting nuclides are listed in Table 1-1.

Table 1-1 Gamma-Ray and Neutron Emitting Nuclides

Nuclide	Half-Life (yr)	Radiation
^{242}Cm	0.45	n, spontaneous fission
^{244}Cm	18.1	n, spontaneous fission
^{134}Cs	2.06	γ, 605, 796 keV
^{137}Cs	30.0	γ, 662 keV

In the application of burnup credit, the fuel assemblies have been cooled for over 5 years, which greatly simplifies the analysis of the emitted radiation. For shorter cooling times many more isotopes are significant emitters, but most have decayed to insignificance after several years because of the predominance of short half-lives in the fission and activation products. After 5 years the predominant neutron emitter is curium-244, which is formed by successive neutron capture beginning with uranium-238. The neutron emission is found to follow a power law relationship with burnup in which the neutron signal increases with about the fourth power of the burnup. The neutron signal is therefore very sensitive to burnup. An additional advantage to the neutron measurement is to reduce the problem of self-shielding of the internally generated radiation. The attenuation is greater for gamma-rays than it is for neutrons, so that neutrons that reach the detector can originate from rods deeper inside the assembly than could be sampled by gamma-rays alone. The major gamma emitter, after several years of cooling, is cesium-137 which is produced as a fission product so that its production is essentially a linear function of burnup. The shorter-lived isotopes of curium and cesium are activation products that are insignificant after a few years of cooling. The combination of the gamma and neutron measurements allows both the burnup and the cooling time of each assembly to be checked. The purposes of this verification operation at Oconee Nuclear Station were to generate a database of measurements with the FORK detector at an operating nuclear utility, to examine the interfaces between the requirements of the measurement and the utility operations, and to obtain utility input to the development of an operational plan for implementing such a measurement.

FORK System

The FORK system measures the passive neutron and gamma-ray emission from individual spent fuel assemblies while in the storage pool. The system, designed at Los Alamos National Laboratory, 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. Comparison tests of this technique with more complex active and high-resolution measurement techniques have indicated essentially equal effectiveness[5].

The FORK detector and its associated electronics are shown in Figure 1-2. Figure 1-3 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 1-3) to measure the yield of neutrons, and one ion chamber (the inner brass cylinders shown between the fission chambers in Figure 1-3) to measure gross gamma-ray emission. One fission chamber (the epithermal detector) in each arm is imbedded in a polyethylene cylinder that is surrounded by a thin sheet of cadmium. The other fission chamber is outside the cadmium cover and is sensitive to thermal neutrons and the boron content of the water in the spent fuel pool. The polyethylene cylinders containing the detectors are inserted into the polyethylene outer cover shown in Figure 1-3. 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 locations. In the present use, the thermal detectors serve as a back-up measurement to the epithermal data. The system is diagrammed in an operational arrangement in Figure 1-4. The detector is moved in the storage pool to the location of the spent fuel assembly to be examined. The detector head is positioned several feet above the top of the storage rack so that the radiation shielding provided by the water of the storage pool is adequate to ensure that the measurement is not influenced by radiation from nearby assemblies. The assembly is raised in the storage rack so that its midpoint 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 100 seconds. A burnup profile can be obtained by performing the measurements at various points along the length of the assembly. 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.

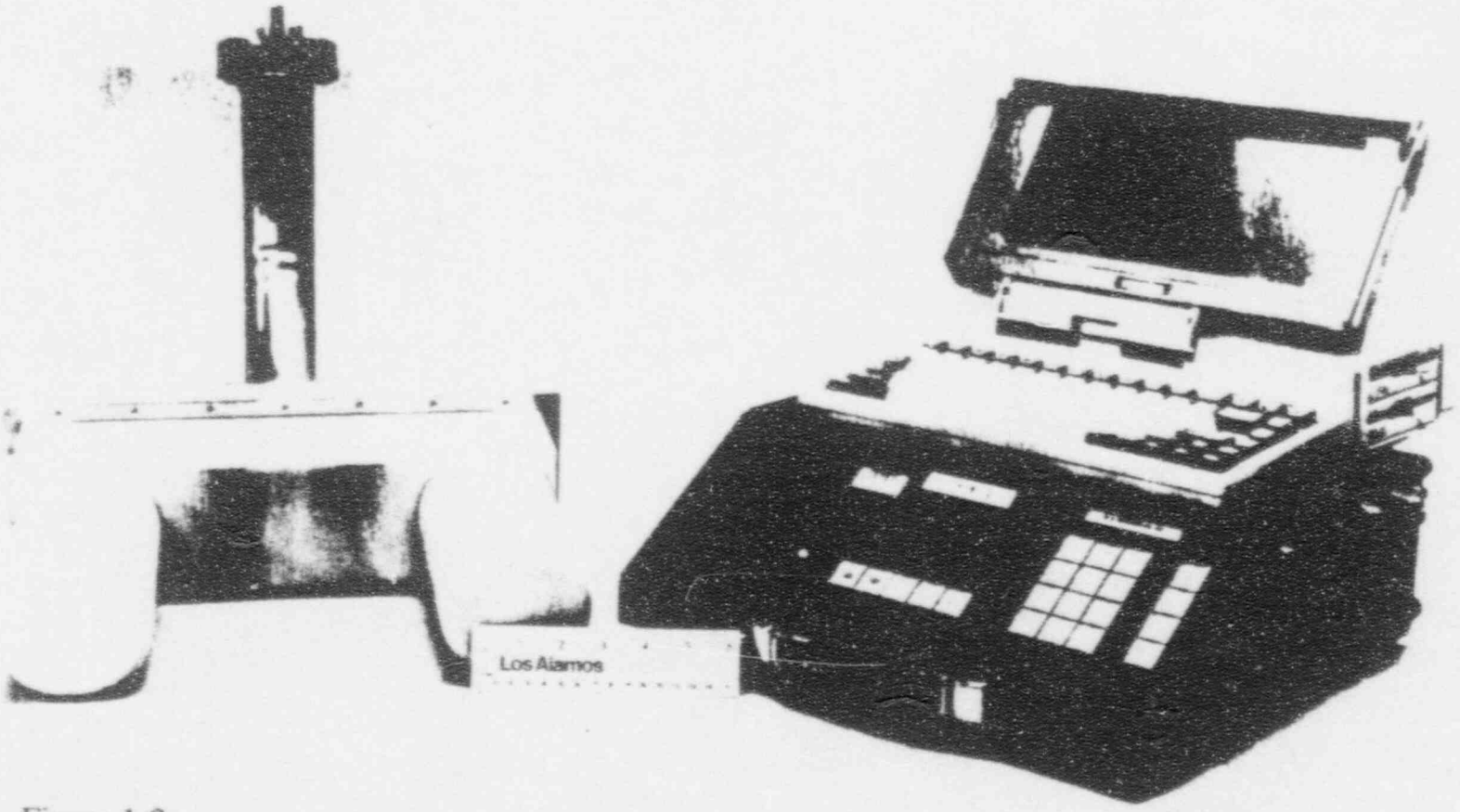

Figure 1-2
Fork Detector and Control Electronics

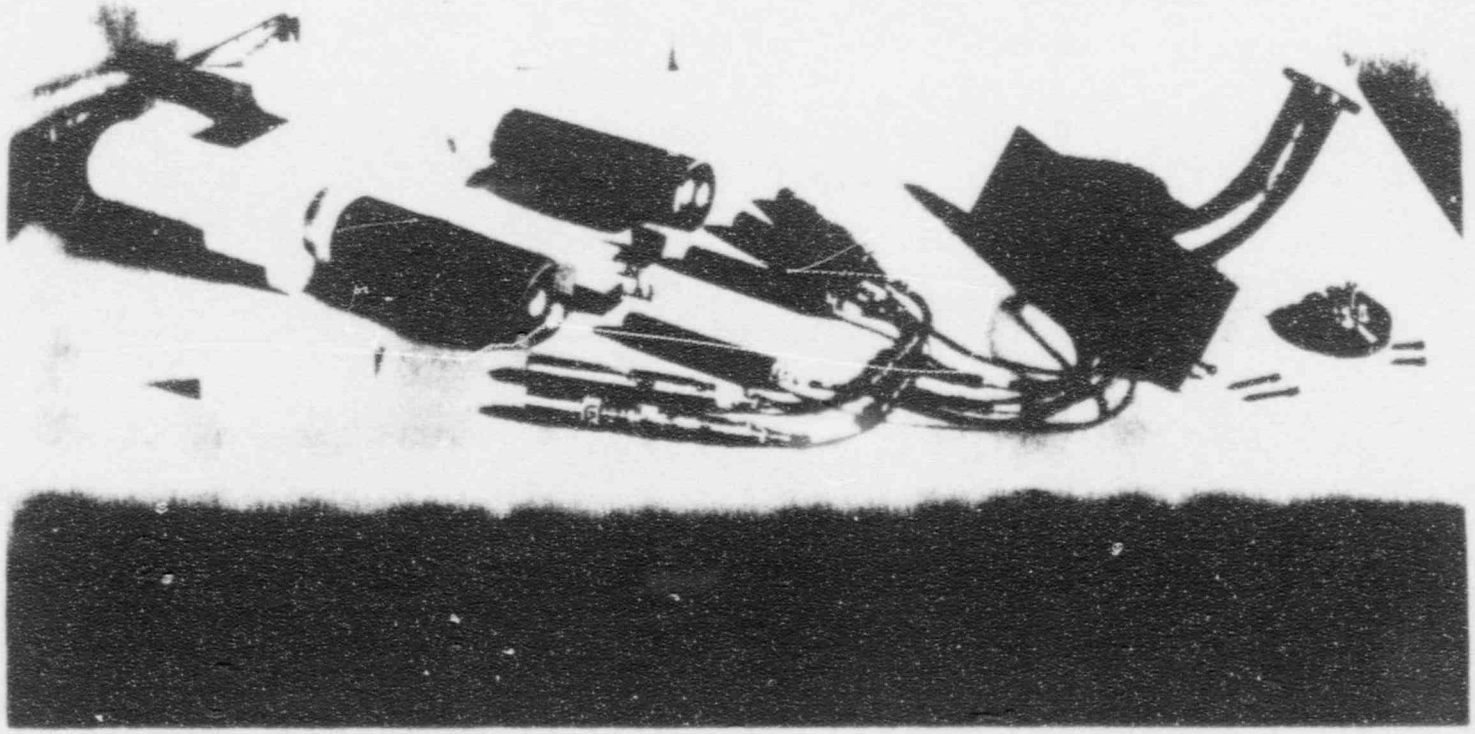

Figure 1-3
Dissassembled Fork Detector Head

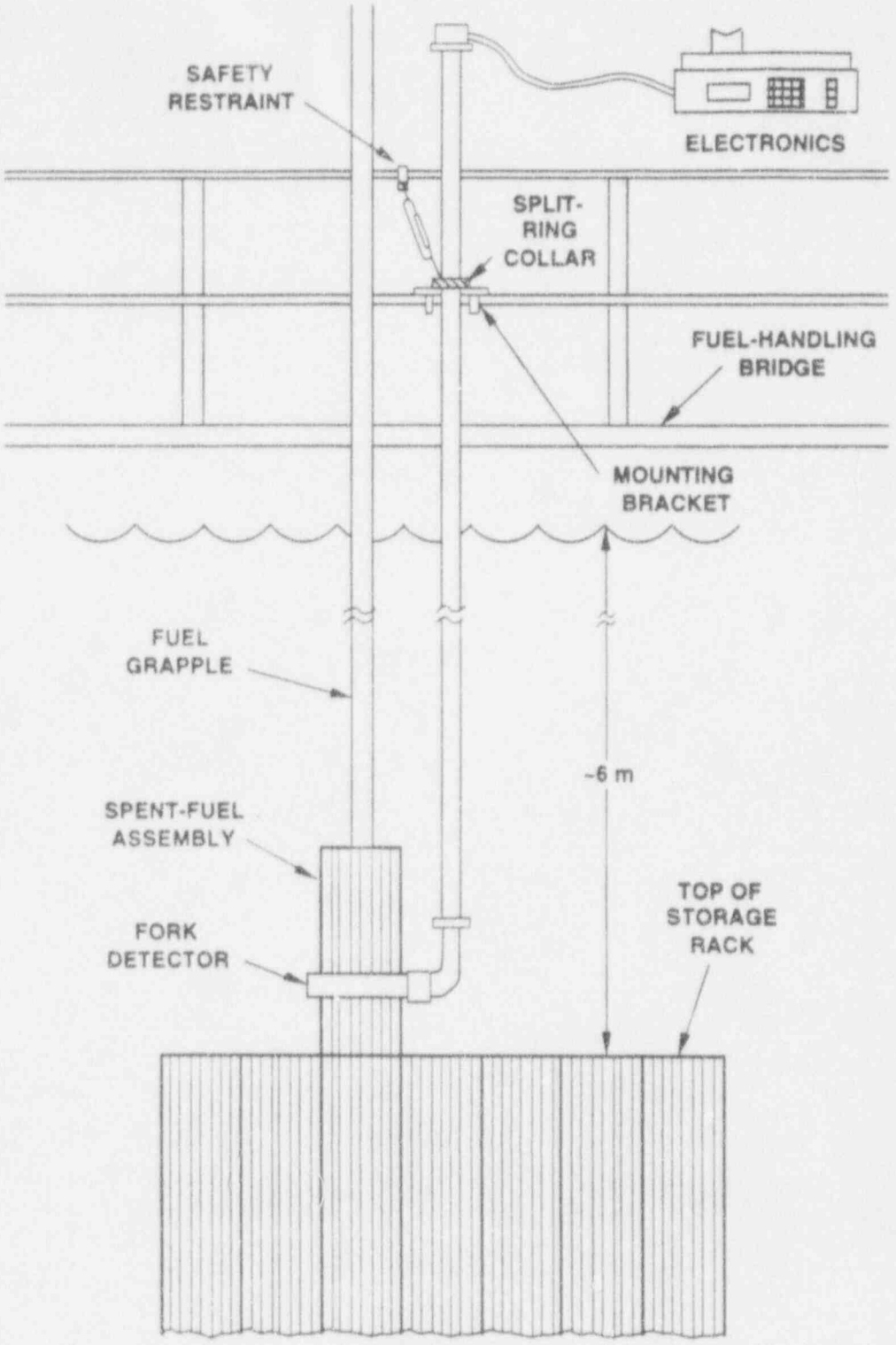

Figure 1-4
Fork System Arrangement in Spent Fuel Pool

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VERIFICATION TESTING AT OCONEE NUCLEAR STATION

Oconee Fuel

Oconee Nuclear Station is a three unit generating site utilizing Babcock and Wilcox 2568 MW (thermal) Pressurized Water Reactors. Duke Power Company began commercial operation on the site in 1973. The FORK measurements were performed in the spent fuel storage pool that serves Units 1 and 2. The spent fuel assembly design is a Babcock and Wilcox 15 x 15 array that accepts separate control components such as control rods, burnable poison rods, and neutron source rods. Each assembly contains 208 fuel rods and 16 guide tubes. The maximum cross section is 8.54 inches, and overall length is 165.6 inches. The nominal uranium weight is 464 kilograms.

Procedure

The FORK detector was suspended from a moveable carriage on the fuel handling bridge over the spent fuel pool. The demineralized water in the pool contained approximately 2000 parts per million boron. The top of the storage rack is about 25 feet below the water level. During testing, the fuel assemblies were lifted in the storage racks by means of an auxiliary hoist mounted on the Stearns-Roger fuel handling bridge. No assembly was completely removed from the rack. The detector head was fixed at a location about 6 feet above the top of the storage rack in the spent fuel pool. The shielding provided by the 6 feet of water was adequate to produce the lowest background reading. Each selected assembly was raised in its rack until the detector was at the center point of the asssembly. The detector was placed in contact with the assembly, and data were accumulated for 100 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 about one second. The assembly was then 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.

Results and Analysis

Ninety-three assemblies were measured in about 3 1/2 working days of operation. The initial enrichment of the assemblies ranged from 2.91 to 3.92 weight percent uranium-235. The range in assembly average burnup was from 20.3 to 58.3 GWD/MTU. The cooling times varied from 4.2 to 14.8 years. Background data were found in all cases to be less than 1% of the signal from the assembly. Appendix A lists the data and analysis values for all assemblies.

The approach used in the analysis described here is to accumulate data from a large 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 that are introduced by external calibration techniques, while retaining the sensitivity to detect measurements that are inconsistent with the burnup from the reactor records.

The neutron data were extrapolated back to the date of discharge using an exponential factor of half-life 18 years, the half-life of the principal neutron emitter, curium-244. The extrapolated data for the epithermal neutron detectors are shown in Figure 2-1, a log-log plot of neutron signal versus burnup (reactor record) for each assembly. The data are shown with and without a correction for the initial enrichment of the assemblies. The relationship of the neutron signal to burnup depends on the initial enrichment since curium-244 is produced by activation of uranium-238 rather than by fission reactions. The "uncorrected data" (uncorrected for initial enrichment) for 91 assembly measurements can be fit by a power law curve determined by a least squares fit such that the average absolute deviation in burnup is about 10%. This would be the best fit to the data if the initial enrichments were unknown. A factor to adjust the observed count rates for the variation in initial enrichment among the assemblies was calculated as described in Appendix B. The enrichment correction factor is normalized to an arbitrarily chosen enrichment of 3.0 weight percent uranium-235. For the Oconee data, the correction factor for initial enrichment varied from -7% to +53%.

The "Enrichment Corrected Data" are fit by the calibration curve shown in Figure 2-1, for which the analytical expression derived from a least squares fit to the data is

$$N = C \bullet B^{3.81} \qquad \text{(eq. 1)}$$

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 0.000788. The neutron signal is proportional to the 3.81 power of the burnup. This value closely matches the values observed in earlier operations with the FORK system. With the enrichment correction applied, the data have an average absolute deviation in burnup from the calibration curve of about 2.2%. Among the 91 assemblies fit by the calibration curve, only one assembly deviated by more than 6%.

The two data points marked "Outliers-Not Explained" in Figure 2-1 indicate two assemblies that exhibited much higher neutron signals than expected from the burnup records. These two data were not included in fitting the calibration curve. Both sets of neutron detectors indicated anomalous data for these two assemblies, but the corresponding gamma signals were not anomalous. The anomalies were noted at the time of measurement and the assemblies were remeasured with the same results. Since the objective of this operation was to build a substantial database of measurements with the FORK detector, it was necessary to measure as many assemblies as possible in the

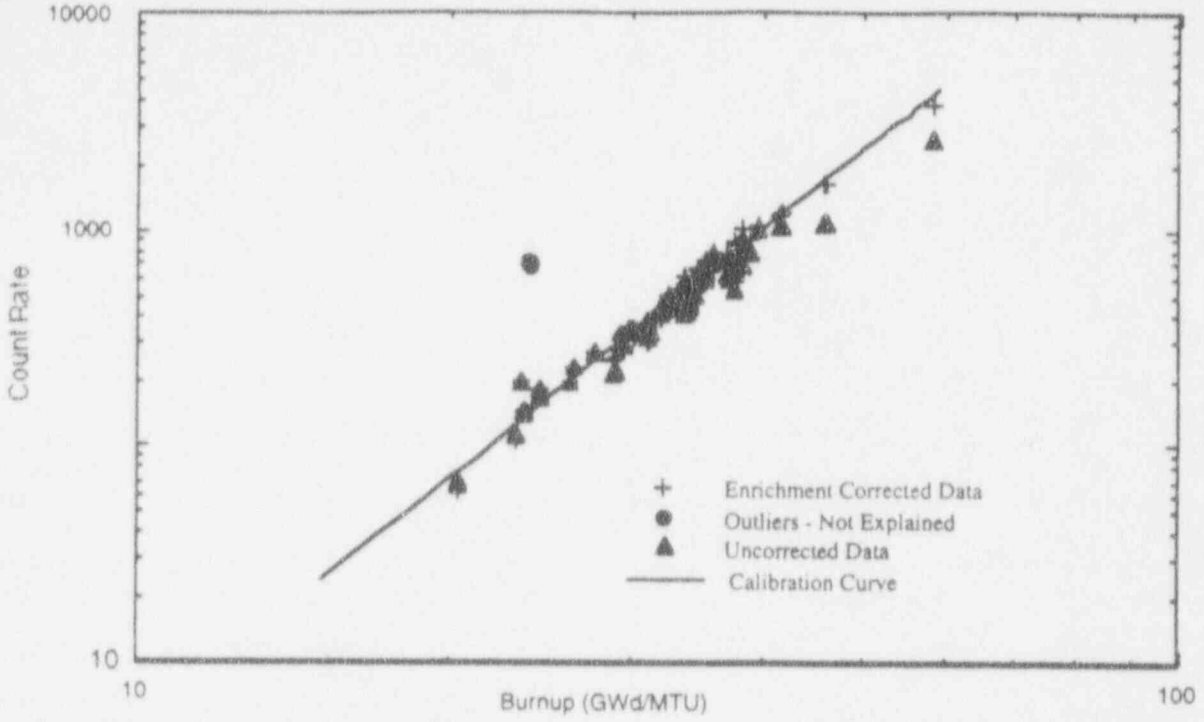

Figure 2-1
Neutron Data and Calibration

time available; no further measurements were made on the anomalous assemblies. Subsequent examination of the reactor records and histories for these assemblies did not reveal any explanation for the anomalous results. The two anomalous assemblies would require further study in a verification exercise to thoroughly eliminate the possibility of an instrumentation problem.

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 about 15%. Since the neutron data is a far more sensitive indicator of 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. The two assemblies that produced the lowest gamma-ray readings at about 1900 days cooling time did not produce anomalous neutron data and, for purposes of this exercise, were not considered to be significant deviations.

To investigate the capability of the FORK detector to measure the variation of burnup along the length of an assembly (burnup profiles), measurements were performed at several locations on two assemblies. The locations of the measurements were not determined precisely, but were approximately midway between structural bands on the assemblies. The gamma-ray data are shown in Table 2-1. The results are similar to other PWR profiles.

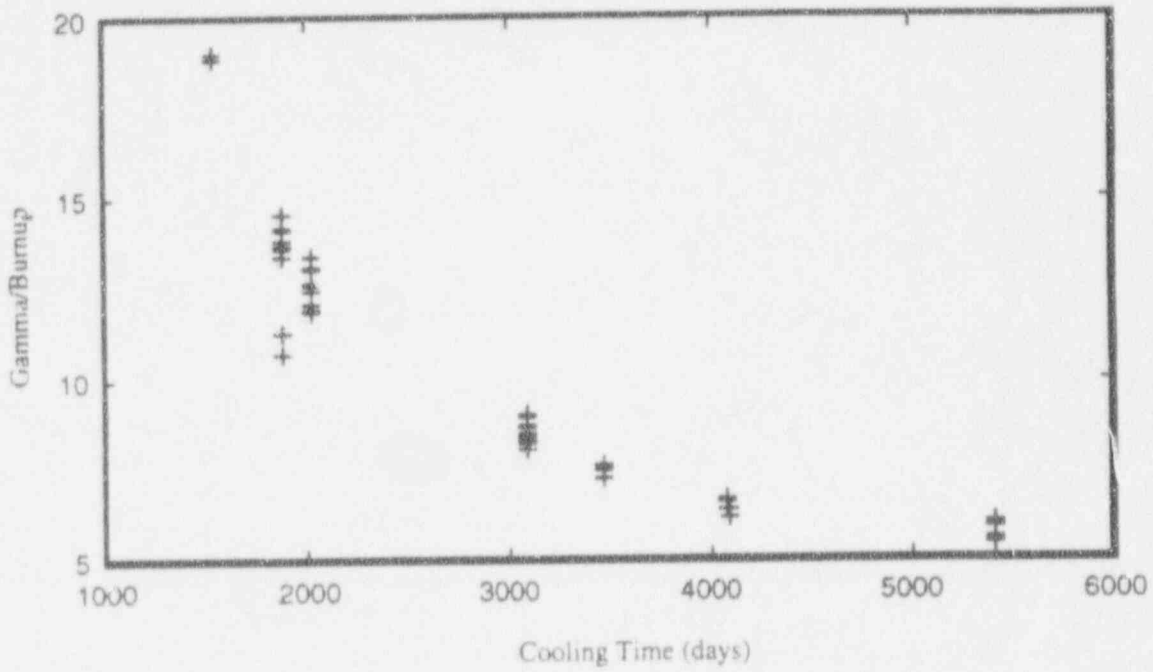

Figure 2-2
Gamma-Ray Data

Table 2-1 Gamma-Ray Data

Inches Above Midpoint (approx.)	Relative Gamma Signal	
0	221	246
21	220	
42	200	229
63	109	
Burnup (GWD/MTU)	29.42	32.70

Utility Comments

In general, the FORK detector performed quite well and proved relatively easy to set up and operate. It could provide an acceptable means for verifying burnup of fuel assemblies before loading into a burnup credit cask or canister. Measurements should be used to screen for gross errors in reactor records, such as inadvertently assigning the burnup of one assembly to another. Qualification of a fuel assembly for loading should be based on the verified reactor records for burnup since the records are likely to have less uncertainty than the measurement[3]. While the equipment is simple and straightforward when used by itself, its use could potentially interfere with loading operations if measurements were performed at the time of cask or canister loading. The preferred mode of operation, if 100% verification is required, would be to verify a large number of

assemblies in a single campaign and to administratively control access to qualified assemblies until the loading operation. The loading operation could then proceed efficiently without interruption or delay for measurements and decisions. An example of one possibility of administrative control would be to physically segregate qualified assemblies in a special section of the spent fuel pool. Some utilities may prefer to have the verification campaign performed by a certified vendor rather than to commit utility resources and personnel to an additional training, certification, and maintenance program specifically to perform the measurements. Additionally, site-specific safety reviews should be completed in advance of the campaign and the utility should be assisted in specifying the radiation dose to the FORK operator, including worst case scenarios. A number of specific recommendations concerning operations, interfaces, shielding, radiation protection, decontamination, etc., have been noted and will be integrated into further tests of the FORK system.

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CONCLUSIONS

The FORK measurements correlated well with the Oconee reactor records. The average deviation of the reactor burnup records from the calibration was 10% without corrections for initial enrichment and 2.2% with corrections. The derived calibration indicated that the neutron signal was proportional to the burnup raised to the 3.81 power. The gamma-ray signals were in general agreement (15%) with the burnup records. Ninety-three assemblies were measured in about 3 1/2 days of operation in the spent fuel pool. Two anomalous assemblies were detected that produced much higher neutron signals than the burnup would explain. In a verification campaign these two assemblies would require further study or be excluded from the acceptable fuel. The system is capable of generating burnup profiles with very short measuring time. The effectiveness of the FORK system is due to the sensitivity of the epithermal neutron yield to burnup, the self-calibration generated by a series of measurements, and the redundancy provided by three detection systems. The system proved to be compatible with utility operations, and appears to be adequate to verify reactor records for assemblies to be loaded into burnup credit casks.

Acknowledgments

The authors appreciate the essential help of the staff at Oconee Nuclear Station and Duke Power Company in the training, planning, and execution of this operation. Sandia National Laboratories managed the project with the sponsorship of Sandia's Laboratory Directed Research and Development Program (discretionary funds). Los Alamos National Laboratory made available a FORK system and the experienced personnel to operate it. The Electric Power Research Institute (EPRI) sponsored the utility measurement and publication programs. Duke Power Company made its facilities and personnel available in support of the EPRI proposal to perform the measurements.

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APPENDIX A: ASSEMBLY AND FORK DATA

Columns 1 through 4 were obtained from the reactor records for the assemblies tested. Column 1 is the assembly identification, column 2 is the initial enrichment, column 3 is the average burnup in gigawatt days per metric ton of uranium, and column 4 is the time interval between discharge from the reactor and the date of the measurement in days. Column 5 is the observed epithermal neutron count rate in counts per second, background subtracted. Column 6 is the observed thermal neutron count rate in counts per second, background subtracted. Column 7 is the epithermal data of column 5 extrapolated to the date of discharge ($e^{\lambda T}$, $\lambda = 0.0001048\ d^{-1}$, T = data of column 4) using the 18 year half-life of curium-244, and is the "uncorrected (for enrichment) data" of Figure 2-1. Column 8 is the correction factor for initial enrichment described in Appendix B. Column 9 is the epithermal neutron count rate of column 7 multiplied by the factor of column 8, and is the "corrected data" of Figure 2-1. Column 10 is the burnup value determined from the calibration line, which is derived from the best fit to the data of column 9. Column 11 is the absolute deviation (in percent) of the burnup determined from the calibration (column 10) and the reactor record burnup (column 3). Column 12 is the observed gamma (ion chamber) signal in milliamperes. Column 13 is the gamma signal divided by the burnup of column 3, and is plotted in Figure 2-2.

ID	E_i (wt%)	$\bar{B}$ (GWD/MTU)	T (d)	n (cps)	n_{th} (cps)	N_o (cps)	E_i factor	N (cps)	$\bar{B}_{cal}$ (GWD/MTU)	ΔB/B (%)	G (ma)	G/B
NJO1Q1	2.900	20.287	4097	45.10	57.45	69.30	0.931	64.52	19.41	4.32	131.35	6.47
NJO161	2.908	32.409	3479	341.10	438.67	491.24	0.944	463.55	32.55	0.43	248.88	7.68
NJO159	2.912	29.229	3478	232.85	279.14	335.31	0.944	316.63	29.45	0.77	219.74	7.52
NJO169	2.912	32.704	3473	347.00	456.00	499.43	0.946	472.65	32.72	0.04	246.50	7.54
NJO15C	2.914	29.476	3479	207.10	281.40	298.26	0.946	282.07	28.57	3.06	225.93	7.66
NJO168	2.914	30.141	3478	236.74	335.98	340.91	0.946	322.53	29.60	1.81	231.07	7.67
NJO14R	2.914	30.151	3479	243.73	320.73	351.02	0.946	332.09	29.82	1.08	229.19	7.60
NJO15Q	2.914	30.972	3478	259.65	355.60	373.90	0.947	353.91	30.33	2.09	234.47	7.57
NJO167	2.914	31.025	3477	261.71	361.05	376.83	0.947	356.69	30.39	2.05	235.87	7.60
NJO15T	2.914	31.763	3473	284.00	396.00	408.75	0.947	387.09	31.05	2.25	241.30	7.60
NJO14Y	2.915	23.092	4097	73.20	98.85	112.48	0.943	106.08	22.11	4.24	155.00	8.71
NJO15L	2.915	30.131	3478	235.14	328.75	338.61	0.947	320.56	29.55	1.93	229.75	7.63
NJO15J	2.915	31.001	3479	223.60	307.53	322.02	0.947	305.02	29.17	5.92	227.48	7.34
NJO15K	2.915	31.831	3479	306.30	402.80	441.12	0.948	418.05	31.68	0.48	242.37	7.61
NJO151	2.916	20.288	4097	45.20	61.60	69.45	0.942	65.44	19.48	3.97	134.60	6.63
NJO15X	2.916	20.288	4102	43.25	59.24	66.49	0.942	62.65	19.26	5.06	131.35	6.47
NJO15U	2.916	20.365	4102	45.18	57.77	69.46	0.942	65.45	19.48	4.33	127.00	6.24
NJO15F	2.916	29.424	3473	203.00	277.50	292.17	0.947	276.67	28.43	3.38	221.05	7.51
NJO15Y	2.916	31.082	3478	232.43	325.67	334.70	0.948	317.25	29.47	5.19	227.82	7.33
NJO15W	2.916	32.388	3479	320.43	412.60	461.48	0.949	437.79	32.06	0.99	243.46	7.52
3C54	2.997	29.820	5404	197.85	277.47	348.66	0.998	348.01	30.19	1.25	174.62	5.86
3C57	2.999	23.648	5404	80.22	88.97	141.37	0.999	141.28	23.84	0.80	132.28	5.59
3C32	3.001	23.562	5404	83.40	135.22	146.97	1.001	147.07	24.09	2.24	142.83	6.06
3C03	3.001	24.430	5404	95.10	147.51	167.59	1.001	167.70	24.93	2.06	146.00	5.98
3C38	3.001	26.286	5404	132.29	160.34	233.13	1.001	233.28	27.19	3.43	155.31	5.91
3C40	3.001	27.592	5403	155.06	199.02	273.23	1.001	273.40	28.34	2.72	164.00	5.94
3C19	3.003	23.460	5404	111.53	98.57	196.54	1.002	196.93	26.01	10.85	139.73	5.96
3C01	3.003	24.295	5404	96.61	157.69	170.25	1.002	170.58	25.04	3.09	147.17	6.06
3C02	3.003	24.311	5404	103.48	123.45	182.36	1.002	182.71	25.50	4.89	143.75	5.91
3C29	3.003	26.100	5405	111.63	169.97	196.75	1.002	197.12	26.01	0.34	155.22	5.95
NJO1Q5	3.064	32.460	3090	318.58	505.35	440.47	1.038	457.26	32.43	0.08	284.57	8.77
NJO1PQ	3.064	35.464	3090	509.79	725.24	704.84	1.036	730.54	36.67	3.40	320.62	9.04
NJO1PT	3.064	35.917	3092	532.07	637.43	735.80	1.036	762.41	37.08	3.25	313.24	8.72
NJO1Q4	3.065	32.463	3090	355.53	461.73	491.56	1.039	510.59	33.38	2.84	280.60	8.64
NJO1QE	3.065	35.108	3090	448.47	612.94	620.06	1.037	643.15	35.47	1.02	294.00	8.37

ID	E_i (wt%)	B (GWD/MTU)	T (d)	n (cps)	n_{th} (cps)	No (cps)	E_i factor	N (cps)	B_{cal} (GWD/MTU)	ΔB/B (%)	G (ma)	G/B
NJO1JT	3.065	39.624	3090	739.51	1046.27	1022.46	1.033	1056.27	40.39	1.94	355.93	8.98
NJO1J2	3.065	33.645	3090	736.55	991.62	1018.37	1.033	1052.02	40.35	1.78	349.06	8.80
NJO1QB	3.066	33.369	3090	344.45	534.75	476.24	1.039	494.73	33.11	0.78	285.24	8.55
NJO1Q2	3.066	33.377	3090	356.80	492.71	493.32	1.039	512.46	33.42	0.12	278.83	8.35
NJO1P5	3.066	33.481	3090	367.00	505.22	507.42	1.039	527.08	33.66	0.55	293.85	8.78
NJO1P6	3.066	33.913	3090	343.60	476.37	475.07	1.039	493.36	33.09	2.44	290.15	8.56
NJO1QC	3.066	34.438	3090	362.24	426.22	500.83	1.038	519.97	33.54	2.60	283.87	8.24
NJO1Q3	3.066	35.116	3090	448.45	583.90	620.03	1.038	643.47	35.47	1.01	292.32	8.32
NJO1PA	3.066	35.589	3091	511.45	639.70	707.21	1.038	733.73	36.71	3.16	308.76	8.68
NJO1PP	3.066	35.590	3090	520.61	660.72	719.21	1.038	746.80	36.88	3.64	308.61	8.67
NJO1PD	3.067	33.368	3091	374.48	460.44	517.82	1.039	538.22	33.85	1.44	280.69	8.41
NJO1PV	3.067	33.932	3090	321.66	470.04	444.73	1.039	462.11	32.52	4.15	286.95	8.46
NJO1PE	3.067	34.683	3090	446.63	555.53	617.52	1.039	641.37	35.44	2.19	294.18	8.48
NJO150	3.068	32.454	3090	341.30	521.59	471.89	1.040	490.99	33.04	1.82	284.18	8.76
NJO1Q7	3.068	33.667	3090	418.73	670.22	578.94	1.040	601.99	34.86	3.54	300.77	8.93
NJO1PU	3.068	33.820	3090	337.57	416.27	466.73	1.040	485.27	32.94	2.60	280.20	8.29
NJO1QD	3.068	34.682	3090	450.04	588.99	622.23	1.039	646.62	35.52	2.41	288.44	8.32
NJO1PG	3.068	34.991	3090	458.60	675.94	634.07	1.039	658.80	35.69	2.00	305.10	8.72
NJO1PC	3.068	35.106	3090	464.00	585.60	641.53	1.039	666.51	35.80	1.98	293.56	8.36
NJO1Q9	3.068	35.342	3092	490.70	622.7	678.59	1.039	704.91	36.33	2.79	302.85	8.57
NJO1PH	3.068	35.354	3091	462.25	683.74	639.18	1.039	663.97	35.76	1.16	307.20	8.69
NJO1PZ	3.069	35.117	3090	497.47	636.60	687.81	1.039	714.97	38.47	3.84	306.44	8.73
NJO1PW	3.070	33.383	3090	369.50	446.64	510.88	1.041	531.89	33.74	1.08	275.26	8.25
NJO1PL	3.070	33.969	3090	321.08	396.95	443.93	1.041	462.04	32.52	4.26	275.90	8.12
NJO1PJ	3.070	33.989	3090	364.91	445.75	504.53	1.041	525.10	33.63	1.05	285.81	8.41
NJO1PM	3.070	35.909	3090	546.69	752.77	755.86	1.040	785.73	37.38	4.09	325.47	9.06
NJO1PY	3.072	34.699	3090	418.73	670.22	578.94	1.041	602.95	34.87	0.50	300.77	8.67
NJO1PX	3.072	34.703	3090	462.11	574.60	838.92	1.041	665.42	35.78	3.12	296.65	8.55
NJO1QO	3.073	35.112	3091	430.65	628.07	595.49	1.042	620.35	35.13	0.06	296.50	8.44
NJO3AS	3.238	28.738	1876	180.52	233.17	219.76	1.145	251.70	27.73	3.49	308.60	10.74
NJO3AR	3.238	28.845	1876	183.02	251.77	222.80	1.145	255.16	27.83	3.51	325.62	11.29
NJO3BY	3.238	38.093	1876	748.35	664.00	911.02	1.125	1024.83	40.07	5.20	509.99	13.39
NJO3BD	3.242	37.480	1877	516.00	817.00	628.23	1.130	709.64	36.39	2.90	515.00	13.74
NJO3BC	3.242	37.672	1876	556.62	749.74	677.61	1.129	764.90	37.12	1.48	511.88	13.59
NJO33P	3.242	37.679	1876	622.25	700.37	757.51	1.129	855.07	38.22	1.43	517.50	13.73
NJO3BG	3.242	38.073	1877	572.80	811.70	697.38	1.127	786.01	37.38	1.81	529.36	13.90
NJO3BF	3.242	38.250	1877	567.97	822.33	691.50	1.126	778.82	37.29	2.50	521.23	13.63

ID	E_i (wt%)	B (GWD/MTU)	T (d)	n (cps)	n_{th} (cps)	No (cps)	E_i factor	N (cps)	B_{cal} (GWD/MTU)	ΔB/B (%)	G (ma)	G/B
NJO3B2	3.245	38.999	1877	656.45	976.93	799.21	1.124	898.32	38.71	0.73	566.23	14.52
NJO3BO	3.245	41.489	1876	843.17	1151.26	1026.45	1.131	1160.46	41.40	0.21	586.55	14.14
NJO3C4	3.253	41.755	1876	863.34	1159.23	1051.00	1.143	1201.60	41.78	0.06	593.75	14.22
NJO3C9	3.256	36.891	1877	513.40	770.63	625.06	1.139	712.11	36.43	1.26	509.36	13.81
NJO3CA	3.256	36.948	1876	493.02	776.74	600.19	1.139	683.64	36.04	2.46	504.37	13.65
NJO379	3.402	33.768	2029	345.98	448.61	428.00	1.235	528.55	33.69	0.24	401.88	11.90
NJO37A	3.403	33.822	2030	357.57	445.40	442.38	1.235	546.50	33.98	0.48	403.11	11.92
NJO36P	3.406	33.698	2029	369.18	462.71	456.70	1.238	565.26	34.29	1.75	407.95	12.11
NJO37D	3.408	36.995	2029	477.05	620.04	590.14	1.224	722.20	36.56	1.17	459.37	12.42
NJO37E	3.408	37.399	2029	519.08	673.67	642.13	1.221	784.18	37.36	0.11	474.99	12.70
NJO37G	3.410	37.660	2029	580.40	827.97	717.99	1.221	876.33	38.46	2.13	490.94	13.04
NJO368	3.412	37.460	2029	437.08	685.34	540.69	1.223	661.32	35.73	4.63	466.87	12.46
NJO36F	3.412	37.691	2030	511.67	649.37	633.03	1.221	773.22	37.22	1.25	479.36	12.72
NJO369	3.413	33.713	2029	331.25	487.27	409.78	1.242	508.96	33.36	1.06	402.50	11.94
NJO365	3.413	36.995	2029	512.25	718.57	633.68	1.227	777.34	37.27	0.75	466.25	12.60
NJO35Z	3.413	37.526	2029	513.78	701.97	635.57	1.223	777.45	37.27	0.67	469.07	12.50
NJO36E	3.417	37.516	2029	517.75	792.34	640.49	1.226	784.98	37.37	0.39	491.24	13.09
NJO37L	3.919	45.785	2029	863.55	1158.14	1068.26	1.532	1636.92	45.31	1.04	614.38	13.42
NJO37N	3.919	46.105	2029	863.98	1208.47	1068.79	1.530	1634.87	45.29	1.76	603.75	13.10
NJO37K	3.919	58.310	1541	2257.00	2913.94	2652.78	1.441	3823.85	56.60	2.94	1108.12	19.00
NJO37K	3.919	58.310	1541	2241.94	2963.54	2635.07	1.441	3798.32	56.50	3.11	1101.87	18.90
Outliers										avg. = 2.24		
3C09	3.004	23.815	5404	400.68	426.88	706.10	1.003	707.94	36.37	52.72	132.97	5.58
3C28	3.004	23.911	5404	385.46	586.79	679.28	1.003	681.05	36.00	50.57	130.88	5.47
										2.24		
Profile Data - NJ016S												
1&2 Grids	2.912	32.704	3473	297.35	375.00		0.946				229.00	
3&4 Grids	2.912	32.704	3473	347.00	456.00		0.946				246.50	
Profile Data - NJ015F												
Nozzle&1	2.916	29.424	3473	18.34	34.36		0.947				108.95	
1&2 Grids	2.916	29.424	3473	142.00	209.80		0.947				199.80	
2&3 Grids	2.916	29.424	3473	206.50	285.00		0.947				220.50	
3&4 Grids	2.916	29.424	3473	203.00	277.50		0.947				221.05	

APPENDIX B: ENRICHMENT CORRECTION FACTOR

In the analysis of the neutron data the assumption is made that the source of the neutrons 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 depends on the initial enrichment of the assemblies. This is due to the fact that the production of curium depends on the neutron flux in the reactor, rather than solely on the fission rate. The relationship between the production of curium and the burnup depends on 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-CELL[6] and EPRI-CINDER[7] 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 covering the initial enrichments of interest in this report. Since the neutron data for each assembly are multiplied by a correction factor, only relative values are required. 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 routines. The enrichment correction factors are listed for each of the Oconee assemblies in Appendix A, column 8.

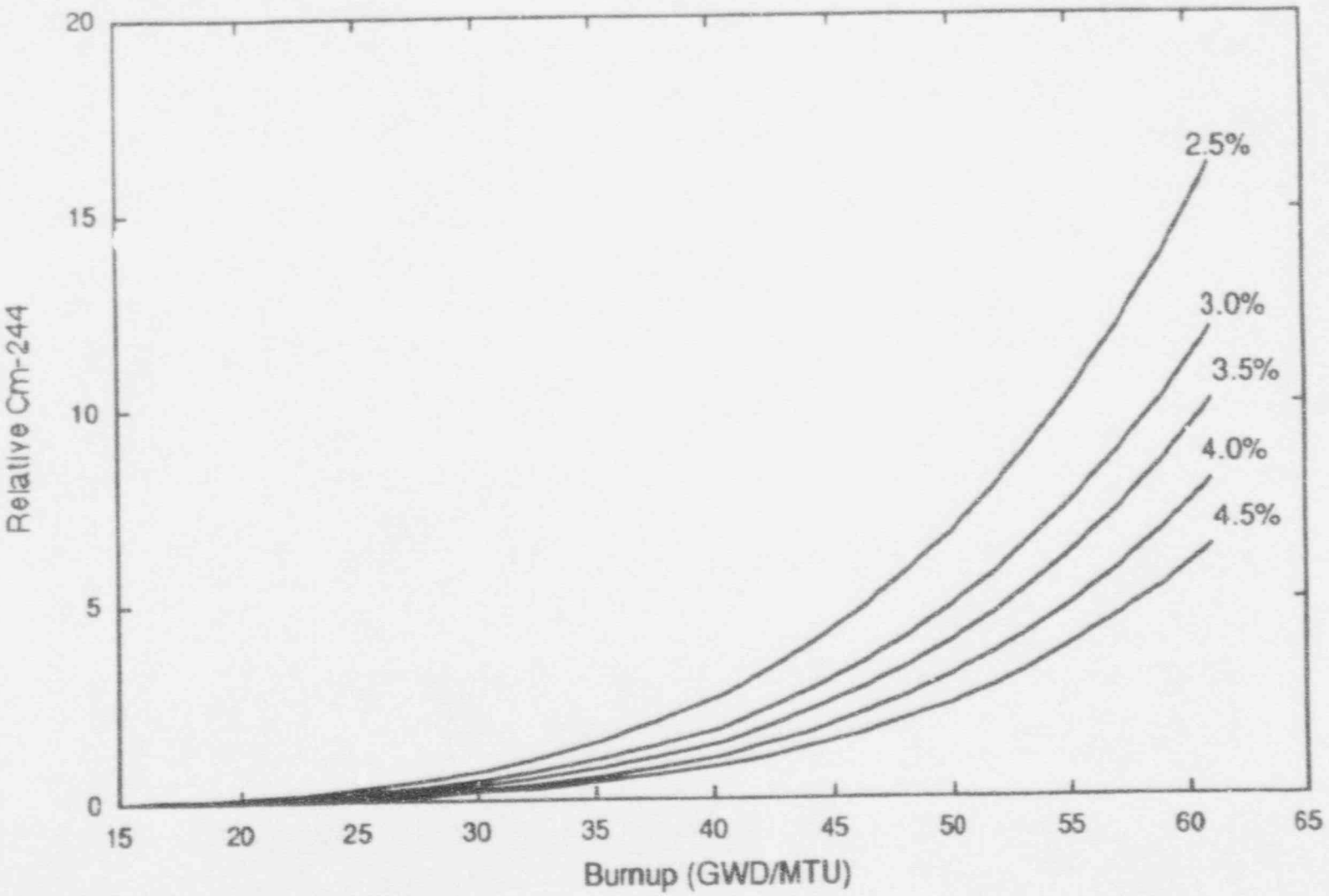

Figure B-1
Relative Cm-244 Production

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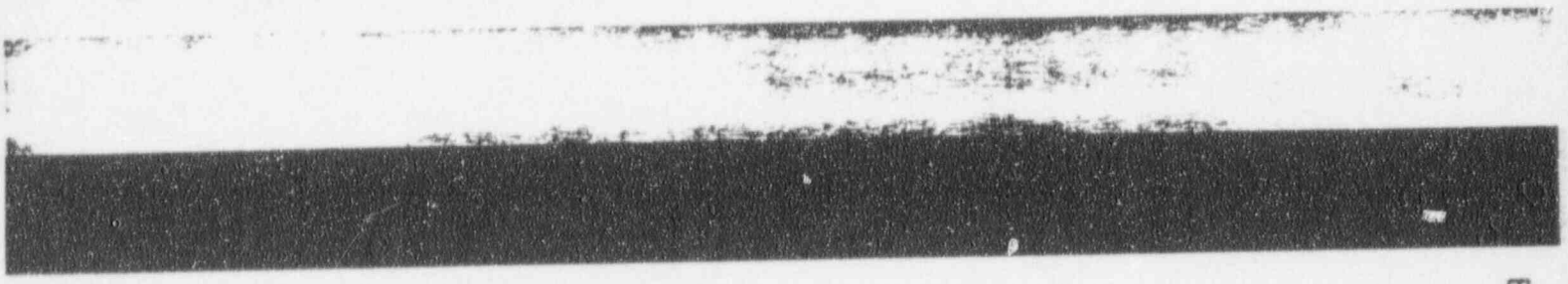

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Burnup Verification Measurements on Spent-Fuel Assemblies

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