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Revision 1

SUB-LEU-METAL-THERM-001: Subcritical Measurements of Low-Enriched Tubular Uranium Metal Fuel Elements Before and After Irradiation

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200

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Richland, Washington

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Fluor Hanford, Inc.

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**SUBCRITICAL MEASUREMENTS OF LOW-ENRICHED
TUBULAR URANIUM METAL FUEL ELEMENTS BEFORE AND
AFTER IRRADIATION**

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List of Acronyms

3DN	neutron diffusion computer code
BNWL	Battelle Northwest Laboratories (now Pacific Northwest National Laboratory)
DCODE	computer program-design code
IDMS	isotope dilution mass spectrometry
MCNP	Monte Carlo N-Particle computer code
MKIA	N Reactor Fuel Type (Mark IA)
MKIV	N Reactor Fuel Type (Mark IV)
ORIGEN	Oak Ridge Isotope Generation (burnup/decay computer code)
WIMS	Winfrith Improved Multigroup Scheme (lattice physics transport code)

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SUBCRITICAL MEASUREMENTS OF LOW-ENRICHED TUBULAR URANIUM METAL FUEL ELEMENTS BEFORE AND AFTER IRRADIATION

IDENTIFICATION NUMBER: SUB-LEU-MET-THERM-001

KEY WORDS: uranium, low-enriched, metal, tubular, thermal, water-moderated, heterogeneous, reflected, lattice array, cylinder, irradiated, N Reactor

1.0 DETAILED DESCRIPTION

1.1 Overview of Experiment

With the shutdown of the Hanford PUREX (Plutonium-Uranium Extraction Plant) reprocessing plant in the 1970s, adequate storage capacity for spent Hanford N Reactor fuel elements in the K and N Reactor pools became a concern. To maximize space utilization in the pools, accounting for fuel burnup was considered. Fuel that had experienced a neutron environment in a reactor is known as spent, exposed, or irradiated fuel. In contrast fuel that has not yet been placed in a reactor is known as green, unexposed, or unirradiated fuel. Calculations indicated that at typical fuel exposures for N Reactor, the spent-fuel critical mass would be twice the critical mass for green fuel. A decision was reached to test the calculational result with a definitive experiment. If the results proved positive, storage capacity could be increased and N Reactor operation could be prolonged. An experiment to be conducted in the N Reactor spent-fuel storage pool was designed and assembled (References 1 and 2) and the services of the Battelle Northwest Laboratories (BNWL) {now Pacific Northwest National Laboratory [PNNL]} critical mass laboratory were procured for the measurements (Reference 3).

The experiments were performed in April 1975 in the Hanford N Reactor fuel storage pool. The fuel elements were MKIA fuel assemblies, comprised of two concentric tubes of low-enriched metallic uranium. Two separate sets of measurements were performed: one with unirradiated fuel and one with irradiated fuel. Both the unirradiated and irradiated fuel, were measured in the same geometry. The spent-fuel MKIA assemblies had an average burnup of 2865 MWd (megawatt days)/t.^a

A constraint was imposed restricting the measurements to a subcritical limit of $k_{\text{eff}} = 0.97$. Subcritical count rate data was obtained with pulsed-neutron and approach-to-critical measurements.

Ten (10) configurations with green fuel and nine (9) configurations with spent fuel are described and evaluated. Of these, three (3) green fuel and four (4) spent fuel loading configurations were considered to serve as benchmark models. However, shortcomings in experimental data, such as the uncertainty in fuel exposure impact on reactivity and the pulse neutron data evaluation methodology, failed to meet the high standards for a benchmark problem. Nevertheless, the data

^a t is short ton, 2000 lb (907.2 kg). This is for U in fuel.

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provided by these subcritical measurements supply useful information to analysts evaluating spent fuel subcriticality.

The original purpose of the subcritical measurements was to validate computer model predictions that spent N Reactor fuel of a particular, typical exposure (2740 MWd/t) had a critical mass equal to twice that of unexposed fuel of the same type. The motivation for performing this work was driven by the need to increase spent fuel storage limits. These subcritical measurements confirmed the computer model predictions.

1.2 Description of Experimental Configuration

1.2.1 Sources of Data

The main sources of the experimental data that have been incorporated into the main text and appendices are the first five references. Details of the experiment, dimensions, and masses are found in References 1 and 2. Other experimental data and measurement details are covered in Reference 3. Elemental compositions and impurity levels were extracted from the *Alcoa Aluminum Handbook*^b and the *Activity of Fuel Batches Processed through Hanford Separation Plants* (Reference 4).

1.2.2 Experimental Arrangement

Since half of the measurements were with highly irradiated fuel, the availability of spent fuel and sufficient radiation shielding were essential. The N Reactor spent-fuel pool represented a suitable location for the measurements. The pool, that contained the test assembly, measured 101 ft 10 inches by 42 ft 6 inches with a water depth of 23 ft. An area was selected at the bottom of the pool that was at least 6 ft from other fissile materials, 2 ft from the nearest wall (sufficient to minimize reactivity impacts), 18 ft under water, and readily accessible from the surface. Other parts of the basin were used in the preparation of the green and spent fuel test tubes (see Figure 1).

^b *Alcoa Aluminum Handbook*, Aluminum Company of America, pp. 46 - 50

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Figure 1. N Reactor Spent-Fuel Pool Showing the Location of the Test Stand and Other Stations Where Fuel Tubes Were Prepared

The entire test assembly was immersed in the clean filtered water of the spent-fuel pool. No records of water temperature could be located. However, based on other information, the water in the basin in April was assumed to be 10 – 15°C. The test assembly, made out of 6061 aluminum, is shown in Figure 2 with empty fuel tubes inserted. The grid plates, with a 3.2-inch hexagonal lattice pitch, are pictured in Figure 2. The test-assembly fixture was constructed out of 2-inch-wide by 0.25-inch-thick Al cross braces, 2-inch by 2-inch by 0.25-inch-thick aluminum angle posts, and by 0.75-inch by 0.75-inch by 0.125-inch-thick aluminum angle braces to hold the lattice plates, see Figure 2. The braces, angle ports and fixtures for holding the detector and source tubes in place were attached to the steel bottom plate of the test assembly.

Figure 3 shows the experiment in progress. A test tube is being moved through the basin towards the partially filled test assembly resting on the basin floor. (The words basin and pool are interchangeably used.)

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Figure 2. Fabrication of Test Assembly. Empty Fuel Tubes Are Inserted in the Lattice Plates.

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Figure 3. Experiment in Progress: A Test Tube Is Being Moved Towards the Partly Loaded Test Assembly.

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All measurements were made with aluminum alloy (6061) fuel tubes containing two MKIA fuel assemblies^c stacked end-to-end. The tubes filled with green fuel were measured in the lattice configuration illustrated in Figures 2 and 3, and tubes filled with spent fuel were measured with the same equipment.

1.2.3 Fuel Tube Preparation

1.2.3.1 Green Fuel Test Tube Preparation

To prepare the fuel tubes, aluminum tubes containing two green MKIA fuel elements were filled with clean water. The bottoms of the tubes were closed by solid 6061-T6 aluminum plugs (see Figure 9). The tops of the tubes were subsequently closed with a stopper and ultrasonically checked for the presence of any entrapped air pockets. This precaution was taken to avoid perturbations and flux distortions and any contamination on the fuel cladding surfaces. Only tubes without entrapped air were used.

Figure 4 shows the hydraulic lifting device, a MKIA fuel assembly being inserted in the aluminum alloy (6061) fuel test tube, and the thermos type stopper used to protect the green fuel from contamination. The stopper consisted of a neoprene cylinder sandwiched between two circular aluminum metal plates and a threaded bolt. Turning the handle of the bolt expanded the neoprene against the tube wall (see Figure 5 for details). The material of the bolt was brass^d and the handle was brass. The approximate weight of the neoprene was 138 g.

The support clips, seen in Figure 4 on the fuel-element outer surface, centered the fuel elements inside the aluminum test tube and maintained fixed separation between the fuel element and the aluminum test tube. The inner cylinder of the MKIA fuel assembly was centered in the outer one by means of special locking clips (see Figure 10).

^c The MKIA (Mark IA) is one of two main N Reactor fuel types; another type is MKIV (Mark IV).

^d Brass is assumed to be 85% Cu and 15% Zn and have a density of 8.6 g/cc (Mark's Standard Handbook for Mechanical Engineers 8th edition, McGraw-Hill, 1978).

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Figure 4. Hydraulic Fuel Tube Lifting Device, MKIA Fuel Assembly Being Placed Inside a Fuel Tube, and the Neoprene Stopper for Green Fuel Protection.

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Figure 5. Neoprene Stopper Used to Seal Fuel Tubes from Basin Water.

Figure 6 shows a top view of the test assembly with the thickness of the aluminum alloy (6061) spacing plate, the lattice pitch and the hole size (3.0625 inches diameter). Detectors, as well as the ^{252}Cf source and the pulsed-neutron source locations, are identified. The detectors, the 0.78-microgram ^{252}Cf source, and the pulsed-neutron generator with a tritium target were positioned such that their active regions were in the same plane as the center of the fueled region.

The fission chambers contained 3 g of fissile material and the proportional counter was filled with up to 10 atmospheres of BF_3 each. The outer diameter of the aluminum detector tubes was 2.5 inches, to fit in the lattice plate holes. The instrumentation and source tubes were sealed on the

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bottom to keep the wires, detectors, and sources dry. The detector tubes were 29-ft-long aluminum tubes, weighted with Pb^e, and extended from above the water to the floor of the basin.

Figure 6. Top View of Test Assembly with Various Dimensions and Showing the Location of Detectors and Sources and the Hole Blocking Plate.

^e Pb weights are attached to the outside of the tubes as shown in Figure 3. They are at least 2 ft above the fueled region of the test assembly.

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The ^{252}Cf source, producing approximately 10^6 neutron/sec, was packed in an aluminum plug that was inside a 4-inch-OD and 29-ft-long tube. The pulsed-neutron source was inside a 4-inch-OD tube 29 ft long. All tubes were Pb weighted and sealed on the bottom to keep the interior dry.

Details on the fission detectors, the proportional counter (BF_3), at the ^{252}Cf source, and pulsed-neutron source are fragmentary.

The hole limiter, or blocking plate, is a safety device to prevent exceeding the k_{eff} limit of 0.97 during green fuel measurements. It is shown in Figures 6, 7 and 8. A sketch of the plates is found in Figure 8. The plate was an engineered safety feature to preclude loading any green-fueled tubes into the outer-row positions. It was used to partially cover the outermost rows of openings during the green-fuel measurement. It was manufactured out of 0.25-inch-thick Al alloy 6061 plate. For the irradiated fuel measurements, the blocking plate was removed. The blocking plate was held in position by three protrusions as shown in Figure 8. The dimensions of the protrusions are given in the same figure.

Figure 7. Top lattice plate^f with hole blocking plate

^f The aluminum lattice or spacer plates were fabricated out of 0.5-inch Al alloy 6061.

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Figure 8. Hole Blocking Plates

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Figure 9 provides a summary of vertical dimensions in the test assembly. It shows the fuel tubes containing the two MKIA elements sitting on top of the tapered bottom plug made out of 6061 aluminum alloy. The solid plug, or positioning cone, guides the fuel tube into the lattice position and keeps the fuel 2 inches above the steel bottom plate. The plug rests on top of the 1-inch-thick stainless steel plate that rests on the concrete basin floor.

The lower lattice plate relative to the bottom of the cone was 2.0 inches and the top plate bottom was 43.26 inches above the bottom of the cone. The two 0.5 inch thick Al plates were spaced 40.76 inches apart. The diameter of the 91 holes was 3.0625 inches.

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Figure 9. Aluminum Fuel Test Tube Showing Locations of Fuel Elements and Elevations Relative to the Basin Floor

N Reactor fuel, as shown in Figure 10, is a concentric arrangement of two zirconium-clad low-enriched uranium metal tubes. The configuration of the two concentric tubes is referred to either as a fuel element or as a fuel assembly. The MKIA fuel element was composed of the outer tube enriched to 1.25 wt% ^{235}U in uranium and the inner tube enriched to 0.947 wt% ^{235}U in uranium.

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The tubes were held in a concentric arrangement using special locking clips and spacers as shown in Figure 10.

The dimensions as given in Figures 6, 7, 8, and 10 and in Table 1 represent the nominal values. The reported dimensions were in inches; the centimeter values shown in Table 1 were calculated from them. Uranium masses in kilograms were derived from the reported weights in pounds (lb). The fuel assembly diameters for the clad outer and inner element represent manufacturer's tolerances. The dimensions of the fuel elements were carefully checked to be within the data bounds given in Table 1.

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Figure 10. N Reactor Tube-in-Tube MKIA Fuel Element or Fuel Assembly

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Table 1. Unirradiated MKIA Fuel Element Dimensions and Isotopic Composition ^(a)

Outer Tube		MKIA
Zirconium Clad OD, inch (cm)		2.404 (6.106) ^(b)
Uranium OD, inch (cm)		2.354 (5.979)
Uranium ID, inch (cm)		1.817 (4.615)
Zirconium Clad ID, inch (cm)		1.767 (4.488) ^(b)
Clad Fuel Length, inch (cm)		20.88 (53.04) ^(b)
Uranium Core Length, inch (cm)		20.45 (51.94)
Weight of Element U, lb (kg)		24.38 (11.06)
Uranium Composition, wt% U-235		1.25
wt% U-238		98.70
wt% U-234		0.01
wt% U-236		0.04
Inner Tube		
Zirconium Clad OD, inch (cm)		1.246 (3.165) ^(b)
Uranium OD, inch (cm)		1.166 (2.961)
Uranium ID, inch (cm)		0.490 (1.245)
Zirconium Clad ID, inch (cm)		0.440 (1.118) ^(b)
Clad Fuel Length, inch (cm)		20.82 (52.88) ^(b)
Uranium Core Length, inch (cm)		20.45 (51.94)
Weight of Element U, lb (kg)		12.19 (5.53)
Uranium Composition, wt% U-235		0.947
wt% U-238		99.00
wt% U-234		0.01
wt% U-236		0.04
Assembly Weight U, lb (kg)		36.57 (16.59)
Density Uranium	18.76 g/cc (c)	
Density Zirconium	6.4 g/cc	

(a) The N Reactor Safety Analysis Report, UNI-M-90 Addendum 1, February 28, 1978.

(Centimeter values were calculated from values in inches.) These are nominal values close to the midpoints of tolerance bounds.

(b) Manufacturing tolerances for the diameter of the clad uranium-metal MKIA fuel in inches are

Outer-tube OD	2.391 – 2.416	Length of outer tube	20.820 – 20.940
Outer-tube ID	1.754-1.779	Length of inner tube	20.760 – 20.880
Inner-tube OD	1.237-1.256		
Inner-tube ID	0.431-0.450		

(c) The U density is based on N Reactor fuel factor weights and the above dimensions. It was the most frequently used value for physics analyses.

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1.2.3.2 Spent-Fuel Tube Preparation

A special preliminary measurement was undertaken to assure that the spent-fuel elements all had, as nearly as possible, the same exposure. The spent-fuel elements used in the test were part of the May 23, 1974 reactor discharge. A gamma ray screening device ("weasel") was used to select discharged fuel elements in a narrow exposure range. The discharged fuel elements with similar exposures were set aside for the spent-fuel test. Sufficient fuel of near uniform exposure was selected to permit loading 91 fuel tubes. The average exposure for the spent fuel was 2865 MWd/t. (See Section 1.3.2 for details.)

The fuel elements were stored in the N Reactor fuel storage basin for approximately 1 year before the experiment was performed in April 1975. This allowed short-lived isotopes to decay.

Two spent MKIA fuel elements were placed into each fuel test tube. The loading process occurred under water. No stoppers were required for spent-fuel tubes since contamination on the surface of the fuel element was not a concern. The hydraulic lifting device and a chain hoist were used to move the fuel under water from the tube loading station to the test stand.

1.2.4 Experimental Procedure

Due to safety constraints, the test k_{eff} values were required to stay below 0.97. This constraint initiated a greater extrapolation to the critical state than is commonly used in approach-to-critical measurements. Using both approach-to-critical and pulsed-neutron measurements helped reduce experimental uncertainty in predicting criticality.

Neutronic data was acquired with fission chambers and a proportional counter. The detectors both tracked the approach-to-critical and also provided data for the pulsed-neutron decay traces. The detectors and sources and their accompanying electronics were standard equipment used by the (then) BNWL critical-mass laboratory.

Count rates were recorded and plotted to generate inverse multiplication curves as a function of test tubes in the test assembly. For more reactive loadings, pulsed-neutron data were obtained to estimate the subcriticality of the test configuration. The pulsed-neutron source was repetitively pulsed at up to 50 pulses per second and yielded about 10^9 neutrons/sec.

The test configuration was loaded with fuel test tubes in incremented steps. The loading proceeded from the center out in concentric rings as shown in Figures 11 and 12.

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Figure 11. Green Fuel Loading Map

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Figure 12. Spent-Fuel Loading Map

Count-rate data were acquired for 80 seconds for each loading. Background-corrected count rates were converted into inverse count rates and plotted to track neutron multiplication in the test configuration. Based on prior knowledge from other tests with 1.25 wt% ²³⁵U uranium metal tube-in-tube fuel (References 5, 6, and 7) and some supportive calculations, data acquisition started with 36 fuel tubes loaded. After each loading, a prediction of critical was made and half the fuel needed for critical was loaded.

During the measurements, detector #3, a fission chamber, occupied several different locations. During unloading of the test configurations, data points were obtained with detector #3 at the assembly center, to obtain information for spatial correction factors for the two detectors at the edge of the lattice plates. (See Figure 6) Detector #3 was in location "4" during the approach-to-critical loading with green fuel. During the loading of spent fuel, detector #3 was originally in location "5". For the spent-fuel measurements, the proportional counter in location "1" in Figure 6 was replaced with a fission chamber.

The detector #3 readings during unloading, with the center fuel column moved to the periphery, were very consistent with each other near the maximum loading, and indicated a predicted critical number of fuel columns significantly larger than extrapolated from detector #1 and #2 readings.

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1.3 Description of Material Data

1.3.1 Green (Unirradiated) Fuel

Each N Reactor green fuel uranium extrusion billet was analyzed for impurities and isotopic composition. Acceptable impurity levels from measurements are given in Table 2.

The MKIA fuel element was composed of an outer tube enriched to 1.250 ± 0.006 wt% ^{235}U in uranium and the inner tube enriched to 0.947 ± 0.007 wt% ^{235}U in uranium. Table 2 gives the isotopic composition for unirradiated uranium and the uranium impurities. These are typical values for MKIA fuel. The isotopic composition for the green fuel was obtained from the N Reactor Safety Analysis Report, UNI-M-90 Addendum 1, February 28, 1978.

N Reactor fuel was co-extruded out of 445-lb uranium billets with the zirconium cladding.

Table 2. Uranium Isotopic Composition and Impurities

MKIA Fuel Composition ⁸	Element	Fuel Impurities, PPM N Reactor 1972-1980
Outer Tube	H	2.
w/o U-238 98.70 ± 0.010	Be	10.
w/o U-236 0.04 ± 0.001	B	0.25
w/o U-235 1.25 ± 0.006	C	550.
w/o U-234 0.01 ± 0.001	Na	8.
	Mg	25.
Inner Tube	Al	800.
w/o U-238 99.00 ± 0.010	Si	124.
w/o U-236 0.04 ± 0.002	Cl	1.
w/o U-235 0.947 ± 0.007	Cr	65.
w/o U-234 0.01 ± 0.001	Mn	25.
	Fe	350.
	Co	1.
	Ni	100.
	Cu	75.
	Zr	65.
	Cd	0.25

1.3.2 Spent (Irradiated) Fuel Exposure Determination

N Reactor was a dual-purpose production reactor producing Pu of a certain quality and quantity for the weapons program and generating 860 MW of electric power with the steam byproduct. As a production reactor, the N Reactor had to be opened so that a uniform quality of Pu was produced. Cumulative exposure of fuel was closely monitored by measuring coolant flow and temperature for

⁸ Matsumoto, W. Y., *Burnup and Associated Analysis Results for N-Reactor Spent Fuel sections - Data Sheets*, HEDL-810541, Hanford Environmental Development Laboratories, Richland, Washington, 1986.

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each horizontal process tube, or pressure tube, which contained the fuel elements. Exposure was controlled by manipulating power level and charge-discharges. The tube powers were known to better than 1%^h. For each discharge, a record was kept of which tubes were involved, their total operating hours, and average kilowatts generated. Appendix A provides additional insights.

A typical core loading map is shown in Figure 13. Each square represents a pressure tube loaded with fuel elements. A pressure tube and its 34.8-ft-long fuel charge are identified by the grid location numbers. Most of the pressure tubes contain MKIV fuel assemblies. Figure 13 shows the spike ring as circles inside bigger squares. The spike or enrichment ring in N Reactor contained higher-enrichment fuel elements (such as MKIA), to help flatten the thermal neutron flux. The MKIA fuel elements used in the spent-fuel test assemblies were irradiated in these spike-fueled tubes. The 16 pressure tubes discharged on May 23, 1974, which provided fuel for the test, are shown in Figure 13 as black dots inside circles. Here a spike charge consisted of 4 MKIV fuel assemblies (2 at each end) and 15 MKIA fuel assemblies, called "loading class 625," which refers to the types of fuel loaded in the charged pressure tube.

Figure 13. N Reactor Front Face Discharge Pattern or Loading Map

The exposure on each fuel element was determined by using the axial power profile and the average tube power. Table 3 lists the 16 spike tube numbers and the average tube exposure received by the

^h Personal communication, Dr. W.D. Wittekind, N Reactor physicist (01/04/2006).

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fuel. The 3DNⁱ neutron-diffusion computer code was used in the past to calculate three-dimensional flux distributions in the N Reactor. Characteristic axial power distributions for typical process tube loadings were generated and documented^j. All 16 of the tubes selected for the subcritical assembly were of the same "loading class 625" and clustered in a specific region of the core; thus one axial power shape applies. This axial power profile was used to compute "power sharing," which represents the fraction of the total tube power generated in that particular fuel assembly, normalized to the number of assemblies for that loading class. The power shape is assumed to be the same as the integral exposure shape; the axial exposure distribution for each of the 16 tubes is therefore reconstructable. The assembly exposure distribution (for $16 \times 15 = 240$ MKIA assemblies) was created in a spreadsheet and then sorted. The number of assemblies taken from the available set was 182, which are sufficient to fill 91 tubes for the experiment. The 182 assemblies were taken from the low end of the discharge distribution, the average exposure for the 182 assemblies being 2865 MWd/t. See Appendix A for details. The value 2865 MWd/t is close to the 2740 MWd/t value that was postulated to yield a factor-of-2 difference in critical mass between spent and green fuel.

Table 3. Exposure in N Reactor of MKIA Fuel Elements Used in the Test Assemblies

Pressure Tube Number	MWd/Pressure Tube	Details of Determining Average Fuel Exposure
1945	940.23	Average weight ^(a) of MKIA fuel is 16.59 kg U per assembly. Average weight of MKIV fuel is 23.43 kg U per assembly. Process tube spike charge contains 15 MKIA and 4 MKIV fuel assemblies. Total uranium mass of a spike charge is 548.62 lbs or 248.85 kg MKIA.
2045	939.68	
1969	833.16	
1870	903.98	
2070	919.61	
2043	1019.24	Process tube average exposure is 2436.6 MWd/t of spike charge.
2044	908.03	
0564	846.45	
1644	879.74	
1745	945.01	MKIA fuel exposure in a spike charge is 1.25 times the tube average exposure, or 3042.9 MWd/t (details see Appendix A)
2445	905.00	
1645	937.64	t is short ton (2000 lb, 907.2 kg)
1545	940.11	
1544	888.99	
2343	995.57	Discharge date for MKIA fuel was May 23, 1974 Testing started April 1975
2245	918.16	
Average exposure 920.04 MWd/pressure tube		
(a) sometimes called "factor weight".		

ⁱ Burnside, R. J. and R. M. Wu, *3 DN Computer Code User's Manual*, UNI-M-182, UNC Nuclear Industries, Richland, Washington, 1985.

^j Schwinkendorf, K. N. and W. D. Wittekind, *K-East Basin Sludge Characterization Using Exposure-Adjusted Radioisotopic Source Terms for N Reactor Fuel*, HNF-8760, Rev. 1, Fluor Hanford, Inc., Richland, Washington, 2002.

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After reaching a certain product quality, fuel was discharged from select tubes of the 1003 pressure tubes. The fuel was cooled in the N Reactor spent-fuel pool before it was reprocessed. In chemical reprocessing, the Pu, U and Np were recovered while the rest of the residue became waste to be stored in Hanford's waste tanks. MKIV and MKIA fuel were processed in separate batches.

After the fuel was discharged, two approaches were used to obtain actinide and some fission product isotopics. First, fuel elements with well known power history were sectioned; the sections were dissolved and then analyzed chemically, radiolytically, and mass spectroscopically. Second, fuel was sent to the separations plant and dissolver samples were obtained and analyzed for actinides and selected fission products. None of the test fuel was analyzed, but other MKIA and MKIV were analyzed, covering the exposure range of the spent fuel in the test. These data served as a basis for validating different isotope production and depletion codes. The latest one is WIMSD-5B. Results of the validation effort are found in Appendix A. With the validated code, concentration of actinides and fission products for different exposures can be calculated.

1.3.3 Fuel Element Isotopic Determination

During the time of Hanford's production reactor operation, it was important to know the elemental and isotopic composition of the discharged fuel. A combination of radiochemical analysis and burnup or depletion calculations were used to generate documents called production tables. A production table would provide, for example, ^{235}U depletion, Pu buildup, Np creation, ^{240}Pu , and other isotopic inventories as a function of fuel exposure for the various fuel types.

At the time the experiment was performed, the analysts used computer codes such as DCODE^k or ORIGEN^l to predict composition of exposed fuel. Interpolation and extrapolation of experimental data employing supplemental computer analysis was used extensively. The experimental data were obtained from the PUREX reprocessing plant fuel-dissolver samples or from sectioned fuel elements. Slices of fuel elements, with well known exposure in MWd/t, would be dissolved and then analyzed using a complex state-of-the-art process.

Burnup, uranium and plutonium isotopic compositions and relative concentrations were obtained using the triple-spike isotope dilution mass spectrometry (IDMS) method with ^{148}Nd as the fission monitor. Neptunium-237 was determined using a double ion-exchange plus solvent-extraction chemical-separation procedure, coupled with absolute alpha counting and alpha spectrometry. Cesium-137 and ^{144}Ce were determined by direct high-resolution gamma spectrometry.

The analyzed sample data was carefully checked by rerunning samples through the process and by recalculating the computational sequences associated with the chemical and mass spectroscopic analysis. For data to be acceptable, reprocessed samples had to agree within expected limits with the original results for Pu isotopic composition and/or burnup values^m.

^k Thierer, I., *DCODE - A Fuel Design/analysis Code for the UNIVAC 1108*, DUN-5330, UNC Nuclear Industries, Incorporated, Richland, Washington, 1968.

^l Croff, A.C., *A User's Manual for the ORIGEN2 Computer Book*, ORNL TM-7175, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1980.

^m Matsumoto, W.Y., *Results of Analysis of Purex D-5 and A-3 Tank Samples to Characterize Key 14011 and Key 14306 N Reactor Spent Fuel Batches*, HEDL-810526, Westinghouse Hanford Company, Richland, Washington, October 1985.

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Example data for MKIA fuel at an exposure of 2000 MWd/t and the validation data used for the WIMSD-5B^o code will be found in Appendix A. A validated WIMSD-5B model is essential for generalizing exposed fuel concentration needed for the MCNP code^p to compute subcritical configuration reactivities.

As can be seen, isotopic analysis and associated calculations were carried out multiple times to obtain mutually consistent results. In Appendix A, Table A-1 contains the actinide concentrations as determined prior to the experiment, matching experimental data in Reference 9. For example, the numeric values expressed as grams/ton of U were generated for an exposure of 2000 MWd/t and were incorporated into production tables.

In more recent times, the WIMS^{qr} code has been employed successfully to calculate burnup details that matched production table results.

1.3.4 Basin Water

The water in the N Reactor Spent Fuel Pool was clean, filtered water. The contamination levels were controlled by regulating the water throughput. The experiment required clear water conditions to be able to observe the loading and manipulating of equipment. Data on contaminants in water was scarce. No water data were acquired during the test. An example of such information for N Reactor basin is found in Table 4.

No records of water temperatures could be located. However, based on other information, the water in the basin in April was assumed to be 10 – 15°C.

Table 4. N Reactor Spent Fuel Basin Coolant Radionuclide Concentration

	Radionuclide	Activity (pCi/L)	Half-life (sec)	Decay C (sec^-1)	Num Den (a/b-cm)	At. Weight (g/mole)	Mass Den (g/cm^3)
	H-3	23000	3.881E+08	1.786E-09	4.765E-16	3.016049	2.387E-15
	Mn-54	270000	2.697E+07	2.570E-08	3.888E-16	53.940359	3.482E-14
	Fe-59	96000	3.846E+06	1.802E-07	1.971E-17	58.934875	1.929E-15
	Co-60	53000	1.663E+08	4.167E-09	4.706E-16	59.933817	4.683E-14
	Sr-89	110000	4.365E+06	1.588E-07	2.563E-17	88.907451	3.784E-15
	Sr-90	56000	9.183E+08	7.548E-10	2.745E-15	89.907738	4.098E-13
wtf	Zr-95	23669	5.531E+06	1.253E-07	6.988E-18	94.908043	1.101E-15
0.5	Nb-95	43331	3.021E+06	2.294E-07	6.989E-18	94.906836	1.101E-15

^a Matsumoto, W.Y., Burnup and Associated Analysis Results for N Reactor Spent Fuel Section-Data Sheets, HEDL-810541, Westinghouse Hanford Company, Richland, Washington, February 1986.

^o Schwinkendorf, K.N., *Software Certification Report for the WIMSD-5B Computer Code*, HNF-28910, Rev. 0, Richland, Washington, 2006.

^p MCNP is a trademark of the Regents of the University of California, Los Alamos National Laboratory.

^q *WIMSD5B – A Neutronics Code for Standard Lattice Physics Analysis*, ANSWERS Software Service, AEA Technology, June 1997.

^r Halsall, M.J. and C. J. Taubman, *The '1986' WIMS Nuclear Data Library*, Reactor Physics Division, AEE Winfrith, 1986.

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	Radionuclide	Activity (pCi/L)	Half-life (sec)	Decay C (sec ⁻¹)	Num Den (a/b-cm)	At. Weight (g/mole)	Mass Den (g/cm ³)
1	Zr-95+Nb-95	67000			1.398E-17		2.203E-15
	Ru-103	3900	3.393E+06	2.043E-07	7.063E-19	102.906324	1.207E-16
	Ru-106	9900	3.219E+07	2.153E-08	1.701E-17	105.907329	2.991E-15
	Cs-134	41000	6.516E+07	1.064E-08	1.426E-16	133.906718	3.171E-14
	Cs-137	190000	9.521E+08	7.280E-10	9.656E-15	136.907089	2.195E-12
wtf	Ba-140	349	1.102E+06	6.292E-07	2.052E-20	139.910605	4.767E-18
0.5	La-140	2651	1.450E+05	4.781E-06	2.052E-20	139.909478	4.767E-18
1	Ba-140+La-140	3000			4.103E-20		9.533E-18
	Ce-141	6600	2.808E+06	2.468E-07	9.893E-19	140.908276	2.315E-16
	Ce-144	100000	2.459E+07	2.819E-08	1.313E-16	143.913647	3.137E-14
	Pu-238	58	2.768E+09	2.505E-10	8.568E-18	238.049555	3.387E-15
wtf	Pu-239	243	7.605E+11	9.114E-13	9.872E-15	239.052163	3.919E-12
0.05	Pu-240	47	2.070E+11	3.348E-12	5.174E-16	240.053814	2.063E-13
0.0526316	Pu-239+Pu-240	290			1.039E-14		4.125E-12

Radionuclides were introduced into the spent-fuel coolant by the following processes:

- Discharge of fuel elements with failed cladding
- Discharge of reactor coolant during charge-discharge operation
- Fuel breakage occurring during charge-discharge and fuel handling operations
- Erosion of crud deposits from fuel element surfaces
- Surface erosion of materials and equipment taken from reactor systems

Table 4 provides the activity and concentrations, in g/cm³ of radionuclides dissolved or suspended in the spent-fuel storage basin cooling water under normal operating conditions. The data is from the N Reactor Safety Analysis Report, UNI-M-90 Addendum 1, February 28, 1978.

The spent-fuel coolant is replaced at an average rate of 800 gallons per minute. To maintain clarity basin water during fuel handling or special operations, this replacement rate is increased to 1,000 to 1,500 gallons per minute for periods of up to 3 days. The test assembly was close to the water inlet, see Figure 1.

During the testing, the concentrations would have been less than shown in Table 4. N Reactor did not use boron or gadolinium in reactor coolants.

Since the primary coolant was not borated, boron concentration would have been negligible and representative of potable Columbia river water levels. The filters in use at the time of the experiment were present to remove any particles. The radioactive contaminants are too dilute to impact experimental results. Therefore, water contaminants will not be considered further.

1.3.5 Zircaloy Cladding

Impurity levels of Zircaloy cladding are given in Table 5. Tables 6 and 7 provide information on composition and density of aluminum alloy 6061 and the composition of secondary materials such as 304 L stainless, wet concrete, and neoprene.

Table 5. N Reactor Zirconium Cladding Composition Including Impurities

Element	Current Model (ppm) by weight
H	25
Be	974
B	0.5
C	275
N	80
Na	20
Mg	20
Al	75
Si	100
Ti	50
V	50
Cr*	1000
Mn	50
Fe*	1350
Co	10
Ni*	550
Cu	50
Zr*	98,1400
Mo	50
Cd	0.5
Sn*	14,500
Hf	200
W	50
Pb	100
U	2.5

Table 5 contains the zirconium impurities as well as alloying elements identified by an *.

1.3.6 Secondary Materials

Tables 6 and 7 provide information on composition and density of the alloy 6061 and the composition of secondary materials 304L stainless steel, wet concrete, and neoprene. These compositions were obtained from different handbooks; they were not derived from measurements on material that are part of the experiment.

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Table 6. Measured Densities and Standard Compositions of Aluminum Alloy 6061 (Reference 5)

Element	Wt%
6061 Aluminum (density – 2.69 g/cm ³) ^s	
Si	0.40-0.80 (0.6 nominal)
Fe	0.7 (maximum)
Cu	0.15-0.40 (0.25 nominal)
Mn	0.15 (maximum)
Mg	0.8-1.2 (1.0 nominal)
Cr	0.04-0.35 (0.2 nominal)
Zn	0.25 (maximum)
Ti	0.15 (maximum)
Al	Remainder (96.00-98.61)

Table 7. Secondary Material Compositions

Fractional Density For 304L Stainless ^t		Fractional Density for Concrete ^{uu}		Neoprene Composition ^v			
Element	Density g/cm ³	Element	Density g/cm ³	Neoprene (C ₄ H ₅ Cl) _n	density= 1.23 g/cm ³		
C	0.0023	H	0.0230	I	wt%	%	Density
Si	0.0587	C	0.0023	C	54.27	40	0.03348
P	0.0035	O	1.2201	H	5.69	50	0.04185
S	0.0023	Na	0.0369	Cl	40.04	10	0.00837
Cr	1.5660	Mg	0.0046				
Mn	0.1566	Al	0.0784				
Fe	5.1009	Si	0.7773				
Ni	0.9396	K	0.0299				
		Ca	0.1014				
		Fe	0.0323				
Total Density 7.8300		Total Density 2.3062					

^s Alcoa Aluminum Handbook, Aluminum Company of America, pp. 46-50.^t CSB Detailed Design Radiation Protection and Shielding, Westinghouse Hanford Company, Richland, Washington, 1996, pp. 47-48.^u Typical wet Hanford concrete.^v Brady, G. S., Materials Handbook, McGraw-Hill Book Company, New York, New York, 1971.

1.4 Supplemental Experimental Data and Subcritical Determination Methodology

1.4.1 Approach-to-Critical Data Evaluation

The approach-to-critical discussion, data, and graphics are found in Appendix E.

1.4.2 Pulsed-Neutron Measurements^w

Pulsed-neutron data was acquired at higher mass loadings with the same detectors at the same loading configuration as the approach-to-critical data. The main data was acquired with a detector in location "1." This detector consistently produced higher count rates. A schematic of the pulsed-neutron-source data acquisition system is shown in Figure 14. Additional details are provided in the appendices.

Figure 14. Pulsed-Neutron Source and Data-Acquisition System

^w Further information on Pulsed-Neutron Source Theory and measurement techniques are found in Appendix F.

Raw pulsed-neutron data, processed data, and the pulsed-neutron methodology are explained in Appendix F. Tables 8 and 9 contain evaluated pulsed-neutron data for green and exposed fuel loadings. The results in the two tables can be generated from the pulsed-neutron plots in Appendix F. Counts per channel are plotted against time in seconds following a burst. Processing the data yields reactivity as computed by the Gozani^{xy} and the Garelis-Russell formalisms^z, giving an average value of $k_{\text{eff}}^{\text{aa}}$. In addition the prompt-neutron decay rate is calculated. Given a value for β and ℓ , the other parameters can be determined.

The delayed-neutron fraction β for thermal systems needed for pulsed-neutron data evaluation was obtained from *Introduction to Nuclear Reactor Theory*^{bb}. The total delayed-neutron fraction β for ^{235}U was 0.0065 and for ^{239}Pu was 0.0021. Using the ^{239}Pu concentration for 2865 MWd/t from production tables in Appendix A, a mass-weighted β value was computed to be 0.0056 for irradiated fuel.

A summary of the processed pulsed-neutron information is captured in Figure 16 where prompt-neutron decay rate, α , is plotted against the number of fuel columns in the test stand for unirradiated and irradiated MKIA fuel. The prompt-neutron projection to critical in Figure 15 indicates that the mass difference between green and spent fuel is approximately a factor of 2. Differences between a fuel tube in the center location and detector #3 in the center are apparent.

The linear relationship between α and the number of fuel columns is illustrated. For the unirradiated fuel, the loadings with fuel in the center are more reactive than those with the center fuel tube removed and replaced with detector #3. The same observation is made for spent fuel. Again the scarcity of the experimental points is a problem.

An experimentally determined correction factor for the k_{eff} value was established by measuring pulsed-neutron data for loadings with and without detectors (#1) at the edge of and (#3) at the center of the test configuration. The information derived is plotted in Figure 16 as the " $-\ln k_{\text{eff}}$ " against tubes loaded. The correction factor is constructed from the ratio of k_{eff} readings associated with the center detector to the k_{eff} value corresponding to the readings at the edge, 21 inches from the center of the lattice plate. The k_{eff} values were obtained from Tables 8 and 9. The correction-factor curve shows that as the loading approaches criticality the correction factor approaches 1.0.

Corrected average k_{eff} values are listed in Tables 8 and 9 for different loading configurations. These k_{eff} values have been corrected for the spatial dependency. Detectors in different regions of the test setup will see variations in prompt and delayed fission distributions, such as at the

^x Gozani, T., *A Modified Procedure for the Evaluation of Pulsed Source Experiments in Subcritical Reactors*, Nukleonik, 1962, Vol. 4, p. 348.

^y Gozani, T., P. DeMarmels, T. Hurlimann and H. Winkler, *On the Modified Pulsed Source Techniques*, IAEA, Karlsruhe Symposium on Pulsed Neutron Research, 1965.

^z Garelis, E. and J. L. Russell, Jr., *Theory of Pulsed Neutron Source Measurements*, Nuclear Science and Engineering, 1963, Vol. 16, pp. 263-270.

^{aa} k_{eff} is based on literature discussions in *Theory of Pulsed Neutron Source Measurements, A Modified Procedure for the Evaluation of Pulsed Source Experiments in Subcritical Reactors, and On the Modified Pulsed Source Techniques*. The Garelis-Russell Formalisms and the Gozani method bound the calculated value.

^{bb} Lamarsh, J. R., *Introduction to Nuclear Reactor Theory*, Addison Wesley Publishing Company Inc., 1966, p. 100.

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edge of the configuration and in the center. Assuming the distributions are the same in the center region, correction factors can be established by recording neutrons in the center of the test configuration compared to at the edge. Therefore, during unloading, detector #3 in the center position was obtaining data as well as detector #1.

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Table 8. Experimental Data on 3.2 – Inch Pitch Hexagonal Lattice of Unirradiated MKIA Fuel in Water

Assembly Loading (Fuel Columns) (1)	Prompt Neutron Decay Rate (sec ⁻¹)	$k\beta/\lambda$ (sec ⁻¹)	Reactivity ⁽²⁾		$k_{eff}^{(3)}$ (avg)	Corrected $k_{eff}^{(4)}$ (avg)
			ρ_{GR} (\$)	ρ_{GO} (\$)		
0	4058 ± 32	---	---	---	---	---
7	3954 ± 40	---	---	---	---	---
19	3756 ± 31	---	---	---	---	---
37	2139 ± 40	36.27	-57.98	-48.00	0.74	---
43	1824 ± 14	58.25	-30.04	-27.82	0.84	---
43	1825 ± 17	56.30	-31.41	-29.74	0.83	---
49	1546 ± 11	68.43	-21.60	-21.38	0.88	---
49 ⁽⁵⁾	1440 ± 27	65.85	-20.88	-20.52	0.88	---
51	1439 ± 11	73.56	-18.56	-18.93	0.89	0.92
54	1257 ± 6	87.09	-13.43	-13.92	0.92	0.95
54	1229 ± 7	73.45	-15.29	-16.22	0.91	0.94
78.2	---	---	---	---	---	1.0065
80.4 ⁽⁶⁾	---	---	---	---	---	1.0065
54 ⁽⁶⁾	1414 ± 15	82.51	-16.14	-16.57	0.90	0.93
54 ⁽⁶⁾⁽⁷⁾	1442 ± 9	129.33	-10.15	-10.86	0.94	0.94
48 ⁽⁶⁾	1763 ± 11	57.95	-29.43	-29.74	0.84	---
42 ⁽⁶⁾	2052 ± 29	40.05	-50.24	-45.73	0.77	---
36 ⁽⁶⁾	2391 ± 54	22.75	-104.11	-84.85	0.62	---

(1) Each fuel column is two fuel elements in length.

(2) Reactivity determined by Garelis-Russell and Gozani methods.

(3) Average of values determined from Garelis-Russell and Gozani methods assuming $\beta_{eff} = 0.0065$.

(4) Corrected for kinetic distortion and assuming $\beta_{eff} = 0.0065$.

(5) Data accumulated using Detector 2 (fission chamber) on opposite side of assembly from Detector 1.

(6) Center fuel column placed in outer ring of fuel and Detector 3 placed in center of assembly.

(7) Data accumulated using Detector 3 (fission chamber) in center of assembly.

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**Table 9. Experimental Data on 3.2-inch Pitched Hexagonal Lattice of
Irradiated MKIA Fuel in Water**

Assembly Loading (Fuel Columns) (1)	Prompt Neutron Decay Rate (sec ⁻¹)	k_B/λ (sec ⁻¹)	Reactivity ⁽²⁾		$k_{eff}^{(3)}$ (avg)	Corrected $k_{eff}^{(4)}$ (avg)
			ρ_{GR} (\$)	ρ_{GO} (\$)		
67	1676 ± 89	24.10	-68.56	-70.94	0.72	0.76
73	1574 ± 24	68.82	-21.87	-24.51	0.88	0.91
79	1495 ± 42	79.33	-17.84	-22.49	0.90	0.92
85	1342 ± 25	93.44	-13.37	-15.92	0.92	0.94
91	1261 ± 21	108.02	-10.67	-12.59	0.94	0.96
158.5	---	---	---	---	---	1.0056
150.9 ⁽⁵⁾	---	---	---	---	---	1.0056
90 ⁽⁵⁾	1356 ± 33	99.62	-12.61	-14.44	0.93	0.95
90 ⁽⁵⁾⁽⁶⁾	1361 ± 6	131.59	-9.34	-10.35	0.95	0.95
78 ⁽⁵⁾	1626 ± 30	96.52	-15.80	-19.60	0.91	0.93
78 ⁽⁵⁾⁽⁶⁾	1634 ± 13	120.94	-12.51	-14.33	0.93	0.93
60 ⁽⁵⁾	2047 ± 80	54.41	-36.63	-43.48	0.82	0.90
54 ⁽⁵⁾⁽⁶⁾	2285 ± 21	102.78	-21.23	-25.47	0.88	0.88

- (1) Each fuel column is two fuel elements in length.
- (2) Reactivity determined by Garelis-Russell and Gozani methods.
- (3) Average of values determined from Garelis-Russell and Gozani methods assuming $\beta_{eff} = 0.0056$.
- (4) Corrected for kinetic distortion and assuming $\beta_{eff} = 0.0056$.
- (5) Center fuel column placed in outer ring of fuel and Detector 3 placed in center of assembly.
- (6) Data accumulated using Detector 3 (fission chamber) in center of assembly.

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Figure 15. Prompt-Neutron Decay Rate as Function of Fuel Loading

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Figure 16. Spatial Correction Factors

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Table 10 captures the impacts of the correction factors at different test-assembly loadings for both irradiated and unirradiated fuel.

Table 10. Correction Factors for Kinetic Distortion

Assembly Loading (Fuel Columns)	Measured $k_{eff}^{(a)}$		Measured Correction Factors Detector 1 Location (2)
	Assembly Center	Assembly Edge ^(b)	
90 irradiated	0.948	0.930	1.019
78 irradiated	0.930	0.908	1.024
60 irradiated	---	0.815	1.102
54 irradiated	0.884	---	1.198
54 unirradiated	0.936	0.904	1.035

(a) $\beta = 0.0056$ used for irradiated fuel and 0.0065 for unirradiated fuel

(b) Twenty-one inches horizontally from assembly center

Supplemental experimental data with water-moderated MKIA fuel tubes at different lattice pitches is contained in References 5, 6, and 7 (see Appendix A). The results from these measurements substantiate the extrapolated critical mass for MKIA fuel at the lattice pitch of 3.2 inch.

Table 11. Cases Selected as Potential Benchmarks ^a

Unirradiated Case Identification	Fuel # of tubes	Prompt Neutron Decay Rate (Sec ⁻¹)	$k \beta/\ell$ (Sec ⁻¹)	ρ_{GR} (\$)	ρ_{GP} (\$)	k_{eff} (avg)	Corrected k_{eff} (avg)
1G Loading	51	1439±11	73.56	-18.56	-18.93	0.89	0.92
2G Loading	54	1243±09	80.27	-14.36	-15.07	0.915	0.945
3G Unloading	54	1414±15	82.51	-16.14	-16.57	0.90	0.93
Irradiated Case Identification	Fuel # of tubes						
1s Loading	85	1342±25	93.44	-13.37	-15.92	0.92	0.94
2s Loading	91	1261±21	108.02	-10.67	-12.59	0.94	0.96
3s Unloading	90	1356±33	99.62	-12.61	-14.44	0.93	0.95
4s Unloading	78	1626±30	96.52	-15.80	-19.60	0.91	0.93

^aCases extracted from data in Tables 8 and 9.

Uncertainties associated with the corrected k_{eff} values are found in Section 2.3.

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2.0 EVALUATION OF EXPERIMENTAL DATA

2.1 Critical Loading Predictions

The unirradiated and irradiated fuel experiments were performed with the same type of fuel, MKIA. The unirradiated and irradiated MKIA fuel elements of the same length were used in the tests. Both the approach-to-critical and the pulsed-neutron subcritical data indicate that the spent fuel is less reactive than the green fuel. The mass difference between spent and green fuels varies as a function of subcriticality. Approach-to-critical data of the number of tubes extrapolated to critical for green fuel is consistent with measurements of a 3.1 inch lattice of similar type fuel (References 5, 6 and 7). Unirradiated 1.25-wt%-enriched tubular fuel moderated and reflected by water had an extrapolated critical mass between 60 and 66 fuel tubes.

Based on the plots in Figure 15, the prompt critical condition from pulsed-neutron data for unirradiated MKIA fuel in the test assembly is approximately 78 fuel tubes. For irradiated fuel, the prompt critical value is approximately 148 fuel tubes. With the detector in the center position, prompt criticality is estimated to occur for 80 unirradiated fuel tubes and for 150 irradiated fuel tubes. The approach-to-critical data provides extrapolations to critical, which are approximately different by a factor of two. These results confirm the original objective of the test, which was to demonstrate that MKIA fuel irradiated to 2865 MWd/t has a critical mass approximately twice as large as for unirradiated fuel.

2.2 Sensitivity Computations

The MKIA incoming uranium extrusion billets, and the co-extruded fuel materials for a production reactor, such as N Reactor, were carefully measured to assure they were within the specified manufacturing and material composition tolerances (maximum allowed values, or bounds). The fuel used in the experiment had passed the quality-control dimensional acceptance testing in the fuel fabrication and material preparation facility.

The model described in Appendix C was tested for sensitivity to changes in various parameters, such as fuel dimensions, uranium density, zirconium density, aluminum density, fuel exposure, fuel enrichment, bottom reflector variations, and water density. The following sections discuss the sensitivity calculations. The k_{eff} calculations were performed using MCNP4C with 1,000 neutron generations of 5,000 neutrons each. These calculations are based on unexposed fuel, and are assumed to be valid for exposed fuel as well because of the low exposures of N Reactor fuel.

The following sections contain tables of the results of the parameter perturbations that were used to obtain sensitivity coefficients. The Z-Test is performed for each perturbation result to evaluate the significance of the change. For statistically significant perturbations, the change in k_{eff} is evaluated in terms of reactivity coefficients and the physical uncertainty. The total uncertainty effect due to the significant contributors is determined by combining the results, yielding a total k_{eff} uncertainty of 8.50 mk^{cc} for irradiated fuel, and 2.19 mk for unirradiated MKIA fuel.

^{cc} mk = milli-k (10^{-3} k or Δk)

2.2.1 Changes in Fuel Dimensions

The MKIA fuel element tube dimensions were changed in the calculations to evaluate the effect on reactivity. An increase in the MKIA fuel element tube thickness corresponding to maximum thickness possible within MKIA tube diameter tolerances^{dd} resulted in a Δk change of 1.09 mk^{ee}, and a decrease in fuel tube thickness resulted in $\Delta k = -1.46$ mk. Statistical Z-Tests^{ffg} on each of these results compared to the base case show statistically discernable differences in the k_{eff} values that produced the respective reactivity coefficients, at the 95% confidence level. Reactivity coefficients for either increasing or decreasing each of the four radial dimensions (inner and outer radii for each of the inner and outer fuel elements) were calculated using a dimension change of either 0.05 or 0.1 cm. These changes are much larger than the manufacturing tolerances, and were selected in order to obtain statistically significant differences in k_{eff} , so that reliable reactivity coefficients could be defined. The reactivity calculated for each of the dimensional changes was scaled to the tolerances with the reactivity coefficients. These reactivity changes were then quadratically combined to obtain the total effect of variations equal to tolerances. The reactivity sensitivity to dimensional tolerance was evaluated using the formula:

$$\Delta k = \sqrt{\sum_{i=1}^4 (\alpha_i \delta r_i)^2} \quad (2.1)$$

The summation index, i , runs from 1 to 4, for each of the four radii. The δr_i term is the change in nominal dimensions to the tolerance. The sensitivity coefficient, α_i , is equal to

$$\alpha_i = \frac{\partial k}{\partial r_i} = \frac{1000 (k_2 - k_1)}{(r_2 - r_1)} (mk), \quad (2.2)$$

where subscript 1 refers to the base case (all dimensions nominal), and subscript 2 refers to the perturbed case. For minimum thickness, the perturbed radii were -0.05 cm for the outer radii and

^{dd} The radial thicknesses of the fuel elements were either maximized or minimized to ascertain the effect of changes in radial dimensions on reactivity, to within manufacturing tolerances. Both inner and outer fuel elements have manufacturing tolerances on the inner and outer diameters. For maximum thickness, the maximum outer diameter and minimum inner diameter were chosen, and for minimum thickness, the minimum outer diameter and maximum inner diameter were chosen. See Table 1 for the dimensional ranges allowed for MKIA fuel elements.

^{ee} 1 mk = 10^{-3} k_{eff}

^{ff} The Z-Test is a statistical test based on the standard normal distribution. It is used here to test the null hypothesis that two independent quantities (k_{eff} , calculated from MCNP4C simulations) are the same. If the absolute value of the difference between the two k_{eff} s, divided by the quadratically combined (i.e., pooled) standard deviation of the two values, exceeds a 95% confidence coefficient of 1.96, then the null hypothesis is rejected, and the two values are taken to be different with 95% confidence. If the value is less than 1.96, then there is sufficient overlap of the distributions that one cannot say with 95% confidence that the two values are different, and the null hypothesis is accepted. The test statistic is calculated using the formula:

$$Z = \frac{|k_2 - k_1|}{\sqrt{\sigma_1^2 + \sigma_2^2}}$$

^{gg} Mendenhall, W., Scheaffer, R. L., and Wackerly, D. D., *Mathematical Statistics With Applications*, Duxbury Press, Boston, Massachusetts, 1981.

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+0.10 cm for the inner radii. For maximum thickness, the perturbed radii were +0.05 cm for the outer radii and -0.10 cm for the inner radii. The " $k_{\text{eff}} \pm 2\sigma$ " results do not enter into either the calculation of sensitivity coefficients or the Z-Test results, but these columns merely illustrate the 95% confidence intervals for the MCNP4C k_{eff} results.

Case	k_{eff}	σ	$k_{\text{eff}} + 2\sigma$	$k_{\text{eff}} - 2\sigma$	Δk (mk)	Z-Test	Conclusion
Base Case	0.95896	0.00025	0.95946	0.95846			
Max thickness							
$r_1 = r_1 + 0.05$ cm	0.96023	0.00024	0.96071	0.95975	1.27	3.66	Significant
$r_2 = r_2 - 0.10$ cm	0.95482	0.00023	0.95528	0.95436	-4.14	12.19	Significant
$r_3 = r_3 + 0.05$ cm	0.95602	0.00025	0.95652	0.95552	-2.94	8.32	Significant
$r_4 = r_4 - 0.10$ cm	0.95766	0.00025	0.95816	0.95716	-1.30	3.68	Significant
Min thickness							
$r_1 = r_1 - 0.05$ cm	0.95482	0.00023	0.95528	0.95436	-4.14	12.19	Significant
$r_2 = r_2 + 0.10$ cm	0.95688	0.00025	0.95738	0.95638	-2.08	5.88	Significant
$r_3 = r_3 - 0.05$ cm	0.96068	0.00024	0.96116	0.96020	1.72	4.96	Significant
$r_4 = r_4 + 0.10$ cm	0.95989	0.00025	0.96039	0.95939	0.93	2.63	Significant

2.2.2 Changes in Uranium Density

Uranium metal densities encountered in safety analyses varied from 18.9 to 18.8 g/cm³. Uranium densities were obtained from measurements of incoming billets and subsequently from factor weights. The most commonly used value is 18.8 g/cm³. A $\pm 1\%$ (0.188 g/cm³) change in density would be very conservative. A 3% increase in uranium metal density resulted in $\Delta k = 1.74$ mk and a decrease of 2% produced $\Delta k = -1.70$ mk. One percent bounds most anticipated density fluctuations for uranium. Statistical Z-Tests were performed on each of these results compared to the base case. The +3%/-2% density perturbations do show different k_{eff} values at the 95% confidence level. For credible variations in uranium metal density ($\pm 1\%$) the uncertainty in k_{eff} was found by using the average sensitivity coefficient, with the result $\Delta k = 0.72$ mk.

Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
+3% / -2% in U density							
maximum density	0.96070	0.00023	0.96116	0.96024	1.74	5.12	Significant
nominal density	0.95896	0.00025	0.95946	0.95846			Base case
minimum density	0.95726	0.00025	0.95776	0.95676	-1.70	4.81	Significant

2.2.3 Changes in Zirconium Density

The zirconium clad density of 6.4 g/cm³ was varied by $\pm 10\%$ (0.64 g/cm³). A 10% increase in zirconium clad density resulted in $\Delta k = -0.57$ mk and a decrease of 10% produced $\Delta k = 0.00$ mk. A 1% variation bounds most anticipated density fluctuations for zirconium. Statistical Z-Tests on each of these results compared to the base case failed to show different k_{eff} values at the 95% confidence level. Therefore, the effect of a 1% change in clad density is assumed to be negligible.

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Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
$\pm 10\%$ in Zr density							
maximum density	0.95839	0.00024	0.95887	0.95791	-0.57	1.64	Not significant
nominal density	0.95896	0.00025	0.95946	0.95846			Base case
minimum density	0.95896	0.00025	0.95946	0.95846	0.00	0.00	Not significant

2.2.4 Changes in Aluminum 6061 Density

The aluminum density of 2.69 g/cm^3 (test tubes and lattice plates) was varied by $\pm 10\%$. A 10% increase in aluminum density resulted in $\Delta k = -1.32 \text{ mk}$, while a decrease of 10% in aluminum density produced $\Delta k = 0.35 \text{ mk}$. Statistical Z-Tests on each of these results compared to the base case indicates different k_{eff} values at the 95% confidence level (for the density increase). One percent bounds most anticipated density variations for aluminum (0.0269 g/cm^3). The resulting uncertainty in k_{eff} , using a sensitivity coefficient based on the density increase, is 0.132 mk .

Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
$\pm 10\%$ in Al density							
maximum density	0.95764	0.00024	0.95812	0.95716	-1.32	3.81	Significant
nominal density	0.95896	0.00025	0.95946	0.95846			Base case
minimum density	0.95931	0.00024	0.95979	0.95883	0.35	1.01	Not significant

2.2.5 Change in Water Density

Many criticality calculations make the assumption that water density is equal to 1.0 g/cm^3 . This is a good approximation in most cases, but to greater precision, the actual density of water under conditions of increased pressure (under 18 ft of water), and colder temperature, is higher if a greater number of significant figures is carried. Under 18 ft of water (about 1.5 atm) and assuming a lowest possible water temperature (above freezing), the water density is calculated to be approximately 1.00337 g/cm^3 . The change in k_{eff} with water density increased to 1.01 g/cm^3 was evaluated. The result was $\Delta k = 0.76 \text{ mk}$. A Statistical Z-Test on this result compared to the base case shows a significant difference in the k_{eff} value at the 95% confidence level. For a maximum credible increase in water density ($+0.00337 \text{ g/cm}^3$), the increase in k_{eff} is 0.26 mk .

Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
Change water density to 1.01 g/cm^3							
nom rho	0.95896	0.00025	0.95946	0.95846			Base case
min rho	0.95972	0.00025	0.96022	0.95922	0.76	2.15	Significant

The radioactive contaminants are too dilute to impact experimental results. It is assumed that low concentrations of radioactive contaminants are indicative of low concentrations of other impurities. Therefore, water contaminants will not be considered further.

2.2.6 Fuel Exposure Changes

The sensitivity of reactivity for the irradiated loading can be separated into two fundamental uncertainties. The first is the uncertainty in the average exposure, and the distribution of exposure across the set of fuel assemblies selected for inclusion in the experiment. Second, given the exposure distribution, there is no knowledge of which particular assemblies were loaded into which aluminum tubes, and then loaded into the experimental core. Different loading patterns of the fuel elements, each with a different individual exposure, and thus reactivity, will result in different real k_{eff} s. Appendix A provides supportive information on determination of fuel exposure.

An attempt was made to address both of these effects separately. The average exposure uncertainty was addressed by first assuming that all 182 fuel assemblies loading 91 aluminum tubes were chosen from the lower end of the discharge exposure of the 240 available fuel assemblies. This average exposure was 2865 MWd/ton, with an MCNP4C result of $k_{\text{eff}} = 0.94364 \pm 0.00023$. If the average exposure were 5% higher (3008 MWd/ton), the result was $k_{\text{eff}} = 0.93994 \pm 0.00024$. If the 182 fuel assemblies were chosen from the high end of the discharge exposure distribution, the average exposure of these 182 fuel assemblies would be 3326 MWd/ton (see Figure A-20 in Appendix A), with an MCNP4C result of $k_{\text{eff}} = 0.93530 \pm 0.00025$. The 5% perturbation in average exposure results in a reactivity coefficient of about -0.02583 mk/(MWd/ton). The real uncertainty in average exposure is physically bounded between 2865 and 3326 MWd/ton. If the actual average exposure was the average of these two bounding limits, the difference between the upper limit and the average would be $3326 - (3326 + 2865)/2 = 230 \text{ MWd/ton}^{\text{hh}}$. The bounding sensitivity of reactivity to the uncertainty in average exposure is therefore

$$(0.02583 \text{ mk / (MWd/ton)}) \times 230 \text{ MWd/ton} = 5.95 \text{ mk.}$$

The second contribution to reactivity uncertainty is due to the uncertainty in the loading arrangement. For the lower-limit average exposure of 2865 MWd/ton, these 182 fuel assemblies were ranked in order of exposure and then binned into three groups of approximately 60 fuel assemblies each, and the average exposure of each of these groups of assemblies was calculated.

There were three average exposures:

$$E_1 = 2173 \text{ MWd/ton}$$

$$E_2 = 3055 \text{ MWd/ton}$$

$$E_3 = 3350 \text{ MWd/ton}$$

Two models were used to bound the reactivity sensitivity to loading arrangement uncertainty. The first assumed that all 91 tubes were loaded as in Figure 17, with the lowest exposure fuel in

^{hh} The foot note refers to calculations carried out during planning for the experiment. Their computer calculations indicated that if the average fuel exposure for the irradiated fuel was 2740 MWd/ton the irradiated fuel critical mass would be twice that for unirradiated fuel at the same lattice pitch. Therefore, the higher exposure fuel from the discharged fuel assemblies was screened out and not used in the experiment. Therefore, the fuel used in this experiment averaged 2865 MWd/ton, close to the desired 2740 MWd/ton.

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the center, surrounded by the middle exposure fuel, with the highest exposure fuel around them. The numbers in the loading map correspond to the exposures, E_1 through E_3 . In the second model, the positions of the lowest and highest exposure tubes were exchanged, as shown in Figure 18. With the highest reactivity fuel in the center of the core, the k_{eff} would be expected to be the highest, and the result was $k_{\text{eff}} = 0.94910 \pm 0.00025$. With the loading arrangement in Figure 18, the result was $k_{\text{eff}} = 0.93779 \pm 0.00024$. The actual loading arrangement most likely was somewhere in between the two arrangements shown in the figures. The reactivity sensitivity is therefore taken to be half the reactivity difference between these two bounding cases. The result is a k_{eff} uncertainty of 5.66 mk.

Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
Min Exp. +5%:	0.93994	0.00024	0.94042	0.93946	-3.70	11.13	Significant
min Exposure:	0.94364	0.00023	0.94410	0.94318			Base case
max Exposure:	0.93530	0.00025	0.93580	0.93480	-8.34	24.55	Significant
load variation 1:	0.94910	0.00025	0.94960	0.94860	5.46	16.07	Significant
load variation 2:	0.93779	0.00024	0.93827	0.93731	-5.85	17.60	Significant

Figure 17. Irradiated Core Arranged with Ascending Exposures from Center of Core

```

      3 3 3 3 3 3
    3 2 2 2 2 2 3
  3 2 2 1 1 2 2 3
3 2 1 1 1 1 1 2 3
3 2 1 1 1 1 1 1 2 3
3 2 2 1 1 1 1 1 2 2 3
3 2 1 1 1 1 1 1 2 3
3 2 1 1 1 1 1 2 3
3 2 2 1 1 2 2 3
  3 2 2 2 2 3
    3 3 3 3 3

```

Figure 18. Irradiated Core Arranged with Descending Exposures from Center of Core

```

      1 1 1 1 1 1
    1 2 2 2 2 2 1
  1 2 2 3 3 2 2 1
3 1 2 3 3 3 3 2 1
3 1 2 3 3 3 3 3 2 1
3 1 2 2 3 3 3 3 2 2 1
3 1 2 3 3 3 3 3 2 1
3 1 2 3 3 3 3 2 1
3 1 2 2 3 3 2 2 1
  1 2 2 2 2 2 1
    1 1 1 1 1 1

```


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2.2.7 Fuel Enrichment Changes

The effect of enrichment on reactivity was evaluated. Enrichment was changed by ± 0.05 wt%. At the higher enrichment, $\Delta k = 11.1$ mk, and at the lower enrichment, $\Delta k = -12.8$ mk. These results are statistically significant. The as-manufactured enrichment tolerance for N Reactor fuel was ± 0.006 wt%, and so the sensitivity of reactivity to enrichment tolerance is calculated to be 1.44 mk.

Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
± 0.006 w/o ^{235}U in U							
max E:	0.97009	0.00024	0.97057	0.96961	11.13	32.12	Significant
nom E:	0.95896	0.00025	0.95946	0.95846			Base case
min E:	0.94615	0.00024	0.94663	0.94567	-12.81	36.96	Significant

2.2.8 Concrete Comparison

The iron and water contents in the concrete below the test assembly were varied. The composition approximately matches the concrete used. The effect of uncertainties in composition is negligible, since it is outside the fueled region and is separated from the fuel by 2 inches of water. Both iron and hydrogen are significant absorbers in the concrete mix. Even if the hydrogen content in the concrete is increased by a factor of 10, $\Delta k = -0.54$ mk. The iron content in the concrete was both increased and decreased by an order of magnitude. For the increased iron content case, the result was $\Delta k = -0.33$ mk. For the decreased iron case, the result was $\Delta k = -0.17$ mk. Z-Tests on these results compared to the base case show no significant differences in the k_{eff} values at the 95% confidence level. Because of the insignificant effects of these large variations, the effects of uncertainties in concrete composition are assumed to be negligible.

Case	k_{eff}	σ_{MC}	$k_{\text{eff}} + 2\sigma_{\text{MC}}$	$k_{\text{eff}} - 2\sigma_{\text{MC}}$	Δk (mk)	Z-Test	Conclusion
Hydrogen content in concrete increased by factor of 10							
Nominal H	0.95896	0.00025	0.95946	0.95846			Base case
Maximum H	0.95842	0.00026	0.95894	0.95790	-0.54	1.50	Not significant
Iron content in concrete changed by factor of 10							
Maximum Fe	0.95863	0.00025	0.95913	0.95813	-0.33	0.93	Not significant
Nominal Fe	0.95896	0.00025	0.95946	0.95846			Base case
Minimum Fe	0.95879	0.00025	0.95929	0.95829	-0.17	0.48	Not significant

2.2.9 Summary of Effects of Uncertainties

The sensitivity analyses have evaluated the k_{eff} uncertainties due to several input parameters. The approach was to calculate reactivity coefficients (i.e., sensitivities) using calculational changes that were sufficiently large to obtain statistically reliable reactivity coefficients (at the

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95% confidence level) using MCNP4C, and then to evaluate the k_{eff} uncertainty using these coefficients and the actual tolerances or uncertainties:

$$\Delta k = \frac{\partial k}{\partial P_i} \delta P_i = 1000 \frac{(k_2 - k_1)}{(P_2 - P_1)_i} \delta P_i \text{ (mk)} \quad (2.3)$$

In this equation, subscript 1 refers to the (unperturbed) base case, subscript 2 refers to the perturbed case, and subscript i refers to the i^{th} parameter evaluated. δP_i is the physical uncertainty or tolerance. In all cases, except for zirconium density variation and absorber content in the basin concrete, statistically significant reactivity coefficients were derived. Reactivity sensitivity to basin concrete composition variation was low because of the neutronic isolation of the basin floor from the fuel assembly. Zirconium is known to be a very poor neutron absorber, and changing its density by an order of magnitude more than the physical uncertainty failed to indicate any change in reactivity.

2.3 Pulsed-Neutron Experimental Uncertainty Estimation

Both the Gozani and Garelis-Russellⁱⁱ approaches for estimating the system reactivity rely on estimating the areas under the prompt and delayed portions of the pulsed-neutron response curve. The Gozani method, being an improvement on the Sjöstrand method, fits the prompt-neutron decay curve to estimate the fundamental mode. Assuming a fitted equation of the form:

$$N_p(t) = P e^{Bt}, \quad (2.4)$$

the fitted coefficients, P and B , represent the peak of the prompt-neutron decay curve extrapolated back to zero time, and the prompt-neutron decay rate, respectively. Taking the natural log of both sides of equation (2.4):

$$\ln N_p(t) = y(t) = \ln P + Bt \quad (2.5)$$

Linear least squares regression on equation (2.5) yields estimates of both P and B , as well as estimates on the standard error in both parameters. Care must be taken to select a portion of the prompt-neutron curve that starts just past the initial peak, but does not include too much delayed-neutron contribution. For n points considered, the regression parameters may be written:

$$B = \frac{n \sum_{i=1}^n t_i y_i - \sum_{i=1}^n t_i \sum_{i=1}^n y_i}{n \sum_{i=1}^n t_i^2 - \left(\sum_{i=1}^n t_i \right)^2} \quad (2.6)$$

$$\ln P = \bar{y} - B \bar{t}$$

ⁱⁱ Garelis, E. and J. L. Russell, Jr., *Theory of Pulsed Neutron Source Measurements*, Nuclear Science and Engineering, 1963, Vol. 16, pp. 263-270.

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The standard, or regression, errors on B and $\ln P$ may be written:

$$\begin{aligned}\delta B &= S \sqrt{c_{11}} \\ \delta(\ln P) &= S \sqrt{c_{00}}, \text{ where :} \\ S^2 &= \frac{1}{n-2} \sum_{i=1}^n (y_i - \hat{y}_i)^2 \\ &= \frac{1}{n-2} \sum_{i=1}^n (y_i - \ln P - B t_i)^2\end{aligned}\quad (2.7)$$

The $n-2$ term reflects that two parameters have been inferred from the regression analysis. It is also possible to show that the error in P is:

$$\delta P = P \left(e^{\delta(\ln P)} - 1 \right) \quad (2.8)$$

The variance terms, c_{00} and c_{11} , are:

$$\begin{aligned}c_{00} &= \frac{\sum_{i=1}^n t_i^2}{n \sum_{i=1}^n (t_i - \bar{t})^2} \\ c_{11} &= \frac{1}{\sum_{i=1}^n (t_i - \bar{t})^2}\end{aligned}\quad (2.9)$$

Another regression analysis is performed on the delayed-neutron portion of the response curve, with the result being an average value of N_d , and δN_d . Given the fitted, analytical shapes of the prompt and delayed-neutron curves, and their respective regression errors, it is now possible to derive the error in reactivity. First, the integrals are evaluated, and then inserted into the Gozani equation. The upper integration limit $1/R$, is simply the time interval between pulses, Δt :

$$\begin{aligned}\rho_s &= - \frac{\int_0^{1/R} P e^{Bt} dt}{\int_0^{1/R} N_d dt} \\ &= - \frac{\left(\frac{P}{B} \right) (e^{B \Delta t} - 1)}{N_d \Delta t} = \frac{P/B}{N_d \Delta t}\end{aligned}\quad (2.10)$$

The last equality in equation (2.10) is valid because the prompt-neutron exponential goes to zero for times large compared to the time required for the prompt-neutrons to decay.

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The Garelis-Russell equation assumes that the delayed-neutron baseline is subtracted off from the total neutron response curve, leaving only the prompt contribution. The integrals are therefore evaluated from the data directly. However, if the same fitted regression models are used for the prompt and delayed contributions, it may be shown that the two methods yield the same result for reactivity.

The Garelis-Russell equation is:

$$\int_0^{1/R} N_p e^{\frac{k\beta}{\ell} t} dt = \int_0^{1/R} N_p dt + \frac{N_d}{R} \quad (2.11)$$

Substituting in for the prompt-neutron curve and integrating:

$$\begin{aligned} \int_0^{1/R} P e^{Bt} e^{\frac{k\beta}{\ell} t} dt &= \int_0^{1/R} P e^{Bt} dt + \frac{N_d}{R} \\ \int_0^{1/R} P e^{\left(B + \frac{k\beta}{\ell}\right)t} dt &= \int_0^{1/R} P e^{Bt} dt + \frac{N_d}{R} \\ \frac{P}{B + k\beta/\ell} \left(e^{\left(B + \frac{k\beta}{\ell}\right)\Delta t} - 1 \right) &= \frac{P}{B} (e^{B\Delta t} - 1) + N_d \Delta t \\ \frac{-P}{B + k\beta/\ell} &= \frac{-P}{B} + N_d \Delta t \end{aligned} \quad (2.12)$$

Again, the exponentials go to zero, when evaluated at times as large as the full time interval between pulses. We now solve for the parameter $k\beta/\ell$:

$$\frac{k\beta}{\ell} = B \left(\frac{P}{P - BN_d \Delta t} - 1 \right) \quad (2.13)$$

The equation used for deducing reactivity in the Garelis-Russellⁱⁱ method is:

$$\rho_s = 1 + B / (k\beta/\ell) \quad (2.14)$$

(Note that B has a large, negative value.) Substituting equation (2.13) into equation (2.14), the result is:

ⁱⁱ Garelis, E. and J. L. Russell, Jr., *Theory of Pulsed Neutron Source Measurements*, Nuclear Science and Engineering, 1963, Vol. 16, pp. 263-270.

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$$\begin{aligned}
 \rho_s &= 1 + \frac{1}{\frac{1}{1 - \frac{BN_d \Delta t}{P}} - 1} \\
 &= 1 + \frac{1}{\frac{P}{P - BN_d \Delta t} - 1} \\
 &= 1 + \frac{P - BN_d \Delta t}{P - (P - BN_d \Delta t)} = 1 + \frac{P - BN_d \Delta t}{BN_d \Delta t} = \frac{P / B}{N_d \Delta t}
 \end{aligned} \tag{2.15}$$

This is the same result as equation (2.10).

Finding the error in k_{eff} is now a straightforward exercise in error propagation. Given the regression parameters, P , B , and N_d , and their associated regression errors, the error in reactivity is:

$$\delta \rho_s = \left[\left(\frac{\partial \rho_s}{\partial P} \delta P \right)^2 + \left(\frac{\partial \rho_s}{\partial B} \delta B \right)^2 + \left(\frac{\partial \rho_s}{\partial N_d} \delta N_d \right)^2 \right]^{1/2} \tag{2.16}$$

The three partial derivatives are analytically derived:

$$\begin{aligned}
 \frac{\partial \rho_s}{\partial P} &= \frac{1}{B N_d \Delta t} \\
 \frac{\partial \rho_s}{\partial B} &= \frac{-(P / B^2)}{N_d \Delta t} \\
 \frac{\partial \rho_s}{\partial N_d} &= \frac{-(P / B)}{N_d^2 \Delta t}
 \end{aligned} \tag{2.17}$$

Given that the reactivity in dollars is equal to $(k - 1) / (k\beta)$, the error relationship of reactivity to k_{eff} is:

$$\delta k = \beta k^2 (\delta \rho_s) \tag{2.18}$$

For pulse number 41075-4 (54 unexposed fuel tubes loaded), with a prompt-neutron decay constant of 1414 s^{-1} , the estimated experimental error is 2.71 mk. For pulse number 42275-3 (90 irradiated fuel tubes loaded), with a prompt-neutron decay constant of 1356 s^{-1} , the estimated experimental error is 9.65 mk.

Table 12 was imported from a spreadsheet where the pulsed-neutron numerical data were analyzed. The upper half of the table reproduces the original pulsed-neutron data analysis as documented in the original experimental report. The original writeup included experimental

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uncertainty values for the prompt neutron decay rate, but did not include error estimates for the reactivity values derived from either the Gozani or Garelis-Russell methodologies. Previously, it was accepted (with support from references) that the true value of k_{eff} is bracketed by the Gozani and Garelis-Russell methods. The raw pulsed-neutron data, once in electronic form, were reanalyzed for this report in order to perform the error propagation analysis all the way through to an estimate of the error on k_{eff} . Somewhat different values of prompt-neutron decay rate, $k\beta/\ell$, and reactivities were obtained upon reanalysis. This is believed to be the result of judgment used by different analysts in exactly which ranges of the neutron response curves are used in the construction of the regression parameters (prompt neutron decay rate, intercept, and delayed neutron baseline). The data reduction software used to output the response curves (which also contained a legend where these values were output) appears to be no longer available. In Tables 8 and 9, the older results are retained, and the new analysis results are used to estimate the errors. The kinetic distortion factor is calculated directly from the ratio of two k_{eff} values, determined from pulse-neutron data obtained from two different detector locations in the core, in the center and twenty-one inches horizontally from the center. The core loading is identical between these two cases, and the k_{eff} value obtained with the detector in the center of the core is considered to be the better estimate; the other k_{eff} value is essentially normalized to the value obtained from the center detector. Given the kinetic distortion factor:

$$f = \frac{k_c}{k_e}, \quad (2.19)$$

where k_c is the k_{eff} value obtained from the center detector, k_e is the k_{eff} value obtained from the edge detector, and the uncertainties in each of these numbers, the uncertainty in the ratio can be estimated from:

$$\delta f = \left[\left(\frac{\partial f}{\partial k_c} \delta k_c \right)^2 + \left(\frac{\partial f}{\partial k_e} \delta k_e \right)^2 \right]^{1/2} \quad (2.20)$$

The partial derivatives are:

$$\frac{\partial f}{\partial k_c} = \frac{1}{k_e}, \quad \frac{\partial f}{\partial k_e} = -\frac{k_c}{k_e^2} \quad (2.21)$$

The following table was copied from a spreadsheet where this error analysis was performed on three of the kinetic distortion factors listed in Table 10.

Kinetic Distortion Factor Error Analysis

90 Irradiated	Value	uncertainty	value-err	value+err
k_c =	0.948	0.0013		
k_e =	0.93	0.0065		
f_KD =	1.0194	0.0073	1.0121	1.0266
78 Irradiated				
k_c =	0.930	0.0022		

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$k_e =$	0.908	0.0114		
$f_{KD} =$	1.0242	0.0130	1.0112	1.0373
54 Unirradiated				
$k_c =$	0.936	0.0011		
$k_e =$	0.904	0.0017		
$f_{KD} =$	1.0354	0.0022	1.0332	1.0376

These new error estimates are used in the final Z-Tests of measured versus calculated k_{eff} s, to determine whether there is sufficient experimental and calculational error to state that the calculational results are correct to within these errors.

Plots of the prompt neutron decay rate versus the number of fuel tubes added (for both loading and unloading, and for both green and exposed fuel) were also created and extrapolated to zero prompt neutron decay rate (i.e., prompt critical). The predictions of number of tubes required to attain the prompt critical state are consistent with the previous analysis, even though the individual plotted points (decay rates from individual response curves) are slightly different.

Figure 19. Prompt Decay Rate vs. Fuel Loading

Table 12. Analysis of Pulsed-Neutron Experiment Uncertainties

Beta	Old Analysis Sequence	Number of Fuel Tubes	Pulse	alpha z	Error	kb/l	Reactivity		rho Go	Error	k-effective	k_GR	Error (mk)
							Rho_GR	Error					
0.0065	Loading	51	4975-1	1439	11	73.56	-18.5623		-18.93		0.8923		
	Loading	54	41075-3	1257	6	87.09	-13.4333		-13.92		0.9197		
	Loading	54	41075-1	1229	7	73.45	-15.7325		-16.22		0.9072		
	Loading	54	average	1243	9	80.27	-14.4852		-15.07		0.9139		
	Unloading	54	41075-4	1414	15	82.51	-16.1373		-16.57		0.9051		
	Unloading	54	41075-5	1442	9	129.33	-10.1498		-10.86		0.9381		
0.0056	Loading	85	42275-1	1342	25	93.44	-13.3622		-15.92		0.9304		
	Loading	91	42275-2	1261	21	108.02	-10.6738		-12.59		0.9436		
	Unloading	90	42275-3	1356	33	99.62	-12.6117		-14.44		0.9340		
	Unloading	90	42375-1	1361	6	131.59	-9.34273		-10.35		0.9503		
	Unloading	78	42375-2	1626	30	96.52	-15.8462		-19.6		0.9185		
New Analysis	Sequence	Number of Fuel Tubes	Pulse	alpha z	Error	kb/l	Reactivity		rho Go	Error	k-effective	k_GR	Error (mk)
							Rho_GR	Error					
							-17.2853	0.162	-18.935	1.357	0.8990		0.851
							-11.9019	0.179	-14.0404	1.031	0.9282		1.004
							-13.9	0.245	-16.107	1.194	0.9171		1.340
0.0058	Loading	54	average	1041.19	6.013	75.484	-12.7936	0.212	-15.0737	1.112	0.9232		1.172
	Unloading	54	41075-4	1313.92	12.439	85.710	-14.3298	0.320	-15.1812	0.520	0.9148		1.739
	Unloading	54	41075-5	1391.67	11.223	126.336	-10.0157	0.207	-10.7843	0.309	0.9389		1.188
	Loading	85	42275-1	1136.02	30.256	84.301	-12.4757	0.809	-15.115	2.631	0.9325		4.083
	Loading	91	42275-2	1043.54	25.906	96.775	-9.7832	0.591	-12.417	1.656	0.9463		3.067
0.0065	Unloading	90	42275-3	1221.26	27.526	95.170	-11.8324	0.802	-14.8622	2.577	0.9358		4.076
	Unloading	90	42375-1	1148.45	9.546	119.643	-8.5990	0.181	-10.5934	0.479	0.9525		0.954
	Unloading	78	42375-2	1410.34	28.777	79.147	-16.8193	0.959	-20.1571	4.780	0.9111		4.617

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k Go	Error (mk)	avg-keff	Error (mk)	kinetic DF	Corrected	
0.8904		0.8914		1.035	0.9226	
0.9170		0.9184		1.035	0.9505	
0.9046		0.9059		1.035	0.9376	
0.9108		0.9123		1.035	0.9443	
0.9028		0.9039		1.035	0.9356	
0.9341		0.9361		1.000	0.9361	
0.9181		0.9243		1.021	0.9437	
0.9341		0.9389		1.018	0.9558	
0.9252		0.9296		1.019	0.9473	
0.9452		0.9477		1.000	0.9477	
0.9011		0.9098		1.024	0.9316	
k Go	Error (mk)	Avg-keff	Error (mk)	kinetic DF	Corrected	Error (mk)
0.8904	6.994	0.8947	3.523	1.035	0.9260	4.187
0.9164	5.627	0.9223	2.858	1.035	0.9546	3.641
0.9052	6.360	0.9112	3.250	1.035	0.9431	3.964
0.9108	5.993	0.9170	3.053	1.035	0.9491	3.800
0.9102	2.802	0.9125	1.649	1.035	0.9444	2.706
0.9345	1.754	0.9367	1.059	1.000	0.9367	2.337
0.9194	12.898	0.9260	6.764	1.021	0.9454	14.123
0.9328	8.360	0.9396	4.452	1.018	0.9565	8.296
0.9206	12.667	0.9282	6.653	1.019	0.9458	9.653
0.9421	2.463	0.9473	1.321	1.000	0.9473	7.007
0.8953	22.223	0.9032	11.349	1.024	0.9249	16.742

Error Analysis for Experimental k-effs						
Unirradiated:	Fuel Columns	Alpha z	delta-a z	k*beta/l	rho - GR	rho - GO
					k - GR	k - GO
						k-avg

		(s ⁻¹)			(\$)	(\$)		
	51 Loading	1439	11	73.56	-18.56	-18.93	0.89234	0.89044
	54 Loading	1257	6	87.09	-13.43	-13.92	0.91970	0.91703
	54 Loading	1229	7	73.45	-15.73	-16.22	0.90723	0.90463
	54 Loading	1243	9	80.27	-14.49	-15.07	0.91395	0.91078
	54 Unloading	1414	15	82.51	-16.14	-16.57	0.90507	0.90277
	beta =	0.0065						
Irradiated:	85 Loading	1342	25	93.440	-13.37	-15.92	0.93034	0.91815
	91 Loading	1261	21	108.020	-10.67	-12.59	0.94362	0.93414
	90 Unloading	1356	33	99.620	-12.61	-14.44	0.93404	0.92519
	78 Unloading	1626	30	96.520	-15.8	-19.6	0.91871	0.90110
	beta =	0.0056						

	Uncertainty (mk)								
k-corrected	(k_GR - k_avg)	Error (mk)		k-MCNP4C	Sig-MCNP	Error (abs)	Z_Test		comment
0.9226	0.950	4.187		0.93903	0.00025	0.002201258	3.476	failed	
0.9505	1.334	3.641							
0.9376	1.300	3.964							
0.9443	1.582	3.800		0.94699	0.00025	0.002201258	0.613	passed	
0.9356	1.149	2.706		0.93010	0.00026	0.0022020416	1.563	passed	
0.9437	6.099	14.123		0.93881	0.00024	0.008500868	0.294	passed	
0.9558	4.739	8.296		0.94364	0.00023	0.008500592	1.022	passed	
0.9473	4.428	9.653		0.93073	0.00024	0.008500868	1.286	passed	
0.9317	8.808	16.742		0.91899	0.00023	0.008500592	0.679	passed	

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2.4 Conclusions

Three unirradiated and four irradiated cases were selected as potential benchmarks in the previous section. Experimentally determined values of k_{eff} arise from analysis of the raw pulsed-neutron data. Limited error analysis was done at the time of the original documentation, and subsequently the raw data was reanalyzed, with slightly different detailed results (e.g., prompt neutron decay rate and reactivities). Whether the MCNP4C results are compared to the old data analysis results or the new results changes slightly the outcome of the Z-Tests. All four of the irradiated cases passed the Z-Test, whether the old results or the new results were considered. As for the unirradiated cases, only one case, with the lowest number of fuel tubes loaded (51) failed the Z-Test in both cases. However, the Z-Test value did not exceed the 95% confidence value (1.96) by a large amount, which is impressive, especially considering the small sigma associated with the MCNP4C calculation (only ~ 2 mk). Admittedly, one reason all four irradiated cases passed the Z-Test was the larger uncertainties arising from exposure uncertainty (the statistical overlap was greater).

Attempts to qualify this experiment as a formal benchmark were not successful because the uncertainties on spent-fuel definitions were not sufficient and the system was far from critical (safety concerns limited k_{eff} to a maximum of 0.97). Nevertheless much useful information and insights were gained:

- At an average fuel exposure of 2865 MWd/t, the critical mass for low-enriched metallic uranium tubular fuel is about twice as large as for green fuel.
- There was reasonable agreement between pulse-neutron and approach-to-critical methods in terms of spent-to-green-fuel critical-mass ratios.
- Even for far subcritical systems, decay constants for prompt neutrons show a linear relationship with fuel loading.
- The loading configuration was sensitive to fuel or detector presence in the central lattice hole.
- The WIMSD-5B and MCNP4C codes are reasonable tools for spent fuel analyses.
- The agreement between measured and analytically determined k_{eff} values for higher loading values showed reasonable agreement. Six out of seven values passed the Z-test.

3.0 BENCHMARK SPECIFICATIONS

Neither of the configurations is considered acceptable for use as criticality safety benchmarks, and so benchmark specifications are not provided. However, additional information related to a model specification can be found in Appendix C.

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4.0 RESULTS OF SAMPLE CALCULATIONS

Neither of the configurations is considered acceptable for use as criticality safety benchmarks, and so the results from sample calculations are not provided here. However, additional information and the results of sample calculations, can be found in Appendix D.

5.0 REFERENCES

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APPENDIX A: SPENT FUEL CONCENTRATION AND EXPOSURE DETERMINATION BACKUP DATA

A.1 Concentration Determination Methodology

Valid burnup concentration for MKIA fuel can be generated using the following approaches:

1. Perform radiolytic, chemical and mass spectroscopic analysis of the fuel used in the experiment. This is the ideal approach to validate the fuel isotopic concentration, but was not possible due to the costs involved.
2. Use a burnup analysis code such as WIMSD-5B, employed extensively throughout the industry, and validate it with MKIA data and or with MKIV radiolytical chemical spectroscopic data. The MKIV fuel element is very similar to MKIA fuel. A validation of WIMSD-5B provides assurance that burnup concentrations can be generated successfully for other metallic uranium low enriched fuel such as MKIA.
3. Obtain a burnup analysis code and use it without validation. This approach is less acceptable than approach 2.

A set of burnup data for MKIA fuel focusing on actinides, at an exposure lower than the experiment value was located. The data are listed in Table A-1. Agreement between WIMSD-5B and the data is good. More extensive data existed for MKIV fuel at various exposures. Such data, serving as the basis for production tables, was used to determine how well WIMSD-5B could match the measured isotopics. Several codes besides WIMSD-5B were tested. A series of graphs plotting grams of Pu isotopes and ^{235}U were prepared as a function of fuel exposure. Inner and outer tubes were analyzed separately. The inner tubes for MKIV (see Table A-2) and MKIA (see Table 1) fuel have the same uranium enrichment and nearly the same physical dimensions. Therefore, the MKIV inner serves as an excellent benchmark for MKIA inner. The most notable disagreement exists for ^{236}U . The systematic deviation observed in the ^{236}U plot arises from the fact that the initial ^{236}U value was not well established. The information in Figures A-2 through A-16 becomes part of the production tables.

Figure A-17 shows the ability of WIMSD-5B and other codes to match criticality data of water-moderated green MKIA fuel. The WIMSD-5B results were conservative while MCNP results duplicated experimental data quite well. After the depletion concentrations were generated by WIMSD-5B, they were input into MCNP for calculation of the experimental configuration for spent fuel.

For the subsequent figures MKIV is spelled out, MKIV and Mark IV are the same fuel type just like Mark IA and MKIA are the same.

The WIMSD-5B generated concentrations are listed in Appendix B. They are used as input in MCNP calculations. The MCNP input streams are also listed in Appendix B.

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Table A-1. Irradiated MKIA Fuel Concentrations as Determined in 1975

Average Fuel-Assembly Exposure: 2000 MWd/t				
<u>Component</u>	<u>Nuclei/barn-cm</u>	<u>g/T of U</u>	<u>Calculated prior to experiment, wt. %</u>	<u>Physical, Chemical Analysis after Experiment^(a) wt. % Isotopic</u>
U-235	3.935×10^{-4}	8207.86	0.823	.919 ± .0027
U-236	4.152×10^{-5}	869.74	0.087	.048 ± .0007
U-238	4.680×10^{-2}	988671.3	99.090	99.023 ^(b)
Np-237	1.379×10^{-5}	290.09		
Pu-238	6.650×10^{-7}	14.05	0.716	.040 ± .0013
Pu-239	8.026×10^{-5}	1702.66	86.826	87.841 ^(b)
Pu-240	9.263×10^{-6}	197.33	10.063	10.04 ± .0378
Pu-241	2.063×10^{-6}	44.13	2.250	1.964 ± .0213
Pu-242	1.191×10^{-7}	2.56	0.130	.113 ± .0013
Am-241	1.236×10^{-8}	0.26	0.013	
Fission products	2.350×10^{-4}			
(a) ISO-647 ^{kk}				
(b) Uncertainty not provided in ISO-647				

^{kk} Lan, J. S., *Requirements and Management Plan Use of MCNP Version 4C for Safety and Significant and Safety Class Applications at Fluor Hanford*, Fluor Hanford, Inc., Richland, Washington, 2005.

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Figure A-1. ^{235}U Content vs. Exposure for Mark IV Inner Element

Figure A-2. ^{235}U Content vs. Exposure for Mark IV Outer Element

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Figure A-3. ^{236}U Content vs. Exposure for Mark IV Inner Element

Figure A-4. ^{236}U Content vs. Exposure for Mark IV Outer Element

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Figure A-5. ^{238}Pu Content vs. Exposure for Mark IV Inner Element

Figure A-6. ^{238}Pu Content vs. Exposure for Mark IV Outer Element

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Figure A-7. ^{239}Pu Content vs. Exposure for Mark IV Inner Element Figure A-8. ^{239}Pu Content vs. Exposure for Mark IV Outer Element

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Figure A-9. ²⁴⁰Pu Content vs. Exposure for Mark IV Inner Element

Figure A-10. ²⁴⁰Pu Content vs. Exposure for Mark IV Outer Element

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Figure A-11. ^{241}Pu Content vs. Exposure for Mark IV Inner Element Figure A-12. ^{241}Pu Content vs. Exposure for Mark IV Outer Element

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Figure A-13. ^{241}Am Content vs. Exposure for Mark IV Inner Element

Figure A-14. ^{241}Am Content vs. Exposure for Mark IV Outer Element

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Figure A-15. ^{242}Pu Content vs. Exposure for Mark IV Inner Element Figure A-16. ^{242}Pu Content vs. Exposure for Mark IV Outer Element

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**Table A-2. Unirradiated MKIV Fuel Element Dimensions and Isotopic Composition
(Reference 1)**

Outer Tube		MKIV
Zirconium Clad OD, inch (cm)		2.425 (6.160)
Uranium OD, inch (cm)		2.375 (6.032)
Uranium ID, inch (cm)		1.741 (4.422)
Zirconium Clad ID, inch (cm)		1.701 (4.320)
Clad Fuel Length, inch (cm)		26.10 (66.29)
Uranium Core Length, inch (cm)		25.73 (65.35)
Weight of Element U, 1b (kg)		34.88 (15.82)
Uranium Composition, wt.% U-235		0.947
wt.% U-238		99.00
wt.% U-234		0.01
wt.% U-236		0.04
Inner Tube		
Zirconium Clad OD, inch (cm)		1.279 (3.249)
Uranium OD, inch (cm)		1.219 (3.096)
Uranium ID, inch (cm)		0.520 (1.321)
Zirconium Clad ID, inch (cm)		0.480 (1.219)
Clad Fuel Length, inch (cm)		26.04 (66.14)
Uranium Core Length, inch (cm)		25.67 (65.20)
Weight of Element U, 1b (kg)		16.73 (7.59)
Uranium Composition, wt.% U-235		0.947
wt.% U-238		99.00
wt.% U-234		0.01
wt.% U-236		0.04
Assembly Weight U, 1b (kg)		51.65 (23.43) ^{II}
Density Uranium	18.76 g/cc	
Density Zirconium	6.4 g/cc	

This data is from the N Reactor Safety Analysis Report, UNI-M-90 Addendum 1, February 28, 1978.

^{II} The assembly weight represents best estimates for historical average factor weights.

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Number of Assemblies for Critical. Cylindrical Geometry (two high).

Figure A-17. Number of MKIA Assemblies Required for Criticality

A.2 Exposure Determination Methodology

The MKIA fuel used in the irradiated-fuel subcritical experiment was discharged on May 23, 1974. All fuel discharged on a particular date was given a unique identifying number. Discharge information was kept in a database file that lists all pressure tubes included in a particular discharge, its operating hours, average kilowatts (determined by coolant flow rate and measured temperature rise), and the "loading class" of the fuel charge. The loading class corresponds to a particular axial loading (e.g., loading class 625 is 2 MKIV assemblies, followed by 15 MKIA assemblies 20.88-inches long, followed by another 2 MKIV assemblies). The stored information was used to reconstruct exposure distributions for a given fuel assembly within a reactor discharge. All process tubes selected for the subcritical experiment were of the same loading class 625. The axial flux profile with loading class 625 tubes yielded an adjusted MKIA to average ratio of 1.249 and a peak-to-average of 1.465, which is in line with Traveling Wire Flux Mapping (TWFM) system results for Area 4 of the core where most of the discharged tubes were located (see Figure A-19). To compute power sharing by fuel element, Figure A-18 was used. "Adjusted" means that the first peak (in Figure A-18) was reduced slightly, from that which was calculated by 3DN, to achieve a peak-to-average that better matched TWFM profile 4 in Figure A-19. Exposure was assumed to be proportional to flux. The numeric data for computing power sharing and finding average exposure is contained in Table A-1 and Table A-2.

Since the early days of operation, the N Reactor had a double-peaked axial power shape due to a backup safety system upset. For the 16 tubes identified as source tubes for the subcritical experiment, the operating hours were multiplied by the average kilowatts for that tube to arrive at the total exposure in MWd/tube. These numbers agreed extremely well with the tube exposures documented in Table 3. Next, the modified power shares from loading class 625 were used to expand the tube average exposure into estimates for the exposure for each fuel assembly in that tube, using the equation:

$$E_i = E_o m_o P_i / m_i$$

where: E_i = assembly exposure

E_o = tube exposure

m_i = assembly factor weight

m_o = sum of factor weights over tube

P_i = power sharing for assembly, i

(A.1)

The resulting 240 assembly exposures were ranked by exposure, as shown in Figure A-20. If all 182 assemblies (taken to fill the 91 tube locations in the subcritical experiment) were taken from the high end of the exposure distribution, the average exposure would be 3666 MWd/MT^{mm}U, $\pm$ 282 MWd/MTU, or 3326 MWd/t, $\pm$ 256 MWd/t. If all 182 assemblies were taken from the low end of the distribution, the average exposure would be 3158 MWd/MTU, $\pm$ 608 MWd/MTU, or 2865 MWd/t, $\pm$ 552 MWd/t. The uncertainty is simply the sample standard deviation from the 182 assemblies. It is interesting to note that the standard deviation is lower for the upper limit exposure. This is because, as Figure A-20 shows, the distribution is flatter in the region of the higher exposures. Any attempt to select assemblies with exposures as close together as possible would tend to favor the higher exposures.

^{mm} MT is metric ton, 1000 kg.

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LC 625 (2 MKIV, 15 MKIA, 2 MKIV).

Figure A-18. Modified 3DN Axial Exposure Profile

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Figure A-19. Traveling Wire Flux Mapper (TWFM) Axial Profiles for Nine Different Regions Covering N Reactor Front Face

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Discharge key = 10832.

Figure A-20. Exposure Distribution, May 23, 1974, with Mean Exposure of Highest 182 Assemblies Indicated

Of 240 MKIA assemblies in 16 tubes (16 x 15 = 240)

If all 182 assemblies came from the low-exposure end of the distribution, $\bar{E} = 3158 \pm 608$ MWd/MTU.

If all 182 assemblies came from the high end, $\bar{E} = 3666 \pm 282$ MWd/MTU.

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Table A-3 displays the arithmetic to arrive at the MKIA-average-to-tube-average power-ratio value of 1.25 for the test fuel and the MKIA tube average exposure of 2,865 MWd/t for the same fuel. The first column on the right identifies the process tubes. The top three lines identify the fuel type, the factor weight in MT, and the distance from the front of the fuel charge to the rear in inches. The rest of the numbers represent the power production in each of the elements. The bottom line represents element average power sharing between the fuel assemblies in the 19-element fuel charge.

The remaining columns display various averages for the 16 process tubes dedicated to the experiment, such as the 1.25 MKIA to tube average value, the average exposure for MKIA fuel involved in the test. Table A-4 provides information about power sharing between the elements, fuel element designation, flux profile adjustments and factor weights.

Table A-3. Results of Exposure Calculation for Each Assembly in the 16 Process Tubes

fuel type factor wt	Reconstructed assembly exposure (MWd/MTU) ...															
	MKIV	MKIV	MKIV	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA	MKIA
0.023428	0.023428	0.023428	0.023428	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659	0.01659
13.05	39.15	39.15	60.03	80.91	101.79	122.67	143.55	164.43	185.31	206.19	227.07	247.95	268.83	289.71	310.59	331.47
869.8992	1637.597	1637.597	2987.696	3749.766	4185.862	4349.889	4358.951	4244.163	4126.018	3876.637	4088.09	4158.575	4041.101	3947.525	3847.049	3746.573
849.7557	1599.677	1599.677	2918.513	3662.936	4088.934	4249.163	4258.015	4145.885	4030.475	3786.87	3993.426	4062.278	3947.525	3847.049	3746.573	3646.101
750.6524	1413.113	1413.113	2578.139	3235.743	3612.059	3753.601	3761.421	3662.368	3560.418	3345.223	3527.69	3588.512	3487.142	3385.772	3284.402	3183.032
758.5602	1428	1428	2605.299	3269.831	3650.111	3793.144	3801.047	3700.95	3597.926	3380.464	3564.853	3626.316	3523.878	3422.508	3321.138	3219.768
801.8557	1509.504	1509.504	2753.998	3456.459	3858.444	4009.641	4017.994	3912.185	3803.281	3573.407	3768.32	3833.291	3725.006	3623.636	3522.266	3420.896
807.9843	1521.041	1521.041	2775.047	3482.877	3887.934	4040.287	4048.704	3942.085	3832.349	3600.719	3797.121	3862.589	3753.476	3652.106	3550.736	3449.366
800.0764	1506.155	1506.155	2747.888	3448.789	3849.883	4000.744	4009.079	3903.504	3794.841	3565.478	3759.958	3824.785	3716.741	3615.371	3514.001	3412.631
802.2511	1510.249	1510.249	2755.356	3458.163	3860.347	4011.618	4019.976	3914.114	3805.156	3575.169	3770.178	3835.181	3726.843	3625.473	3524.103	3422.733
802.0534	1509.876	1509.876	2754.677	3457.311	3859.395	4010.63	4018.985	3913.149	3804.218	3574.288	3769.249	3834.236	3725.924	3624.554	3523.184	3421.814
772.2013	1453.679	1453.679	2652.149	3328.631	3715.75	3861.356	3869.4	3767.503	3662.627	3441.254	3628.959	3691.527	3587.247	3485.877	3384.507	3283.137
771.4105	1452.191	1452.191	2649.433	3325.223	3711.945	3857.401	3865.438	3763.645	3658.876	3437.73	3625.243	3687.747	3583.573	3482.203	3380.833	3279.463
784.6561	1477.126	1477.126	2694.926	3382.319	3775.682	3923.635	3931.81	3828.269	3721.701	3496.758	3687.491	3751.068	3645.106	3543.736	3442.366	3340.996
783.47	1474.893	1474.893	2690.852	3377.206	3769.974	3917.704	3925.866	3822.482	3716.075	3491.472	3681.916	3745.398	3639.595	3538.225	3436.855	3335.485
722.1841	1359.522	1359.522	2480.364	3113.029	3475.073	3611.247	3618.771	3523.474	3425.391	3218.357	3393.904	3452.419	3354.893	3256.367	3157.841	3059.315
774.7713	1458.518	1458.518	2660.976	3339.71	3728.117	3874.207	3882.278	3780.042	3674.817	3452.708	3641.037	3703.814	3599.186	3497.756	3396.326	3294.896
711.0217	1338.508	1338.508	2442.026	3064.912	3421.361	3555.43	3562.837	3469.013	3372.446	3168.612	3341.446	3399.057	3303.038	3207.012	3108.982	3010.952
power sharing	0.380	0.715	0.924	1.159	1.294	1.345	1.348	1.312	1.276	1.199	1.264	1.286	1.250	1.214	1.178	1.142

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Table A-3 (continued). Results of Exposure Calculation for Each Assembly in the 16 Process Tubes

MKIA	MKIA	MKIA	MKIA	MKIA	MKIV	MKIV	MKIV	Sum	MKIA avg	MKIV avg	tube avg	MKIA / tube avg	P/avg	Check
0.01659	0.01659	0.01659	0.01659	0.01659	0.023428	0.023428	0.023428	0.34256						
289.71	310.59	331.47	352.35	378.45	404.55	425.67	446.79							
3759.164	3208.715	2611.276	1997.056	1005.375	525.267	297.535	168.103							
3672.116	3134.413	2550.809	1950.812	982.0947	513.1039	290.854	168.103							
3243.853	2768.86	2253.319	1723.297	867.5573	453.2628	256.78403	146.4819							
3278.026	2798.029	2277.057	1741.451	876.8967	458.0377	259.48915	146.4819							
3465.122	2957.729	2407.022	1840.846	926.7349	484.1806	274.29971	146.4819							
3491.606	2980.335	2425.419	1854.916	933.8179	487.8812	276.39619	146.4819							
3457.433	2951.166	2401.681	1836.761	924.6785	483.1063	273.69106	146.4819							
3466.831	2959.188	2408.209	1841.754	927.1918	484.4194	274.43497	146.4819							
3465.976	2958.458	2407.616	1841.3	926.9634	484.3	274.36734	146.4819							
3336.974	2848.346	2318.005	1772.767	892.4621	466.2745	264.15549	146.4819							
3333.557	2845.429	2315.631	1770.952	891.5482	465.797	263.88498	146.4819							
3390.796	2894.287	2355.392	1801.36	906.8567	473.7951	268.41607	146.4819							
3385.67	2889.911	2351.832	1798.637	905.4858	473.0789	268.0103	146.4819							
3120.831	2663.852	2167.863	1657.941	834.6554	436.0729	247.04557	146.4819							
3348.08	2857.825	2325.72	1778.668	895.4324	467.8264	265.03466	146.4819							
3072.594	2622.678	2134.355	1632.315	821.7545	429.3328	243.2271	146.4819							
1.162	0.992	0.807	0.618	0.439	0.229			19	1.149112	0.440829	0.9553539	1.202813	1.410852	0.9553539

Table A-4. Information About Fuel Charge 625

[illegible][illegible]

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APPENDIX B: CONCENTRATIONS CALCULATED WITH WIMSD-5B AND MCNP INPUT LISTINGS

Appendix B contains the spent fuel isotopic and elemental concentrations as calculated by WIMSD-5B for 244 days burnup in N Reactor. This burnup corresponds to an exposure of 2865 MWd/t for MKIA fuel. The fuel was cooled for 1 year, letting the short-lived isotopes decay. The WIMSD-5B-calculated isotopics were used in MCNP to calculate neutron multiplication factors for spent-fuel loadings. WIMSD-5B used hot operating conditions for the N Reactor graphite lattice computed by DCODE. Correct burnup analysis uses hot operating lattice conditions, and isotopic output includes full-power equilibrium ^{135}Xe . Table B-1 contains isotopic output from WIMSD-5B, with ^{135}Xe effectively zeroed out. The exponent in the weight fraction of ^{135}Xe was changed by hand from "-08" to "-18", which effectively reduces full-power ^{135}Xe concentration by a factor of ten billion.

Furthermore, Appendix B contains the input for MCNP4C to calculate reactivities for the various loadings.

**Table B-1. Actinides and Fission Products Calculated by the
WIMSD-5B Code for MKIA Fuel**

Nuclide	MCNP4C input ZAID	Exp = 2865.78 MWd/t Weight Fraction		assembly (g/MT)
		Inner Tubes	Outer Tubes	
U-234	92234.50c	6.45817E-05	6.37482E-05	64.0236
U-235	92235.50c	7.32728E-03	9.14621E-03	8545.1500
U-236	92236.50c	7.54364E-04	9.49915E-04	885.2945
U-238	92238.50c	9.88968E-01	9.86352E-01	987216.6115
Np-237	93237.50c	2.30487E-05	2.60933E-05	25.0872
Np-239	93239.60c	3.19671E-05	3.66065E-05	35.0734
Pu-238	94238.50c	1.50755E-06	1.92986E-06	1.7903
Pu-239	94239.55c	1.60596E-03	1.65331E-03	1637.6631
Pu-240	94240.50c	2.00464E-04	2.68068E-04	245.7288
Pu-241	94241.50c	4.20008E-05	6.28638E-05	55.9697
Pu-242	94242.50c	2.22848E-06	4.26548E-06	3.5924
Am-241	95241.50c	3.65628E-07	5.31399E-07	0.4766
Am-242m	95242.50c	2.84858E-09	4.61716E-09	0.0040
Am-243	95243.50c	6.20209E-08	1.25688E-07	0.1047
Kr-83	36083.50c	3.85629E-06	5.88228E-06	5.2128
Mo-95	42095.50c	6.15979E-05	9.18920E-05	81.8815
Tc-99	43099.50c	6.43800E-05	9.52125E-05	85.0241
Ru-101	44101.50c	5.61945E-05	8.23978E-05	73.7392
Ru-103	44103.50c	9.93947E-06	1.40394E-05	12.6846
Rh-103	45103.50c	2.87360E-05	4.07958E-05	36.8107
Rh-105	45105.50c	2.31843E-07	3.12458E-07	0.2858
Pd-105	46105.50c	2.01566E-05	2.73129E-05	24.9482
Pd-108	46108.50c	6.66617E-06	9.06923E-06	8.2752
Ag-109	47109.50c	3.71863E-06	5.04485E-06	4.6066
Cd-113	48000.50c	3.28128E-08	3.35480E-08	0.0333
In-115	49000.42c	1.88056E-07	2.55983E-07	0.2335
I-127	53127.55c	3.56612E-06	5.26634E-06	4.7045

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Xe-131	54131.50c	4.09240E-05	5.97379E-05	53.5211
Cs-133	55133.55c	9.08858E-05	1.34579E-04	120.1412
Cs-134	55134.60c	1.56928E-06	2.67322E-06	2.3084
I-135	53135.50c	1.42384E-07	2.06460E-07	0.1853
Xe-135	54135.50c	6.77187E-18	8.27153E-18	7.78E-12
Cs-135	55135.50c	2.84366E-05	3.60735E-05	33.5499
Nd-143	60143.50c	7.83470E-05	1.15525E-04	103.2400
Nd-145	60145.50c	5.56337E-05	8.22158E-05	73.4320
Pm-147	61147.50c	1.44458E-05	2.07201E-05	18.6468
Sm-147	62147.50c	2.50861E-06	3.63487E-06	3.2627
Pm-148	61148.50c	1.06969E-07	1.77827E-07	0.1544
Pm-149	61149.50c	2.94964E-07	4.35201E-07	0.3889
Sm-149	62149.50c	8.90093E-07	1.02731E-06	0.9820
Sm-150	62150.50c	1.83518E-05	2.73100E-05	24.3499
Sm-151	62151.50c	3.97657E-06	4.98395E-06	4.6511
Sm-152	62152.50c	9.35356E-06	1.38100E-05	12.3374
Eu-153	63153.50c	3.96806E-06	5.82796E-06	5.2134
Eu-154	63154.50c	4.01527E-07	6.56948E-07	0.5725
Eu-155	63155.50c	2.99017E-07	3.52465E-07	0.3348
Gd-157	64157.50c	1.11224E-08	1.22660E-08	0.0119
Pseudo	50120.35c	3.54063E-04	5.23555E-04	467.5502
U-237	92237.50c	2.81451E-06	3.37802E-06	3.1918
Total		0.999988179	0.999980627	999983.1133

Continuation of Table B-1

Number densities ...

Nuclide	inner	outer
U-234	3.11784E-06	3.07760E-06
U-235	3.52233E-04	4.39671E-04
U-236	3.61095E-05	4.54700E-05
U-238	4.69406E-02	4.68165E-02
Np-237	1.09862E-06	1.24374E-06
Np-239	1.51093E-06	1.73021E-06
Pu-238	7.15551E-08	9.16000E-08
Pu-239	7.59066E-05	7.81442E-05
Pu-240	9.43548E-06	1.26175E-05
Pu-241	1.96867E-06	2.94657E-06
Pu-242	1.04022E-07	1.99106E-07
Am-241	1.71378E-08	2.49079E-08
Am-242m	1.32966E-10	2.15521E-10
Am-243	2.88309E-09	5.84272E-09
Kr-83	5.25506E-07	8.01592E-07
Mo-95	7.33346E-06	1.09401E-05
Tc-99	7.35467E-06	1.08769E-05
Ru-101	6.29237E-06	9.22649E-06
Ru-103	1.09133E-06	1.54150E-06
Rh-103	3.15517E-06	4.47933E-06
Rh-105	2.49707E-08	3.36533E-08
Pd-105	2.17099E-06	2.94176E-06
Pd-108	6.98032E-07	9.49662E-07
Ag-109	3.85809E-07	5.23404E-07
Cd-113	3.28373E-09	3.35731E-09
In-115	1.84922E-08	2.51717E-08

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I-127	3.17508E-07	4.68887E-07
Xe-131	3.53230E-06	5.15620E-06
Cs-133	7.72662E-06	1.14412E-05
Cs-134	1.32414E-07	2.25563E-07
I-135	1.19249E-08	1.72913E-08
Xe-135	5.67165E-19	6.92766E-19
Cs-135	2.38168E-06	3.02129E-06
Nd-143	6.19436E-06	9.13379E-06
Nd-145	4.33778E-06	6.41040E-06
Pm-147	1.11099E-06	1.59354E-06
Sm-147	1.92932E-07	2.79549E-07
Pm-148	8.17095E-09	1.35836E-08
Pm-149	2.23798E-08	3.30200E-08
Sm-149	6.75346E-08	7.79462E-08
Sm-150	1.38313E-06	2.05829E-06
Sm-151	2.97713E-07	3.73132E-07
Sm-152	6.95663E-07	1.02710E-06
Eu-153	2.93188E-07	4.30610E-07
Eu-154	2.94746E-08	4.82240E-08
Eu-155	2.18080E-08	2.57061E-08
Gd-157	8.00839E-10	8.83178E-10
Pseudo	3.33377E-05	4.92966E-05
U-237	1.34153E-07	1.61013E-07

MCNP CALCULATION INPUT

A total of 1,000 neutrons per batch with 500 batches were used. The computing platform was a Dell Precision 360, Pentium 4 personal computer running the Windows XP™ operating system.

The following MCNP4C input file, scu_base.inp, was the base case. It models unexposed MKIA fuel and other structural materials with material impurities. Rubber stoppers were modeled near the top of the aluminum tubes. The finite fill matrix defined a loading map containing 61 fuel tubes. The mode of execution was the eigenvalue mode, with 5,000 neutrons per generation, 1,050 generations, while the first 50 generations are skipped before statistical information is used. The result for this case was $k_{\text{eff}} = 0.95896$, with a standard deviation of 0.00025.

message:

```

3D N Reactor Mark IA Critical Mass Test (BNWL-B-445)
c full core model in MCNP, Hexagonal lattice (3.2 inch pitch), two high
c Horizontal axis x, Depth axis y, vertical axis z
c Perform flux tallies at detector locations
1 1 -2.3062 -24 94 -8 18 104 -84 46 -110 imp:n=1 $ concrete basin floor
816 44 -7.83 -47 110 -111 112 -113 114 imp:n=1 $ steel base plate
2 2 -1.000 47 -471 62 u=1 imp:n=1 $ #1A basin water
201 3 -2.69 471 -48 62 u=1 imp:n=1 $ #1A lower lattice pl
202 2 -1.0 48 -472 62 u=1 imp:n=1 $ basin water
203 3 -2.69 472 -473 62 u=1 imp:n=1 $ upp latt plate
204 2 -1.0 473 -57 62 u=1 imp:n=1 $ basin water
3 3 -2.690 47 -57 63 -62 u=1 imp:n=1 $ #1A aluminum tube
4 2 -1.000 47 -57 64 -63 u=1 imp:n=1 $ #1A tube water
5 2 -1.000 47 -48 -64 u=1 imp:n=1 $ #1A tube water
6 4 -6.400 48 -51 65 -64 u=1 imp:n=1 $ #1A lo zirc-2 clad
7 4 -6.400 48 -49 66 -65 u=1 imp:n=1 $ #1A lo end cap lower
8 5 -18.7627 49 -50 66 -65 u=1 imp:n=1 $ #1A lo fuel-outer
9 4 -6.400 50 -51 66 -65 u=1 imp:n=1 $ #1A lo end cap upper
10 4 -6.400 48 -51 67 -66 u=1 imp:n=1 $ #1A lo zirc-2 clad
11 2 -1.000 48 -51 68 -67 u=1 imp:n=1 $ #1A lo inner annulus
12 4 -6.400 48 -51 69 -68 u=1 imp:n=1 $ #1A lo zirc-2 clad

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13 4 -6.400      48 -49 70 -69 u=1 imp:n=1 $ #1A lo end cap lower
14 6 -18.7627    49 -50 70 -69 u=1 imp:n=1 $ #1A lo fuel-inner
15 4 -6.400      50 -51 70 -69 u=1 imp:n=1 $ #1A lo end cap upper
16 4 -6.400      48 -51 71 -70 u=1 imp:n=1 $ #1A lo zirc-2 clad
17 2 -1.000      48 -51      -71 u=1 imp:n=1 $ #1A lo center hole
18 4 -6.400      51 -54 65 -64 u=1 imp:n=1 $ #1A hi zirc-2 clad
19 4 -6.400      51 -52 66 -65 u=1 imp:n=1 $ #1A hi end cap lower
20 5 -18.7627    52 -53 66 -65 u=1 imp:n=1 $ #1A hi fuel-outer
21 4 -6.400      53 -54 66 -65 u=1 imp:n=1 $ #1A hi end cap upper
22 4 -6.400      51 -54 67 -66 u=1 imp:n=1 $ #1A hi zirc-2 clad
23 2 -1.000      51 -54 68 -67 u=1 imp:n=1 $ #1A hi inner annulus
24 4 -6.400      51 -54 69 -68 u=1 imp:n=1 $ #1A hi zirc-2 clad
25 4 -6.400      51 -52 70 -69 u=1 imp:n=1 $ #1A hi end cap lower
26 6 -18.7627    52 -53 70 -69 u=1 imp:n=1 $ #1A hi fuel-inner
27 4 -6.400      53 -54 70 -69 u=1 imp:n=1 $ #1A hi end cap upper
28 4 -6.400      51 -54 71 -70 u=1 imp:n=1 $ #1A hi zirc-2 clad
29 2 -1.000      51 -54      -71 u=1 imp:n=1 $ #1A hi center hole
30 2 -1.000      54 -55      -64 u=1 imp:n=1 $ #1A tube water
331 3 -2.69      55 -551      -64 u=1 imp:n=1 $      bot stopper plate
31 45 -1.230     551 -552      -64 u=1 imp:n=1 $ #1A tube stopper
332 3 -2.69      552 -56      -64 u=1 imp:n=1 $      top stopper plate
32 2 -1.000      56 -57      -64 u=1 imp:n=1 $ #1A tube water
33 2 -1.000      -471      47 u=2 imp:n=1 $ water only
333 3 -2.69      471      -48 u=2 imp:n=1 $ solid
334 2 -1.0      48      -472 u=2 imp:n=1 $ lattice
335 3 -2.69      472      -473 u=2 imp:n=1 $ plate
336 2 -1.0      473      u=2 imp:n=1 $
337 2 -1.000     62 -471      47 u=3 imp:n=1 $ water only
338 3 -2.69      62 471      -48 u=3 imp:n=1 $ empty
339 2 -1.0      62 48      -472 u=3 imp:n=1 $ lattice
340 3 -2.69      62 472      -473 u=3 imp:n=1 $ position
341 2 -1.0      62 473      u=3 imp:n=1 $
342 2 -1.0      -62 47      u=3 imp:n=1 $
34 0 -33 34 -37 36 -35 38      lat=2 u=4 imp:n=1 $ fuel loading matrix
      fill=-7:7 -7:7 0:0
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 3 3 3 3 3 3 2 2
      2 2 2 2 2 2 3 1 1 1 1 1 1 3 2 2
      2 2 2 2 2 3 1 1 1 1 1 1 1 3 2 2
      2 2 2 2 3 1 1 1 1 1 1 1 1 3 2 2
      2 2 2 3 1 1 1 1 1 1 1 1 1 3 2 2
      2 2 3 1 1 1 1 1 1 1 1 1 1 3 2 2
      2 2 3 1 1 1 1 1 1 1 1 1 3 2 2 2
      2 2 3 1 1 1 1 1 1 1 1 3 2 2 2 2
      2 2 3 1 1 1 1 1 1 1 3 2 2 2 2 2
      2 2 3 1 1 1 1 1 3 2 2 2 2 2 2
      2 2 3 3 3 3 3 3 3 2 2 2 2 2 2
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      35 0      1 -2 11 -12 -4 14 47 -57 fill=4 imp:n=1 $ fuel core
817 2 -1.0 (111 : -112 : 113 : -114 ) -47 110
      -24 94 -8 18 104 -84      imp:n=1 $ water around base plate
818 2 -1.00 101 -21 91 -81 -5 15 (-1: 2: -11: 12: 4: -14) 47 -57 imp:n=1 $ water
819 2 -1.00 102 -22 92 -82 -6 16 (-101: 21: -91: 81: 5: -15) 47 -57 imp:n=1 $ water
820 2 -1.00 103 -23 93 -83 -7 17 (-102: 22: -92: 82: 6: -16) 47 -57 imp:n=1 $ water
821 2 -1.00 104 -24 94 -84 -8 18 (-103: 23: -93: 83: 7: -17) 47 -57 imp:n=1 $ water
822 2 -1.00 104 -24 94 -84 -8 18 57 -58 imp:n=1 $ water-basin
823 2 -1.00 104 -24 94 -84 -8 18 58 -59 imp:n=1 $ water-basin +100 cm
824 2 -1.00 104 -24 94 -84 -8 18 59 -60 imp:n=1 $ water-basin +200 cm
825 2 -1.00 104 -24 94 -84 -8 18 60 -61 imp:n=1 $ water-basin to surface
826 1 -2.3062 104 -24 94 -84 -8 18 72 -46 imp:n=1 $ concrete floor 2-3 feet
827 0 -104: 24: -94: 8: -18: 84: -72: 61 imp:n=0 $ outside world

1 p      1.00 -0.5773503 0.0 -55.1185 $ core boundary, test assembly
2 p      1.00 0.5773503 0.0 55.1185 $ core boundary, test assembly
4 py     47.734 $ core boundary, useful surface
5 py     77.407 $ core boundary, useful surface
6 py    107.887 $ core boundary, useful surface
7 py    138.367 $ core boundary, useful surface
8 py    168.847 $ core boundary, useful surface
11 p     1.00 0.5773503 0.0 -55.1185 $ core boundary, test assembly
12 p     1.00 -0.5773503 0.0 55.1185 $ core boundary, test assembly
14 py    -47.734 $ core boundary, useful surface
15 py    -77.407 $ core boundary, useful surface
16 py   -107.887 $ core boundary, useful surface
17 py   -138.367 $ core boundary, useful surface

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18 py -168.847          $ core boundary, useful surface
21 p 1.00 0.5773503 0.0 89.382 $ core boundary, test assembly
22 p 1.00 0.5773503 0.0 124.577 $ core boundary, test assembly
23 p 1.00 0.5773503 0.0 159.772 $ core boundary, test assembly
24 p 1.00 0.5773503 0.0 194.968 $ core boundary, test assembly
33 px 4.064
34 px -4.064
35 p -0.57735 1.00 0.0 4.69270 $ lattice boundaries for mkia fuel
36 p 0.57735 1.00 0.0 -4.69270 $
37 p 0.57735 1.00 0.0 4.69270
38 p -0.57735 1.00 0.0 -4.69270
46 pz -60.96 $ bottom of basin concrete floor
47 pz 0.0 $ canister bottom = basin floor
471 pz 3.81 $ bot of lower lattice plate
48 pz 5.080 $ bottom of lower MkIA lo end cap, top of lwr lat plt
49 pz 5.6261 $ top of lower MkIA lo end cap
50 pz 57.5691 $ bottom of upper MkIA lo end cap
51 pz 58.1152 $ top of upper MkIA lo end cap
52 pz 58.6613 $ top of lower MkIA hi end cap
53 pz 110.6043 $ bottom of upper MkIA hi end cap
54 pz 111.1504 $ top of upper MkIA hi end cap
55 pz 125.4379 $ top of water space
551 pz 125.7554 $ top of bottom stopper plate
472 pz 127. $ upper lattice plate
473 pz 128.27 $ upper lattice plate
552 pz 130.8354 $ bottom of top stopper plate
56 pz 131.1529 $ top of stopper
57 pz 146.0754 $ top of aluminum pipe
58 pz 246.0754 $ +100 cm of basin water
59 pz 346.0754 $ +200 cm of basin water
60 pz 446.0754 $ +300 cm of basin water
61 pz 546.0754 $ surface of basin water (guessed only)
62 cz 3.8100 $ MkIA #1A aluminum tube o.r.
63 cz 3.6449 $ MkIA #1A aluminum tube i.r.
64 cz 3.0530 $ MkIA #1A zirc-2 clad
65 cz 2.9895 $ MkIA #1A fuel-outer
66 cz 2.3075 $ MkIA #1A zirc-2 clad
67 cz 2.2440 $ MkIA #1A water inner annulus
68 cz 1.5825 $ MkIA #1A zirc-2 clad
69 cz 1.4810 $ MkIA #1A fuel-inner
70 cz 0.6225 $ MkIA #1A zirc-2 clad
71 cz 0.5590 $ MkIA #1A water center hole
72 pz -91.44 $ bottom of basin concrete floor (3 feet)
81 p 1.00 -0.5773503 0.0 89.382 $ core boundary, test assembly
82 p 1.00 -0.5773503 0.0 124.577 $ core boundary, test assembly
83 p 1.00 -0.5773503 0.0 159.772 $ core boundary, test assembly
84 p 1.00 -0.5773503 0.0 194.968 $ core boundary, test assembly
91 p 1.00 0.5773503 0.0 -89.382 $ core boundary, test assembly
92 p 1.00 0.5773503 0.0 -124.577 $ core boundary, test assembly
93 p 1.00 0.5773503 0.0 -159.772 $ core boundary, test assembly
94 p 1.00 0.5773503 0.0 -194.968 $ core boundary, test assembly
101 p 1.00 -0.5773503 0.0 -89.382 $ core boundary, test assembly
102 p 1.00 -0.5773503 0.0 -124.577 $ core boundary, test assembly
103 p 1.00 -0.5773503 0.0 -159.772 $ core boundary, test assembly
104 p 1.00 -0.5773503 0.0 -194.968 $ core boundary, test assembly
110 pz -2.54 $ bottom of steel base plate
111 px 60.96 $ right
112 px -60.96 $ left
113 py 60.96 $ other right
114 py -60.96

mode n
kcode 5000 1.0 50 1050
ksrc 0.0 2.643 31.5976 $ MkIA #1A lo point 1
      0.0 2.643 84.6328 $ MkIA #1A hi point 1
      -1.421 6.819 31.5976 $ MkIA #2A lo point 1
      -1.421 6.819 84.6328 $ MkIA #2A hi point 1
      1.421 6.819 31.5976 $ MkIA #2B lo point 1
      1.421 6.819 84.6328 $ MkIA #2B hi point 1
m1 1001.50c -0.0099731 $ normal concrete (2.3062 g/cc)
     6000.50c -0.0009973
     8016.50c -0.5290521
    11023.50c -0.0160003
    12000.50c -0.0019946
    13027.50c -0.0339953
    14000.50c -0.3370480
    19000.50c -0.0129651

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	20000.50c	-0.0439684	
	26000.55c	-0.0140057	
mt1	poly.01t	grph.01t	\$ 300 K = 17 C = 62.6 F
m2	1001.50c	0.66667	\$ basin water
	8016.50c	0.33333	
mt2	lwtr.01t		\$ 300 K = 17 C = 62.6 F
m3	14000.50c	-0.0060000	\$ aluminum 6061 t6 (2.69 g/cc)
	26000.55c	-0.0070000	
	29000.50c	-0.0020000	
	25055.50c	-0.0015000	
	12000.50c	-0.0090000	
	24000.50c	-0.0020000	
	30000.40c	-0.0025000	
	22000.50c	-0.0015000	
	13027.50c	-0.9685000	
m4	1001.50c	-0.000024975	\$ zirconium (6.4 g/cc)
	4009.50c	-0.000973044	
	5010.50c	-0.000000099	
	5011.50c	-0.000000400	
	6000.50c	-0.000274730	
	7014.50c	-0.000079921	
	11023.50c	-0.000019980	
	12000.50c	-0.000019980	
	13027.50c	-0.000074926	
	14000.50c	-0.000099902	
	22000.50c	-0.000049951	
	23000.50c	-0.000049951	
	24000.50c	-0.000999018	
	25055.50c	-0.000049951	
	26000.55c	-0.001348675	
	27059.50c	-0.000009990	
	28000.50c	-0.000549460	
	29000.50c	-0.000049951	
	40000.42c	-0.980436721	
	42000.50c	-0.000049951	
	48000.42c	-0.000000500	
	50000.42c	-0.014485768	
	72000.42c	-0.000199804	
	74000.55c	-0.000049951	
	82000.42c	-0.000099902	
	92235.50c	-0.000000018	
	92238.50c	-0.000002480	
m5	92234.50c	-0.00009978	\$ outer fuel (green, with impurities)
	92235.50c	-0.01247248	
	92236.50c	-0.00039912	
	92238.50c	-0.98482712	
	1001.50c	-0.00000200	
	4009.50c	-0.00001000	
	5010.50c	-0.00000005	
	5011.50c	-0.00000020	
	6000.50c	-0.00055000	
	11023.50c	-0.00000800	
	12000.50c	-0.00002500	
	13027.50c	-0.00080000	
	14000.50c	-0.00012400	
	17000.50c	-0.00000100	
	24000.50c	-0.00006500	
	25055.50c	-0.00002500	
	26000.55c	-0.00035000	
	27059.50c	-0.00000100	
	28000.50c	-0.00010000	
	29000.50c	-0.00007500	
	40000.42c	-0.00006500	
	48000.42c	-0.00000025	
m6	92234.50c	-0.00007982	\$ inner fuel (green, with impurities)
	92235.50c	-0.00944915	
	92236.50c	-0.00039912	
	92238.50c	-0.98787040	
	1001.50c	-0.00000200	
	4009.50c	-0.00001000	
	5010.50c	-0.00000005	
	5011.50c	-0.00000020	
	6000.50c	-0.00055000	
	11023.50c	-0.00000800	
	12000.50c	-0.00002500	
	13027.50c	-0.00080000	
	14000.50c	-0.00012400	

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17000.50c -0.00000100
24000.50c -0.00006500
25055.50c -0.00002500
26000.55c -0.00035000
27059.50c -0.00000100
28000.50c -0.00010000
29000.50c -0.00007500
40000.42c -0.00006500
48000.42c -0.00000025
m44 6000.50c -0.0002937 $ m44 is SS-304L 7.830 g/cm3
14000.50c -0.0074969
15031.50c -0.0004470
16032.50c -0.0002937
24000.50c -0.2000026
25055.50c -0.0200003
26000.55c -0.6514643
28000.50c -0.1200015
m45 1001.50c 5. $ neoprene stopper CH2:CHCCl:CH2 1.23 g/cc
6000.50c 4.
17000.50c 1.
mt45 poly.01t
c
totnu
print 30 40 60 110 126

```

The following MCNP4C input file, scu_54.inp, also modeled unexposed MKIA fuel and other structural materials with material impurities. Rubber stoppers were modeled near the top of the aluminum tubes. The finite fill matrix defined a loading map containing 54 fuel tubes. The mode of execution was source/response mode, with 5,000,000 neutrons modeled using the sdf card, positioned where the ^{252}Cf source was located in the actual experiment. The source was further defined as a spontaneous fission Watt spectrum, with fitted constants characteristic of ^{252}Cf (taken from the MCNP4C Users Manual). Three neutron flux tallies were requested, corresponding to the three detector locations in the experiment. All detector and source locations were placed vertically at the core midplane. The flux tally at the center of the test configuration yielded the best statistics. This loading represented the greatest number of tubes containing unexposed fuel that were actually loaded into the experiment.

message:

```

3D N Reactor Mark IA Critical Mass Test (BNWL-B-445)
c full core model in MCNP, Hexagonal lattice (3.2 inch pitch), two high
c Horizontal axis x, Depth axis y, vertical axis z
c Perform flux tallies at detector locations
1 1 -2.3062 -24 94 -8 18 104 -84 46 -110 imp:n=1 $ concrete basin floor
816 44 -7.83 -47 110 -111 112 -113 114 imp:n=1 $ steel base plate
2 2 -1.000 47 -471 62 u=1 imp:n=1 $ #1A basin water
201 3 -2.69 471 -48 62 u=1 imp:n=1 $ #1A lower lattice pl
202 2 -1.0 48 -472 62 u=1 imp:n=1 $ basin water
203 3 -2.69 472 -473 62 u=1 imp:n=1 $ upp latt plate
204 2 -1.0 473 -57 62 u=1 imp:n=1 $ basin water
3 3 -2.690 47 -57 63 -62 u=1 imp:n=1 $ #1A aluminum tube
4 2 -1.000 47 -57 64 -63 u=1 imp:n=1 $ #1A tube water
5 2 -1.000 47 -48 -64 u=1 imp:n=1 $ #1A tube water
6 4 -6.400 48 -51 65 -64 u=1 imp:n=1 $ #1A lo zirc-2 clad
7 4 -6.400 48 -49 66 -65 u=1 imp:n=1 $ #1A lo end cap lower
8 5 -18.7627 49 -50 66 -65 u=1 imp:n=1 $ #1A lo fuel-outer
9 4 -6.400 50 -51 66 -65 u=1 imp:n=1 $ #1A lo end cap upper
10 4 -6.400 48 -51 67 -66 u=1 imp:n=1 $ #1A lo zirc-2 clad
11 2 -1.000 48 -51 68 -67 u=1 imp:n=1 $ #1A lo inner annulus
12 4 -6.400 48 -51 69 -68 u=1 imp:n=1 $ #1A lo zirc-2 clad
13 4 -6.400 48 -49 70 -69 u=1 imp:n=1 $ #1A lo end cap lower
14 6 -18.7627 49 -50 70 -69 u=1 imp:n=1 $ #1A lo fuel-inner
15 4 -6.400 50 -51 70 -69 u=1 imp:n=1 $ #1A lo end cap upper
16 4 -6.400 48 -51 71 -70 u=1 imp:n=1 $ #1A lo zirc-2 clad
17 2 -1.000 48 -51 -71 u=1 imp:n=1 $ #1A lo center hole
18 4 -6.400 51 -54 65 -64 u=1 imp:n=1 $ #1A hi zirc-2 clad

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```

19 4 -6.400      51 -52 66 -65 u=1 imp:n=1 $ #1A hi end cap lower
20 5 -18.7627    52 -53 66 -65 u=1 imp:n=1 $ #1A hi fuel-outer
21 4 -6.400      53 -54 66 -65 u=1 imp:n=1 $ #1A hi end cap upper
22 4 -6.400      51 -54 67 -66 u=1 imp:n=1 $ #1A hi zirc-2 clad
23 2 -1.000      51 -54 68 -67 u=1 imp:n=1 $ #1A hi inner annulus
24 4 -6.400      51 -54 69 -68 u=1 imp:n=1 $ #1A hi zirc-2 clad
25 4 -6.400      51 -52 70 -69 u=1 imp:n=1 $ #1A hi end cap lower
26 6 -18.7627    52 -53 70 -69 u=1 imp:n=1 $ #1A hi fuel-inner
27 4 -6.400      53 -54 70 -69 u=1 imp:n=1 $ #1A hi end cap upper
28 4 -6.400      51 -54 71 -70 u=1 imp:n=1 $ #1A hi zirc-2 clad
29 2 -1.000      51 -54 -71 u=1 imp:n=1 $ #1A hi center hole
30 2 -1.000      54 -55 -64 u=1 imp:n=1 $ #1A tube water
331 3 -2.69      55 -551 -64 u=1 imp:n=1 $ bot stopper plate
31 45 -1.230     551 -552 -64 u=1 imp:n=1 $ #1A tube stopper
332 3 -2.69      552 -56 -64 u=1 imp:n=1 $ top stopper plate
32 2 -1.000      56 -57 -64 u=1 imp:n=1 $ #1A tube water
33 2 -1.000      -471 47 u=2 imp:n=1 $ water only
333 3 -2.69      471 -48 u=2 imp:n=1 $ solid
334 2 -1.0      48 -472 u=2 imp:n=1 $ lattice
335 3 -2.69      472 -473 u=2 imp:n=1 $ plate
336 2 -1.0      473 u=2 imp:n=1 $
337 2 -1.000     62 -471 47 u=3 imp:n=1 $ water only
338 3 -2.69      62 471 -48 u=3 imp:n=1 $ empty
339 2 -1.0      62 48 -472 u=3 imp:n=1 $ lattice
340 3 -2.69      62 472 -473 u=3 imp:n=1 $ position
341 2 -1.0      62 473 u=3 imp:n=1 $
342 2 -1.0      -62 47 u=3 imp:n=1 $
34 0 -33 34 -37 36 -35 38 lat=2 u=4 imp:n=1 $ fuel loading matrix
      fill=-7:7 -7:7 0:0
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 3 3 3 3 3 3 2 2
      2 2 2 2 2 2 2 3 3 1 1 1 1 3 3 2 2
      2 2 2 2 2 3 1 1 1 1 1 1 1 3 2 2
      2 2 2 2 3 1 1 1 1 1 1 1 1 3 2 2
      2 2 2 3 1 1 1 1 1 1 1 1 1 3 2 2
      2 2 3 3 1 1 1 3 1 1 1 3 3 2 2
      2 2 3 1 1 1 1 1 1 1 1 1 3 2 2
      2 2 3 1 1 1 1 1 1 1 1 1 3 2 2 2
      2 2 3 1 1 1 1 1 1 1 1 3 2 2 2
      2 2 3 3 1 1 1 3 3 2 2 2 2 2 2
      2 2 3 3 3 3 3 3 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
      2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
35 0      1 -2 11 -12 -4 14 47 -57 fill=4 imp:n=1 $ fuel core
817 2 -1.0 (111 : -112 : 113 : -114 ) -47 110
      -24 94 -8 18 104 -84 imp:n=1 $ water around base plate
818 2 -1.00 101 -21 91 -81 -5 15 (-1: 2: -11: 12: 4: -14) 47 -57 imp:n=1 $ water
819 2 -1.00 102 -22 92 -82 -6 16 (-101: 21: -91: 81: 5: -15) 47 -57 imp:n=1 $ water
820 2 -1.00 103 -23 93 -83 -7 17 (-102: 22: -92: 82: 6: -16) 47 -57 imp:n=1 $ water
821 2 -1.00 104 -24 94 -84 -8 18 (-103: 23: -93: 83: 7: -17) 47 -57 imp:n=1 $ water
822 2 -1.00 104 -24 94 -84 -8 18 57 -58 imp:n=1 $ water-basin
823 2 -1.00 104 -24 94 -84 -8 18 58 -59 imp:n=1 $ water-basin +100 cm
824 2 -1.00 104 -24 94 -84 -8 18 59 -60 imp:n=1 $ water-basin +200 cm
825 2 -1.00 104 -24 94 -84 -8 18 60 -61 imp:n=1 $ water-basin to surface
826 1 -2.3062 104 -24 94 -84 -8 18 72 -46 imp:n=1 $ concrete floor 2-3 feet
827 0 -104: 24: -94: 8: -18: 84: -72: 61 imp:n=0 $ outside world

1 p 1.00 -0.5773503 0.0 -55.1185 $ core boundary, test assembly
2 p 1.00 0.5773503 0.0 55.1185 $ core boundary, test assembly
4 py 47.734 $ core boundary, useful surface
5 py 77.407 $ core boundary, useful surface
6 py 107.887 $ core boundary, useful surface
7 py 138.367 $ core boundary, useful surface
8 py 168.847 $ core boundary, useful surface
11 p 1.00 0.5773503 0.0 -55.1185 $ core boundary, test assembly
12 p 1.00 -0.5773503 0.0 55.1185 $ core boundary, test assembly
14 py -47.734 $ core boundary, useful surface
15 py -77.407 $ core boundary, useful surface
16 py -107.887 $ core boundary, useful surface
17 py -138.367 $ core boundary, useful surface
18 py -168.847 $ core boundary, useful surface
21 p 1.00 0.5773503 0.0 89.382 $ core boundary, test assembly
22 p 1.00 0.5773503 0.0 124.577 $ core boundary, test assembly
23 p 1.00 0.5773503 0.0 159.772 $ core boundary, test assembly
24 p 1.00 0.5773503 0.0 194.968 $ core boundary, test assembly
33 px 4.064

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34	px	-4.064			
35	p	-0.57735	1.00	0.0	4.69270 \$ lattice boundaries for mkia fuel
36	p	0.57735	1.00	0.0	-4.69270 \$
37	p	0.57735	1.00	0.0	4.69270
38	p	-0.57735	1.00	0.0	-4.69270
46	pz	-60.96			\$ bottom of basin concrete floor
47	pz	0.0			\$ canister bottom = basin floor
471	pz	3.81			\$ bot of lower lattice plate
48	pz	5.080			\$ bottom of lower MkIA lo end cap, top of lwr lat plt
49	pz	5.6261			\$ top of lower MkIA lo end cap
50	pz	57.5691			\$ bottom of upper MkIA lo end cap
51	pz	58.1152			\$ top of upper MkIA lo end cap
52	pz	58.6613			\$ top of lower MkIA hi end cap
53	pz	110.6043			\$ bottom of upper MkIA hi end cap
54	pz	111.1504			\$ top of upper MkIA hi end cap
55	pz	125.4379			\$ top of water space
551	pz	125.7554			\$ top of bottom stopper plate
472	pz	127.			\$ upper lattice plate
473	pz	128.27			\$ upper lattice plate
552	pz	130.8354			\$ bottom of top stopper plate
56	pz	131.1529			\$ top of stopper
57	pz	146.0754			\$ top of aluminum pipe
58	pz	246.0754			\$ +100 cm of basin water
59	pz	346.0754			\$ +200 cm of basin water
60	pz	446.0754			\$ +300 cm of basin water
61	pz	546.0754			\$ surface of basin water (guessed only)
62	cz		3.8100		\$ MkIA #1A aluminum tube o.r.
63	cz		3.6449		\$ MkIA #1A aluminum tube i.r.
64	cz		3.0530		\$ MkIA #1A zirc-2 clad
65	cz		2.9895		\$ MkIA #1A fuel-outer
66	cz		2.3075		\$ MkIA #1A zirc-2 clad
67	cz		2.2440		\$ MkIA #1A water inner annulus
68	cz		1.5825		\$ MkIA #1A zirc-2 clad
69	cz		1.4810		\$ MkIA #1A fuel-inner
70	cz		0.6225		\$ MkIA #1A zirc-2 clad
71	cz		0.5590		\$ MkIA #1A water center hole
72	pz	-91.44			\$ bottom of basin concrete floor (3 feet)
81	p	1.00	-0.5773503	0.0	89.382 \$ core boundary, test assembly
82	p	1.00	-0.5773503	0.0	124.577 \$ core boundary, test assembly
83	p	1.00	-0.5773503	0.0	159.772 \$ core boundary, test assembly
84	p	1.00	-0.5773503	0.0	194.968 \$ core boundary, test assembly
91	p	1.00	0.5773503	0.0	-89.382 \$ core boundary, test assembly
92	p	1.00	0.5773503	0.0	-124.577 \$ core boundary, test assembly
93	p	1.00	0.5773503	0.0	-159.772 \$ core boundary, test assembly
94	p	1.00	0.5773503	0.0	-194.968 \$ core boundary, test assembly
101	p	1.00	-0.5773503	0.0	-89.382 \$ core boundary, test assembly
102	p	1.00	-0.5773503	0.0	-124.577 \$ core boundary, test assembly
103	p	1.00	-0.5773503	0.0	-159.772 \$ core boundary, test assembly
104	p	1.00	-0.5773503	0.0	-194.968 \$ core boundary, test assembly
110	pz	-2.54			\$ bottom of steel base plate
111	px	60.96			\$ right
112	px	-60.96			\$ left
113	py	60.96			\$ other right
114	py	-60.96			

mode n

sdef erg= d2 pos= -16. 43. 58.2 wgt=1000

sc2 neutron energy spectrum

sp2 -3 1.025 2.926

c

m1	1001.50c	-0.0099731		\$ normal concrete (2.3062 g/cc)
	6000.50c	-0.0009973		
	8016.50c	-0.5290521		
	11023.50c	-0.0160003		
	12000.50c	-0.0019946		
	13027.50c	-0.0339953		
	14000.50c	-0.3370480		
	19000.50c	-0.0129651		
	20000.50c	-0.0439684		
	26000.55c	-0.0140057		
mt1	poly.01t	grph.01t		\$ 300 K = 17 C = 62.6 F
m2	1001.50c	0.66667		\$ basin water
	8016.50c	0.33333		
mt2	lwtr.01t			\$ 300 K = 17 C = 62.6 F
m3	14000.50c	-0.0060000		\$ aluminum 6061 t6 (2.69 g/cc)
	26000.55c	-0.0070000		
	29000.50c	-0.0020000		

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	25055.50c	-0.0015000	
	12000.50c	-0.0090000	
	24000.50c	-0.0020000	
	30000.40c	-0.0025000	
	22000.50c	-0.0015000	
	13027.50c	-0.9685000	
m4	1001.50c	-0.000024975	\$ zirconium (6.4 g/cc)
	4009.50c	-0.000973044	
	5010.50c	-0.000000099	
	5011.50c	-0.000000400	
	6000.50c	-0.000274730	
	7014.50c	-0.000079921	
	11023.50c	-0.000019980	
	12000.50c	-0.000019980	
	13027.50c	-0.000074926	
	14000.50c	-0.000099902	
	22000.50c	-0.000049951	
	23000.50c	-0.000049951	
	24000.50c	-0.000999018	
	25055.50c	-0.000049951	
	26000.55c	-0.001348675	
	27059.50c	-0.000009990	
	28000.50c	-0.000549460	
	29000.50c	-0.000049951	
	40000.42c	-0.980436721	
	42000.50c	-0.000049951	
	48000.42c	-0.000000500	
	50000.42c	-0.014485768	
	72000.42c	-0.000199804	
	74000.55c	-0.000049951	
	82000.42c	-0.000099902	
	92235.50c	-0.000000018	
	92238.50c	-0.000002480	
m5	92234.50c	-0.00009978	\$ outer fuel (green, with impurities)
	92235.50c	-0.01247248	
	92236.50c	-0.00039912	
	92238.50c	-0.98482712	
	1001.50c	-0.00000200	
	4009.50c	-0.00001000	
	5010.50c	-0.00000005	
	5011.50c	-0.000000020	
	6000.50c	-0.00055000	
	11023.50c	-0.00000800	
	12000.50c	-0.00002500	
	13027.50c	-0.00080000	
	14000.50c	-0.00012400	
	17000.50c	-0.00000100	
	24000.50c	-0.00006500	
	25055.50c	-0.00002500	
	26000.55c	-0.00035000	
	27059.50c	-0.00000100	
	28000.50c	-0.00010000	
	29000.50c	-0.00007500	
	40000.42c	-0.00006500	
	48000.42c	-0.00000025	
m6	92234.50c	-0.00007982	\$ inner fuel (green, with impurities)
	92235.50c	-0.00944915	
	92236.50c	-0.00039912	
	92238.50c	-0.98787040	
	1001.50c	-0.00000200	
	4009.50c	-0.00001000	
	5010.50c	-0.00000005	
	5011.50c	-0.000000020	
	6000.50c	-0.00055000	
	11023.50c	-0.00000800	
	12000.50c	-0.00002500	
	13027.50c	-0.00080000	
	14000.50c	-0.00012400	
	17000.50c	-0.00000100	
	24000.50c	-0.00006500	
	25055.50c	-0.00002500	
	26000.55c	-0.00035000	
	27059.50c	-0.00000100	
	28000.50c	-0.00010000	
	29000.50c	-0.00007500	
	40000.42c	-0.00006500	
	48000.42c	-0.00000025	

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```

m44      6000.50c -0.0002937      $ m44 is SS-304L    7.830 g/cm3
        14000.50c -0.0074969
        15031.50c -0.0004470
        16032.50c -0.0002937
        24000.50c -0.2000026
        25055.50c -0.0200003
        26000.55c -0.6514643
        28000.50c -0.1200015
m45      1001.50c  5.              $ neoprene stopper CH2:CHCCl:CH2 1.23 g/cc
        6000.50c  4.
        17000.50c 1.
mt45     poly.01t
c
totnu
f5:n      0.0    0.0  58.2  1.
        -48.0   0.0  58.2  1.
        48.0   0.0  58.2  1.
nps      5000000
print    30 40 60 110 126

```

The following MCNP4C input file, sci_ibEx.inp, was the base case model for exposed fuel. It modeled MKIA fuel exposed to 3158 MWd/MT (2865 MWd/ton) and other structural materials with material impurities. Isotopic distributions for both inner and outer fuel elements were calculated using the WIMSD code. Rubber stoppers were removed from the aluminum tubes. The finite fill matrix defines a loading map containing 91 fuel tubes. The mode of execution was eigenvalue mode, with 5,000 neutrons per generation, 1,050 generations, while the first 50 generations were skipped before statistical information was used. The result for this case was $k_{eff} = 0.94364$, with a standard deviation of 0.00023.

message:

```

3D N Reactor Mark IA Critical Mass Test (BNWL-B-445)
c      full core model in MCNP, Hexagonal lattice (3.2 inch pitch), two high
c      Horizontal axis x, Depth axis y, vertical axis z
c      Perform flux tallies at detector locations
1 1 -2.3062 -24 94 -8 18 104 -84 46 -110 imp:n=1 $ concrete basin floor
816 44 -7.83 -47 110 -111 112 -113 114 imp:n=1 $ steel base plate
2 2 -1.000 47 -471 62 u=1 imp:n=1 $ #1A basin water
201 3 -2.69 471 -48 62 u=1 imp:n=1 $ #1A lower lattice pl
202 2 -1.0 48 -472 62 u=1 imp:n=1 $ basin water
203 3 -2.69 472 -473 62 u=1 imp:n=1 $ upp latt plate
204 2 -1.0 473 -57 62 u=1 imp:n=1 $ basin water
3 3 -2.690 47 -57 63 -62 u=1 imp:n=1 $ #1A aluminum tube
4 2 -1.000 47 -57 64 -63 u=1 imp:n=1 $ #1A tube water
5 2 -1.000 47 -48 -64 u=1 imp:n=1 $ #1A tube water
6 4 -6.400 48 -51 65 -64 u=1 imp:n=1 $ #1A lo zirc-2 clad
7 4 -6.400 48 -49 66 -65 u=1 imp:n=1 $ #1A lo end cap lower
8 5 -18.7627 49 -50 66 -65 u=1 imp:n=1 $ #1A lo fuel-outer
9 4 -6.400 50 -51 66 -65 u=1 imp:n=1 $ #1A lo end cap upper
10 4 -6.400 48 -51 67 -66 u=1 imp:n=1 $ #1A lo zirc-2 clad
11 2 -1.000 48 -51 68 -67 u=1 imp:n=1 $ #1A lo inner annulus
12 4 -6.400 48 -51 69 -68 u=1 imp:n=1 $ #1A lo zirc-2 clad
13 4 -6.400 48 -49 70 -69 u=1 imp:n=1 $ #1A lo end cap lower
14 6 -18.7627 49 -50 70 -69 u=1 imp:n=1 $ #1A lo fuel-inner
15 4 -6.400 50 -51 70 -69 u=1 imp:n=1 $ #1A lo end cap upper
16 4 -6.400 48 -51 71 -70 u=1 imp:n=1 $ #1A lo zirc-2 clad
17 2 -1.000 48 -51 -71 u=1 imp:n=1 $ #1A lo center hole
18 4 -6.400 51 -54 65 -64 u=1 imp:n=1 $ #1A hi zirc-2 clad
19 4 -6.400 51 -52 66 -65 u=1 imp:n=1 $ #1A hi end cap lower
20 5 -18.7627 52 -53 66 -65 u=1 imp:n=1 $ #1A hi fuel-outer
21 4 -6.400 53 -54 66 -65 u=1 imp:n=1 $ #1A hi end cap upper
22 4 -6.400 51 -54 67 -66 u=1 imp:n=1 $ #1A hi zirc-2 clad
23 2 -1.000 51 -54 68 -67 u=1 imp:n=1 $ #1A hi inner annulus
24 4 -6.400 51 -54 69 -68 u=1 imp:n=1 $ #1A hi zirc-2 clad
25 4 -6.400 51 -52 70 -69 u=1 imp:n=1 $ #1A hi end cap lower
26 6 -18.7627 52 -53 70 -69 u=1 imp:n=1 $ #1A hi fuel-inner
27 4 -6.400 53 -54 70 -69 u=1 imp:n=1 $ #1A hi end cap upper
28 4 -6.400 51 -54 71 -70 u=1 imp:n=1 $ #1A hi zirc-2 clad
29 2 -1.000 51 -54 -71 u=1 imp:n=1 $ #1A hi center hole
30 2 -1.000 54 -56 -64 u=1 imp:n=1 $ #1A tube water

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```

32 2 -1.000          56 -57          -64 u=1 imp:n=1 $ #1A tube water
33 2 -1.000          -471          47 u=2 imp:n=1 $ water only
333 3 -2.69          471          -48 u=2 imp:n=1 $ solid
334 2 -1.0          48          -472 u=2 imp:n=1 $ lattice
335 3 -2.69          472          -473 u=2 imp:n=1 $ plate
336 2 -1.0          473          u=2 imp:n=1 $
337 2 -1.000          62 -471          47 u=3 imp:n=1 $ water only
338 3 -2.69          62 471          -48 u=3 imp:n=1 $ empty
339 2 -1.0          62 48          -472 u=3 imp:n=1 $ lattice
340 3 -2.69          62 472          -473 u=3 imp:n=1 $ position
341 2 -1.0          62 473          u=3 imp:n=1 $
342 2 -1.0          -62 47          u=3 imp:n=1 $
34 0 -33 34 -37 36 -35 38          lat=2          u=4 imp:n=1 $ fuel loading matrix
          fill=-7:7 -7:7 0:0
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 1 1 1 1 1 1 2 2
2 2 2 2 2 2 1 1 1 1 1 1 1 2 2
2 2 2 2 2 1 1 1 1 1 1 1 1 2 2
2 2 2 2 1 1 1 1 1 1 1 1 1 2 2
2 2 2 2 1 1 1 1 1 1 1 1 1 2 2
2 2 2 1 1 1 1 1 1 1 1 1 1 2 2
2 2 1 1 1 1 1 1 1 1 1 1 1 2 2
2 2 1 1 1 1 1 1 1 1 1 1 1 2 2
2 2 1 1 1 1 1 1 1 1 1 1 2 2 2
2 2 1 1 1 1 1 1 1 1 2 2 2 2 2
2 2 1 1 1 1 1 1 1 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
35 0          1 -2 11 -12 -4 14 47 -57 fill=4 imp:n=1 $ fuel core
817 2 -1.0 (111 : -112 : 113 : -114 ) -47 110
          -24 94 -8 18 104 -84          imp:n=1 $ water around base plate
818 2 -1.00 101 -21 91 -81 -5 15 (-1: 2: -11: 12: 4: -14) 47 -57 imp:n=1 $ water
819 2 -1.00 102 -22 92 -82 -6 16 (-101: 21: -91: 81: 5: -15) 47 -57 imp:n=1 $ water
820 2 -1.00 103 -23 93 -83 -7 17 (-102: 22: -92: 82: 6: -16) 47 -57 imp:n=1 $ water
821 2 -1.00 104 -24 94 -84 -8 18 (-103: 23: -93: 83: 7: -17) 47 -57 imp:n=1 $ water
822 2 -1.00 104 -24 94 -84 -8 18 57 -58          imp:n=1 $ water-basin
823 2 -1.00 104 -24 94 -84 -8 18 58 -59          imp:n=1 $ water-basin +100 cm
824 2 -1.00 104 -24 94 -84 -8 18 59 -60          imp:n=1 $ water-basin +200 cm
825 2 -1.00 104 -24 94 -84 -8 18 60 -61          imp:n=1 $ water-basin to surface
826 1 -2.3062 104 -24 94 -84 -8 18 72 -46          imp:n=1 $ concrete floor 2-3 feet
827 0          -104: 24: -94: 8: -18: 84: -72: 61 imp:n=0 $ outside world

1 p          1.00 -0.5773503 0.0 -55.1185 $ core boundary, test assembly
2 p          1.00 0.5773503 0.0 55.1185 $ core boundary, test assembly
4 py          47.734          $ core boundary, useful surface
5 py          77.407          $ core boundary, useful surface
6 py          107.887          $ core boundary, useful surface
7 py          138.367          $ core boundary, useful surface
8 py          168.847          $ core boundary, useful surface
11 p          1.00 0.5773503 0.0 -55.1185 $ core boundary, test assembly
12 p          1.00 -0.5773503 0.0 55.1185 $ core boundary, test assembly
14 py          -47.734          $ core boundary, useful surface
15 py          -77.407          $ core boundary, useful surface
16 py          -107.887          $ core boundary, useful surface
17 py          -138.367          $ core boundary, useful surface
18 py          -168.847          $ core boundary, useful surface
21 p          1.00 0.5773503 0.0 89.382 $ core boundary, test assembly
22 p          1.00 0.5773503 0.0 124.577 $ core boundary, test assembly
23 p          1.00 0.5773503 0.0 159.772 $ core boundary, test assembly
24 p          1.00 0.5773503 0.0 194.968 $ core boundary, test assembly
33 px          4.064
34 px          -4.064
35 p          -0.57735          1.00 0.0          4.69270 $ lattice boundaries for mkia fuel
36 p          0.57735          1.00 0.0          -4.69270 $
37 p          0.57735          1.00 0.0          4.69270
38 p          -0.57735          1.00 0.0          -4.69270
46 pz -60.96          $ bottom of basin concrete floor
47 pz 0.0          $ canister bottom = basin floor
471 pz 3.81          $ bot of lower lattice plate
48 pz 5.080          $ bottom of lower MkIA lo end cap, top of lwr lat plt
49 pz 5.6261          $ top of lower MkIA lo end cap
50 pz 57.5691          $ bottom of upper MkIA lo end cap
51 pz 58.1152          $ top of upper MkIA lo end cap
52 pz 58.6613          $ top of lower MkIA hi end cap
53 pz 110.6043          $ bottom of upper MkIA hi end cap
54 pz 111.1504          $ top of upper MkIA hi end cap

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55	pz	125.4379			\$ top of water space
472	pz	127.			\$ upper lattice plate
473	pz	128.27			\$ upper lattice plate
56	pz	131.1529			\$ top of stopper
57	pz	146.0754			\$ top of aluminum pipe
58	pz	246.0754			\$ +100 cm of basin water
59	pz	346.0754			\$ +200 cm of basin water
60	pz	446.0754			\$ +300 cm of basin water
61	pz	546.0754			\$ surface of basin water (guessed only)
62	cz		3.8100		\$ MkIA #1A aluminum tube o.r.
63	cz		3.6449		\$ MkIA #1A aluminum tube i.r.
64	cz		3.0530		\$ MkIA #1A zirc-2 clad
65	cz		2.9895		\$ MkIA #1A fuel-outer
66	cz		2.3075		\$ MkIA #1A zirc-2 clad
67	cz		2.2440		\$ MkIA #1A water inner annulus
68	cz		1.5825		\$ MkIA #1A zirc-2 clad
69	cz		1.4810		\$ MkIA #1A fuel-inner
70	cz		0.6225		\$ MkIA #1A zirc-2 clad
71	cz		0.5590		\$ MkIA #1A water center hole
72	pz	-91.44			\$ bottom of basin concrete floor (3 feet)
81	p	1.00	-0.5773503	0.0	89.382 \$ core boundary, test assembly
82	p	1.00	-0.5773503	0.0	124.577 \$ core boundary, test assembly
83	p	1.00	-0.5773503	0.0	159.772 \$ core boundary, test assembly
84	p	1.00	-0.5773503	0.0	194.968 \$ core boundary, test assembly
91	p	1.00	0.5773503	0.0	-89.382 \$ core boundary, test assembly
92	p	1.00	0.5773503	0.0	-124.577 \$ core boundary, test assembly
93	p	1.00	0.5773503	0.0	-159.772 \$ core boundary, test assembly
94	p	1.00	0.5773503	0.0	-194.968 \$ core boundary, test assembly
101	p	1.00	-0.5773503	0.0	-89.382 \$ core boundary, test assembly
102	p	1.00	-0.5773503	0.0	-124.577 \$ core boundary, test assembly
103	p	1.00	-0.5773503	0.0	-159.772 \$ core boundary, test assembly
104	p	1.00	-0.5773503	0.0	-194.968 \$ core boundary, test assembly
110	pz	-2.54			\$ bottom of steel base plate
111	px	60.96			\$ right
112	px	-60.96			\$ left
113	py	60.96			\$ other right
114	py	-60.96			

mode n

kcode 5000 1.0 50 1050

ksrc	0.0	2.643	31.5976	\$ MkIA #1A lo point	1
	0.0	2.643	84.6328	\$ MkIA #1A hi point	1
	-1.421	6.819	31.5976	\$ MkIA #2A lo point	1
	-1.421	6.819	84.6328	\$ MkIA #2A hi point	1
	1.421	6.819	31.5976	\$ MkIA #2B lo point	1
	1.421	6.819	84.6328	\$ MkIA #2B hi point	1
m1	1001.50c	-0.0099731		\$ normal concrete (2.3062 g/cc)	
	6000.50c	-0.0009973			
	8016.50c	-0.5290521			
	11023.50c	-0.0160003			
	12000.50c	-0.0019946			
	13027.50c	-0.0339953			
	14000.50c	-0.3370480			
	19000.50c	-0.0129651			
	20000.50c	-0.0439684			
	26000.55c	-0.0140057			
mt1	poly.01t	grph.01t		\$ 300 K = 17 C = 62.6 F	
m2	1001.50c	0.66667		\$ basin water	
	8016.50c	0.33333			
mt2	lwtr.01t			\$ 300 K = 17 C = 62.6 F	
m3	14000.50c	-0.0060000		\$ aluminum 6061 t6 (2.69 g/cc)	
	26000.55c	-0.0070000			
	29000.50c	-0.0020000			
	25055.50c	-0.0015000			
	12000.50c	-0.0090000			
	24000.50c	-0.0020000			
	30000.40c	-0.0025000			
	22000.50c	-0.0015000			
	13027.50c	-0.9685000			
m4	1001.50c	-0.000024975		\$ zirconium (6.4 g/cc)	
	4009.50c	-0.000973044			
	5010.50c	-0.000000099			
	5011.50c	-0.0000000400			
	6000.50c	-0.000274730			
	7014.50c	-0.000079921			
	11023.50c	-0.000019980			
	12000.50c	-0.000019980			

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	13027.50c	-0.000074926	
	14000.50c	-0.000099902	
	22000.50c	-0.000049951	
	23000.50c	-0.000049951	
	24000.50c	-0.000999018	
	25055.50c	-0.000049951	
	26000.55c	-0.001348675	
	27059.50c	-0.000009990	
	28000.50c	-0.000549460	
	29000.50c	-0.000049951	
	40000.42c	-0.980436721	
	42000.50c	-0.000049951	
	48000.42c	-0.000000500	
	50000.42c	-0.014485768	
	72000.42c	-0.000199804	
	74000.55c	-0.000049951	
	82000.42c	-0.000099902	
	92235.50c	-0.000000018	
	92238.50c	-0.000002480	
m5	92234.50c	-6.37482E-05	\$ outer fuel (2865 MWd/t, no Xe-135)
	92235.50c	-9.14621E-03	
	92236.50c	-9.49915E-04	
	92238.50c	-9.86352E-01	
	93237.50c	-2.60933E-05	
	93239.60c	-3.66065E-05	
	94238.50c	-1.92986E-06	
	94239.55c	-1.65331E-03	
	94240.50c	-2.68068E-04	
	94241.50c	-6.28638E-05	
	94242.50c	-4.26548E-06	
	95241.50c	-5.31399E-07	
	95242.50c	-4.61716E-09	
	95243.50c	-1.25688E-07	
	36083.50c	-5.88228E-06	
	42095.50c	-9.18920E-05	
	43099.50c	-9.52125E-05	
	44101.50c	-8.23978E-05	
	44103.50c	-1.40394E-05	
	45103.50c	-4.07958E-05	
	45105.50c	-3.12458E-07	
	46105.50c	-2.73129E-05	
	46108.50c	-9.06923E-06	
	47109.50c	-5.04485E-06	
	48000.50c	-3.35480E-08	
	49000.42c	-2.55983E-07	
	53127.55c	-5.26634E-06	
	54131.50c	-5.97379E-05	
	55133.55c	-1.34579E-04	
	55134.60c	-2.67322E-06	
	53135.50c	-2.06460E-07	
	54135.50c	-8.27153E-18	
	55135.50c	-3.60735E-05	
	60143.50c	-1.15525E-04	
	60145.50c	-8.22158E-05	
	61147.50c	-2.07201E-05	
	62147.50c	-3.63487E-06	
	61148.50c	-1.77827E-07	
	61149.50c	-4.35201E-07	
	62149.50c	-1.02731E-06	
	62150.50c	-2.73100E-05	
	62151.50c	-4.98395E-06	
	62152.50c	-1.38100E-05	
	63153.50c	-5.82796E-06	
	63154.50c	-6.56948E-07	
	63155.50c	-3.52465E-07	
	64157.50c	-1.22660E-08	
	50120.35c	-5.23555E-04	
	92237.50c	-3.37802E-06	
m6	92234.50c	-6.45817E-05	\$ inner fuel (2865 MWd/t, no Xe-135)
	92235.50c	-7.32728E-03	
	92236.50c	-7.54364E-04	
	92238.50c	-9.88968E-01	
	93237.50c	-2.30487E-05	
	93239.60c	-3.19671E-05	
	94238.50c	-1.50755E-06	
	94239.55c	-1.60596E-03	
	94240.50c	-2.00464E-04	

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```

94241.50c -4.20008E-05
94242.50c -2.22848E-06
95241.50c -3.65628E-07
95242.50c -2.84858E-09
95243.50c -6.20209E-08
36083.50c -3.85629E-06
42095.50c -6.15979E-05
43099.50c -6.43800E-05
44101.50c -5.61945E-05
44103.50c -9.93947E-06
45103.50c -2.87360E-05
45105.50c -2.31843E-07
46105.50c -2.01566E-05
46108.50c -6.66617E-06
47109.50c -3.71863E-06
48000.50c -3.28128E-08
49000.42c -1.88056E-07
53127.55c -3.56612E-06
54131.50c -4.09240E-05
55133.55c -9.08858E-05
55134.60c -1.56928E-06
53135.50c -1.42384E-07
54135.50c -6.77187E-18
55135.50c -2.84366E-05
60143.50c -7.83470E-05
60145.50c -5.56337E-05
61147.50c -1.44458E-05
62147.50c -2.50861E-06
61148.50c -1.06969E-07
61149.50c -2.94964E-07
62149.50c -8.90093E-07
62150.50c -1.83518E-05
62151.50c -3.97657E-06
62152.50c -9.35356E-06
63153.50c -3.96806E-06
63154.50c -4.01527E-07
63155.50c -2.99017E-07
64157.50c -1.11224E-08
50120.35c -3.54063E-04
92237.50c -2.81451E-06
m44 6000.50c -0.0002937 $ m44 is SS-304L 7.830 g/cm3
14000.50c -0.0074969
15031.50c -0.0004470
16032.50c -0.0002937
24000.50c -0.2000026
25055.50c -0.0200003
26000.55c -0.6514643
28000.50c -0.1200015
m45 1001.50c 5. $ neoprene stopper CH2:CHCCl:CH2 1.23 g/cc
6000.50c 4.
17000.50c 1.
mt45 poly.01t
c
totnu
print 30 40 60 110 126

```

The following MCNP4C input file, sci_90.inp, also modeled exposed MKIA fuel and other structural materials with material impurities. The finite fill matrix defined a loading map containing 90 fuel tubes. The mode of execution was source/response mode, with 5,000,000 neutrons modeled using the sdf card, positioned where the ^{252}Cf source was located in the actual experiment. The source was further defined as a spontaneous fission Watt spectrum, with fitted constants characteristic of ^{252}Cf (taken from the MCNP4C Users Manual). Three neutron flux tallies were requested, corresponding to the three detector locations in the experiment. All detector and source locations were placed vertically at the core midplane. The flux tally at the center of the core yielded the best statistics. This loading represented the greatest number of tubes containing exposed fuel that was actually loaded into the experimental test assembly.

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message:

3D N Reactor Mark IA Critical Mass Test (BNWL-B-445)

c full core model in MCNP, Hexagonal lattice (3.2 inch pitch), two high

c Horizontal axis x, Depth axis y, vertical axis z

c Perform flux tallies at detector locations

```

1 1 -2.3062 -24 94 -8 18 104 -84 46 -110 imp:n=1 $ concrete basin floor
816 44 -7.83 -47 110 -111 112 -113 114 imp:n=1 $ steel base plate
2 2 -1.000 47 -471 62 u=1 imp:n=1 $ #1A basin water
201 3 -2.69 471 -48 62 u=1 imp:n=1 $ #1A lower lattice pl
202 2 -1.0 48 -472 62 u=1 imp:n=1 $ basin water
203 3 -2.69 472 -473 62 u=1 imp:n=1 $ upp latt plate
204 2 -1.0 473 -57 62 u=1 imp:n=1 $ basin water
3 3 -2.690 47 -57 63 -62 u=1 imp:n=1 $ #1A aluminum tube
4 2 -1.000 47 -57 64 -63 u=1 imp:n=1 $ #1A tube water
5 2 -1.000 47 -48 -64 u=1 imp:n=1 $ #1A tube water
6 4 -6.400 48 -51 65 -64 u=1 imp:n=1 $ #1A lo zirc-2 clad
7 4 -6.400 48 -49 66 -65 u=1 imp:n=1 $ #1A lo end cap lower
8 5 -18.7627 49 -50 66 -65 u=1 imp:n=1 $ #1A lo fuel-outer
9 4 -6.400 50 -51 66 -65 u=1 imp:n=1 $ #1A lo end cap upper
10 4 -6.400 48 -51 67 -66 u=1 imp:n=1 $ #1A lo zirc-2 clad
11 2 -1.000 48 -51 68 -67 u=1 imp:n=1 $ #1A lo inner annulus
12 4 -6.400 48 -51 69 -68 u=1 imp:n=1 $ #1A lo zirc-2 clad
13 4 -6.400 48 -49 70 -69 u=1 imp:n=1 $ #1A lo end cap lower
14 6 -18.7627 49 -50 70 -69 u=1 imp:n=1 $ #1A lo fuel-inner
15 4 -6.400 50 -51 70 -69 u=1 imp:n=1 $ #1A lo end cap upper
16 4 -6.400 48 -51 71 -70 u=1 imp:n=1 $ #1A lo zirc-2 clad
17 2 -1.000 48 -51 -71 u=1 imp:n=1 $ #1A lo center hole
18 4 -6.400 51 -54 65 -64 u=1 imp:n=1 $ #1A hi zirc-2 clad
19 4 -6.400 51 -52 66 -65 u=1 imp:n=1 $ #1A hi end cap lower
20 5 -18.7627 52 -53 66 -65 u=1 imp:n=1 $ #1A hi fuel-outer
21 4 -6.400 53 -54 66 -65 u=1 imp:n=1 $ #1A hi end cap upper
22 4 -6.400 51 -54 67 -66 u=1 imp:n=1 $ #1A hi zirc-2 clad
23 2 -1.000 51 -54 68 -67 u=1 imp:n=1 $ #1A hi inner annulus
24 4 -6.400 51 -54 69 -68 u=1 imp:n=1 $ #1A hi zirc-2 clad
25 4 -6.400 51 -52 70 -69 u=1 imp:n=1 $ #1A hi end cap lower
26 6 -18.7627 52 -53 70 -69 u=1 imp:n=1 $ #1A hi fuel-inner
27 4 -6.400 53 -54 70 -69 u=1 imp:n=1 $ #1A hi end cap upper
28 4 -6.400 51 -54 71 -70 u=1 imp:n=1 $ #1A hi zirc-2 clad
29 2 -1.000 51 -54 -71 u=1 imp:n=1 $ #1A hi center hole
30 2 -1.000 54 -56 -64 u=1 imp:n=1 $ #1A tube water
32 2 -1.000 56 -57 -64 u=1 imp:n=1 $ #1A tube water
33 2 -1.000 -471 47 u=2 imp:n=1 $ water only
333 3 -2.69 471 -48 u=2 imp:n=1 $ solid
334 2 -1.0 48 -472 u=2 imp:n=1 $ lattice
335 3 -2.69 472 -473 u=2 imp:n=1 $ plate
336 2 -1.0 473 u=2 imp:n=1 $
337 2 -1.000 62 -471 47 u=3 imp:n=1 $ water only
338 3 -2.69 62 471 -48 u=3 imp:n=1 $ empty
339 2 -1.0 62 48 -472 u=3 imp:n=1 $ lattice
340 3 -2.69 62 472 -473 u=3 imp:n=1 $ position
341 2 -1.0 62 473 u=3 imp:n=1 $
342 2 -1.0 -62 47 u=3 imp:n=1 $
34 0 -33 34 -37 36 -35 38 lat=2 u=4 imp:n=1 $ fuel loading matrix
fill=-7:7 -7:7 0:0
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 1 1 1 1 1 1 1 2 2
2 2 2 2 2 2 2 1 1 1 1 1 1 1 2 2
2 2 2 2 2 1 1 1 1 1 1 1 1 1 2 2
2 2 2 2 1 1 1 1 1 1 1 1 1 1 2 2
2 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2
2 2 1 1 1 1 1 1 3 1 1 1 1 1 2 2
2 2 1 1 1 1 1 1 1 1 1 1 1 1 2 2 2
2 2 1 1 1 1 1 1 1 1 1 1 2 2 2 2
2 2 1 1 1 1 1 1 1 1 1 1 2 2 2 2
2 2 1 1 1 1 1 1 1 1 2 2 2 2 2 2
2 2 1 1 1 1 1 1 1 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
35 0 1 -2 11 -12 -4 14 47 -57 fill=4 imp:n=1 $ fuel core
817 2 -1.0 (111 : -112 : 113 : -114 ) -47 110
-24 94 -8 18 104 -84 imp:n=1 $ water around base plate
818 2 -1.00 101 -21 91 -81 -5 15 (-1: 2: -11: 12: 4: -14) 47 -57 imp:n=1 $ water
819 2 -1.00 102 -22 92 -82 -6 16 (-101: 21: -91: 81: 5: -15) 47 -57 imp:n=1 $ water
820 2 -1.00 103 -23 93 -83 -7 17 (-102: 22: -92: 82: 6: -16) 47 -57 imp:n=1 $ water

```


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```

821 2 -1.00 104 -24 94 -84 -8 18 (-103: 23: -93: 83: 7: -17) 47 -57 imp:n=1 $ water
822 2 -1.00 104 -24 94 -84 -8 18 57 -58 imp:n=1 $ water-basin
823 2 -1.00 104 -24 94 -84 -8 18 58 -59 imp:n=1 $ water-basin +100 cm
824 2 -1.00 104 -24 94 -84 -8 18 59 -60 imp:n=1 $ water-basin +200 cm
825 2 -1.00 104 -24 94 -84 -8 18 60 -61 imp:n=1 $ water-basin to surface
826 1 -2.3062 104 -24 94 -84 -8 18 72 -46 imp:n=1 $ concrete floor 2-3 feet
827 0 -104: 24: -94: 8: -18: 84: -72: 61 imp:n=0 $ outside world

1 p 1.00 -0.5773503 0.0 -55.1185 $ core boundary, test assembly
2 p 1.00 0.5773503 0.0 55.1185 $ core boundary, test assembly
4 py 47.734 $ core boundary, useful surface
5 py 77.407 $ core boundary, useful surface
6 py 107.887 $ core boundary, useful surface
7 py 138.367 $ core boundary, useful surface
8 py 168.847 $ core boundary, useful surface
11 p 1.00 0.5773503 0.0 -55.1185 $ core boundary, test assembly
12 p 1.00 -0.5773503 0.0 55.1185 $ core boundary, test assembly
14 py -47.734 $ core boundary, useful surface
15 py -77.407 $ core boundary, useful surface
16 py -107.887 $ core boundary, useful surface
17 py -138.367 $ core boundary, useful surface
18 py -168.847 $ core boundary, useful surface
21 p 1.00 0.5773503 0.0 89.382 $ core boundary, test assembly
22 p 1.00 0.5773503 0.0 124.577 $ core boundary, test assembly
23 p 1.00 0.5773503 0.0 159.772 $ core boundary, test assembly
24 p 1.00 0.5773503 0.0 194.968 $ core boundary, test assembly
33 px 4.064
34 px -4.064
35 p -0.57735 1.00 0.0 4.69270 $ lattice boundaries for mkia fuel
36 p 0.57735 1.00 0.0 -4.69270 $
37 p 0.57735 1.00 0.0 4.69270
38 p -0.57735 1.00 0.0 -4.69270
46 pz -60.96 $ bottom of basin concrete floor
47 pz 0.0 $ canister bottom = basin floor
471 pz 3.81 $ bot of lower lattice plate
48 pz 5.080 $ bottom of lower MkIA lo end cap, top of lwr lat plt
49 pz 5.6261 $ top of lower MkIA lo end cap
50 pz 57.5691 $ bottom of upper MkIA lo end cap
51 pz 58.1152 $ top of upper MkIA lo end cap
52 pz 58.6613 $ top of lower MkIA hi end cap
53 pz 110.6043 $ bottom of upper MkIA hi end cap
54 pz 111.1504 $ top of upper MkIA hi end cap
55 pz 125.4379 $ top of water space
472 pz 127. $ upper lattice plate
473 pz 128.27 $ upper lattice plate
56 pz 131.1529 $ top of stopper
57 pz 146.0754 $ top of aluminum pipe
58 pz 246.0754 $ +100 cm of basin water
59 pz 346.0754 $ +200 cm of basin water
60 pz 446.0754 $ +300 cm of basin water
61 pz 546.0754 $ surface of basin water (guessed only)
62 cz 3.8100 $ MkIA #1A aluminum tube o.r.
63 cz 3.6449 $ MkIA #1A aluminum tube i.r.
64 cz 3.0530 $ MkIA #1A zirc-2 clad
65 cz 2.9895 $ MkIA #1A fuel-outer
66 cz 2.3075 $ MkIA #1A zirc-2 clad
67 cz 2.2440 $ MkIA #1A water inner annulus
68 cz 1.5825 $ MkIA #1A zirc-2 clad
69 cz 1.4810 $ MkIA #1A fuel-inner
70 cz 0.6225 $ MkIA #1A zirc-2 clad
71 cz 0.5590 $ MkIA #1A water center hole
72 pz -91.44 $ bottom of basin concrete floor (3 feet)
81 p 1.00 -0.5773503 0.0 89.382 $ core boundary, test assembly
82 p 1.00 -0.5773503 0.0 124.577 $ core boundary, test assembly
83 p 1.00 -0.5773503 0.0 159.772 $ core boundary, test assembly
84 p 1.00 -0.5773503 0.0 194.968 $ core boundary, test assembly
91 p 1.00 0.5773503 0.0 -89.382 $ core boundary, test assembly
92 p 1.00 0.5773503 0.0 -124.577 $ core boundary, test assembly
93 p 1.00 0.5773503 0.0 -159.772 $ core boundary, test assembly
94 p 1.00 0.5773503 0.0 -194.968 $ core boundary, test assembly
101 p 1.00 -0.5773503 0.0 -89.382 $ core boundary, test assembly
102 p 1.00 -0.5773503 0.0 -124.577 $ core boundary, test assembly
103 p 1.00 -0.5773503 0.0 -159.772 $ core boundary, test assembly
104 p 1.00 -0.5773503 0.0 -194.968 $ core boundary, test assembly
110 pz -2.54 $ bottom of steel base plate
111 px 60.96 $ right
112 px -60.96 $ left

```

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113 py 60.96
114 py -60.96

\$ other right

mode n

sdef erg= d2 pos= -16. 43. 58.2 wgt=1000

sc2 neutron energy spectrum

sp2 -3 1.025 2.926

c

m1 1001.50c -0.0099731 \$ normal concrete (2.3062 g/cc)

6000.50c -0.0009973

8016.50c -0.5290521

11023.50c -0.0160003

12000.50c -0.0019946

13027.50c -0.0339953

14000.50c -0.3370480

19000.50c -0.0129651

20000.50c -0.0439684

26000.55c -0.0140057

mt1 poly.01t grph.01t \$ 300 K = 17 C = 62.6 F

m2 1001.50c 0.66667 \$ basin water

8016.50c 0.33333

mt2 lwtr.01t \$ 300 K = 17 C = 62.6 F

m3 14000.50c -0.0060000 \$ aluminum 6061 t6 (2.69 g/cc)

26000.55c -0.0070000

29000.50c -0.0020000

25055.50c -0.0015000

12000.50c -0.0090000

24000.50c -0.0020000

30000.40c -0.0025000

22000.50c -0.0015000

13027.50c -0.9685000

m4 1001.50c -0.000024975 \$ zirconium (6.4 g/cc)

4009.50c -0.000973044

5010.50c -0.000000099

5011.50c -0.000000400

6000.50c -0.000274730

7014.50c -0.000079921

11023.50c -0.000019980

12000.50c -0.000019980

13027.50c -0.000074926

14000.50c -0.000099902

22000.50c -0.000049951

23000.50c -0.000049951

24000.50c -0.000999018

25055.50c -0.000049951

26000.55c -0.001348675

27059.50c -0.000009990

28000.50c -0.000549460

29000.50c -0.000049951

40000.42c -0.980436721

42000.50c -0.000049951

48000.42c -0.000000500

50000.42c -0.014485768

72000.42c -0.000199804

74000.55c -0.000049951

82000.42c -0.000099902

92235.50c -0.000000018

92238.50c -0.000002480

m5 92234.50c -6.37482E-05 \$ outer fuel (2865 MWd/t, no Xe-135)

92235.50c -9.14621E-03

92236.50c -9.49915E-04

92238.50c -9.86352E-01

93237.50c -2.60933E-05

93239.60c -3.66065E-05

94238.50c -1.92986E-06

94239.55c -1.65331E-03

94240.50c -2.68068E-04

94241.50c -6.28638E-05

94242.50c -4.26548E-06

95241.50c -5.31399E-07

95242.50c -4.61716E-09

95243.50c -1.25688E-07

36083.50c -5.88228E-06

42095.50c -9.18920E-05

43099.50c -9.52125E-05

44101.50c -8.23978E-05

44103.50c -1.40394E-05

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45103.50c	-4.07958E-05	
45105.50c	-3.12458E-07	
46105.50c	-2.73129E-05	
46108.50c	-9.06923E-06	
47109.50c	-5.04485E-06	
48000.50c	-3.35480E-08	
49000.42c	-2.55983E-07	
53127.55c	-5.26634E-06	
54131.50c	-5.97379E-05	
55133.55c	-1.34579E-04	
55134.60c	-2.67322E-06	
53135.50c	-2.06460E-07	
54135.50c	-8.27153E-18	
55135.50c	-3.60735E-05	
60143.50c	-1.15525E-04	
60145.50c	-8.22158E-05	
61147.50c	-2.07201E-05	
62147.50c	-3.63487E-06	
61148.50c	-1.77827E-07	
61149.50c	-4.35201E-07	
62149.50c	-1.02731E-06	
62150.50c	-2.73100E-05	
62151.50c	-4.98395E-06	
62152.50c	-1.38100E-05	
63153.50c	-5.82796E-06	
63154.50c	-6.56948E-07	
63155.50c	-3.52465E-07	
64157.50c	-1.22660E-08	
50120.35c	-5.23555E-04	
92237.50c	-3.37802E-06	
m6 92234.50c	-6.45817E-05	\$ inner fuel (2865 MWd/t, no Xe-135)
92235.50c	-7.32728E-03	
92236.50c	-7.54364E-04	
92238.50c	-9.88968E-01	
93237.50c	-2.30487E-05	
93239.60c	-3.19671E-05	
94238.50c	-1.50755E-06	
94239.55c	-1.60596E-03	
94240.50c	-2.00464E-04	
94241.50c	-4.20008E-05	
94242.50c	-2.22848E-06	
95241.50c	-3.65628E-07	
95242.50c	-2.84858E-09	
95243.50c	-6.20209E-08	
36083.50c	-3.85629E-06	
42095.50c	-6.15979E-05	
43099.50c	-6.43800E-05	
44101.50c	-5.61945E-05	
44103.50c	-9.93947E-06	
45103.50c	-2.87360E-05	
45105.50c	-2.31843E-07	
46105.50c	-2.01566E-05	
46108.50c	-6.66617E-06	
47109.50c	-3.71863E-06	
48000.50c	-3.28128E-08	
49000.42c	-1.88056E-07	
53127.55c	-3.56612E-06	
54131.50c	-4.09240E-05	
55133.55c	-9.08858E-05	
55134.60c	-1.56928E-06	
53135.50c	-1.42384E-07	
54135.50c	-6.77187E-18	
55135.50c	-2.84366E-05	
60143.50c	-7.83470E-05	
60145.50c	-5.56337E-05	
61147.50c	-1.44458E-05	
62147.50c	-2.50861E-06	
61148.50c	-1.06969E-07	
61149.50c	-2.94964E-07	
62149.50c	-8.90093E-07	
62150.50c	-1.83518E-05	
62151.50c	-3.97657E-06	
62152.50c	-9.35356E-06	
63153.50c	-3.96806E-06	
63154.50c	-4.01527E-07	
63155.50c	-2.99017E-07	
64157.50c	-1.11224E-08	

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	50120.35c	-3.54063E-04	
	92237.50c	-2.81451E-06	
m44	6000.50c	-0.0002937	\$ m44 is SS-304L 7.830 g/cm3
	14000.50c	-0.0074969	
	15031.50c	-0.0004470	
	16032.50c	-0.0002937	
	24000.50c	-0.2000026	
	25055.50c	-0.0200003	
	26000.55c	-0.6514643	
	28000.50c	-0.1200015	
m45	1001.50c	5.	\$ neoprene stopper CH2:CHCCl:CH2 1.23 g/cc
	6000.50c	4.	
	17000.50c	1.	
mt45	poly.01t		
c			
totnu			
f5:n	0.0 0.0 58.2 1.		
	-48.0 0.0 58.2 1.		
	48.0 0.0 58.2 1.		
nps	5000000		
print	30 40 60 110 126		

APPENDIX C: PROBLEM SPECIFICATION

C.1 Description of Model

The calculational model is a replica of the actual experimental configuration shown in Figures 2 and 3. The discussion in this section will consider the model and the materials involved, structural details, and the surrounding media. Reference will be made to prior sections of this document that discuss specific features of the experimental configuration.

The potential benchmark model is shown in the series of Figures C-1 through C-4, with further details in Figures 9 and 10. There are two models: one for unirradiated MKIA fuel elements and the other for irradiated MKIA fuel elements. Both green and spent fuels were tested in the same test stand. The basic features of the models are the fuel, the fuel tubes, the test assembly with the lattice or spacer plates, water, and rubber stoppers for green fuel.

The following structures and components are omitted from the model for simplification:

- Nearest wall 2 ft (60.96 cm) from edge of test assembly (See Figure 1)
- The aluminum handle and center post on rubber stoppers (See Figures 4 and 5)
- Cone shape on the bottom of the fuel tubes (See Figure 9)
- Al bracing and corner posts of the test assembly (See Figure 2)
- Detector hardware inside detector tubes (See Figures 3 and 6)
- Hole blocking plates for green fuel tests (See Figures 6, 7 and 8)
- Neutron-source hardware inside source tubes
- Basin water impurities (See Table 4).

The components that were omitted were made out of Al, a material with low absorption and scattering characteristics, and were spaced from the fissile material by water to limit interaction. Impurities in the basin water were ignored due to the very low concentrations of radioactive contaminants.

C.2 Dimensions

There are two sets of dimensions important to the model: the test assembly geometry and the fuel. The dimensions for the models are found in a number of illustrations and tables. Nominal fuel dimensions mostly from the N Reactor Safety Analysis Report are used. Fuel and cladding dimensions are found in Table 1 and Figures 9, 10, C-1 and C-4. The test assembly dimensions and details are contained in Figures 2, 6, 9, C-1 and C-4. Dimensional data on fuel tubes are located in Figures 4, 6, C-1 and C-2, while information about the rubber stopper is found in Figures 5, 6 and C-4. The lattice plates are pictured in Figures 3, 6 11 and 12.

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C.2.1 The Test Assembly

The proposed model consists of the test stand, fuel tubes, detectors and source tubes. There are two 72-cm x 72-cm x 1.27-cm thick lattice plates holding the fuel tubes in a vertical position. The lattice pitch is hexagonal with a center-to-center spacing of 8.13 cm. The 6061 aluminum alloy lattice plates are separated by 41.76 in. (106.07 cm). They contain 91 holes, 7.777 cm in diameter to hold the fuel tubes.

C.2.2 The Fuel and Cladding Dimensions

The fuel and cladding dimensions used in the model are found in Section 1.3, Figures 9 and 10, and Table 1. Each fuel tube is modeled in great detail. The plug in the bottom of the aluminum tube is explicitly described, as are all the details of the tube-in-tube fuel elements with their cladding and end caps. The plug was modeled as aluminum cylinder 2-inches (5.08 cm) high.

C.3 Material Data

The following fuel and cladding material data has been included in the model. The two uranium-metal annular cylinders are clad with zirconium on all surfaces. The outer tube is enriched to 1.25 wt.% ^{235}U in U metal and the inner tube has an enrichment of 0.947 wt.% ^{235}U . The spent fuel is the same (MKIA fuel assemblies), averaging 2865 MWd/t.

The actinides for unirradiated and irradiated fuel are listed in the table below. Fission product concentrations are given in Appendix B. Material concentrations for zirconium cladding and for Al structure components are listed on the following page.

Table C-1. Concentrations of Materials Used in the Model of Green Fuel, in Nuclide/b-cm

Concrete Isotope	304L Stainless Steel		Zircaloy-2		Fuel assembly		Inner	
	N (at/(b-cm))	Isotope	N (at/(b-cm))	Isotope	Isotope	N (at/(b-cm))	Outer	N (at/(b-cm))
Hydrogen Carbon Oxygen Sodium Magnesium Aluminum Silicon Potassium Calcium Iron Total	1.37420E-02	Hydrogen	1.15316E-04	Carbon	9.55030E-05	U-234	4.85237E-06	3.88189E-06
	1.15316E-04	Carbon	1.25863E-03	Silicon	4.16124E-04	U-235	6.03957E-04	4.57558E-04
	4.59232E-02	Oxygen	6.80479E-05	Phosphorous	3.82610E-08	U-236	1.92446E-05	1.92446E-05
	9.66568E-04	Sodium	4.31886E-05	Sulfur	1.40067E-07	U-238	4.70861E-02	4.72316E-02
	1.13973E-04	Magnesium	1.81369E-02	Chromium	8.81552E-05	Hydrogen	2.25847E-05	2.25847E-05
	1.74981E-03	Aluminum	1.71656E-03	Manganese	2.19912E-05	Beryllium	1.26291E-05	1.26291E-05
	1.66666E-02	Silicon	5.50032E-02	Iron	3.34957E-06	Boron-10	5.65502E-08	5.65502E-08
	4.60526E-04	Potassium	9.64095E-03	Nickel	3.16832E-06	Boron-11	2.07021E-07	2.07021E-07
	1.52361E-03	Calcium	8.59827E-02	Total	1.07026E-05	Carbon	5.21178E-04	5.21178E-04
	3.48292E-04	Iron			1.37092E-05	Sodium	3.96057E-06	3.96057E-06
Water Isotope	8.16100E-02	Total			4.02078E-06	Magnesium	1.17070E-05	1.17070E-05
					3.77914E-06	Aluminum	3.37463E-04	3.37463E-04
					7.40499E-05	Silicon	5.02507E-05	5.02507E-05
					3.50422E-06	Chlorine	3.21033E-07	3.21033E-07
					9.30740E-05	Chromium	1.42281E-05	1.42281E-05
					6.53332E-07	Manganese	5.17928E-06	5.17928E-06
					3.60822E-05	Iron	7.13298E-05	7.13298E-05
					3.02954E-06	Cobalt	1.93127E-07	1.93127E-07
					4.14220E-02	Nickel	1.93927E-05	1.93927E-05
					2.00662E-06	Copper	1.34331E-05	1.34331E-05
Neoprene Isotope					1.71261E-08	Zirconium	8.10974E-06	8.10974E-06
					4.70300E-04	Cadmium	2.53127E-08	2.53127E-08
					4.31430E-06	Total	4.88064E-02	4.88064E-02
					1.04713E-06			
					1.85825E-06			
					2.94861E-10			
					4.01445E-08			
					4.27727E-02			
Carbon Hydrogen Chlorine Total	4.35353E-02	Carbon	3.46069E-04	Silicon	3.46069E-04	Cobalt	1.93127E-07	1.93127E-07
	5.43946E-02	Hydrogen	2.03044E-04	Iron	2.03044E-04	Nickel	1.93927E-05	1.93927E-05
	1.08818E-02	Chlorine	5.09841E-05	Copper	5.09841E-05	Copper	1.34331E-05	1.34331E-05
			4.42294E-05	Manganese	4.42294E-05	Zirconium	8.10974E-06	8.10974E-06
			5.99846E-04	Magnesium	5.99846E-04	Molybdenum	2.53127E-08	2.53127E-08
			6.23093E-05	Chromium	6.23093E-05	Cadmium	2.53127E-08	2.53127E-08
			6.19329E-05	Zinc	6.19329E-05	Tin	4.88064E-02	4.88064E-02
			5.07493E-05	Titanium	5.07493E-05	Hafnium		
			5.81468E-02	Aluminum	5.81468E-02	Tungsten		
			5.95660E-02	Total	5.95660E-02	Lead		
Total						Uranium-235		
						Uranium-238		
						Total		

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C.3.1 Fuel and Cladding

Table A-1 in Appendix A contains actinide concentrations as determined in 1975. More recent calculations have been performed with WIMSD-5B. Good agreement between the earlier isotopics and WIMSD-5B results exists. WIMSD-5B has a limited set of fission-product cross sections (see Table B-1 in Appendix B for calculated isotopic concentrations). Not all the fission products are included in the model—only those important ones contained in the WIMSD-5B cross-section library and available in the MCNP cross-section library. The cladding compositions with contaminants are found in Section 1.4.2.

C.3.2 Structural Materials

The composition for the main structural materials used in the model are found in Table 6. The predominant structural material is the aluminum alloy 6061 used in fuel tubes, the lattice plates, and vertical bracing.

C.4 Surrounding Media

The test assembly is submerged in the storage pool water. Water is the moderator as well as the reflector. Material composition for wet concrete is found in Table 6. The same table contains the composition data for the steel plate under the test assembly and the neoprene used in the stoppers.

C.5 Temperature Data

A reasonable assumption of water temperature was close to 10 °C. As such, cross-section libraries at 20 °C should be suitable.

C.6 Model Dimensions

Various cross-sectional views of the model are shown in Figures C-1, C-2, C-3 and C-4. The first figure shows a vertical cut through the test stand loaded with fuel. The fuel tubes and lattice plates are visible, and the MKIA fuel elements are on top of each other. The cylindrical cones and the steel plate can also be discerned. The fuel and stoppers can be seen in more detail in Figure C-2. At the end of the fuel elements, the end-cap regions are clearly observable.

The next view, Figure C-3, represents a horizontal cut through the test configuration containing 91 fuel tubes. The cylindrical regions representing concentric uranium metal tubes, cladding and the wall of the aluminum test tube are clearly visible. All other space outside and inside the assembly is filled with water. Figure C-4 is an enlargement of Figure C-3.

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Figure C-1. Side View of Model Geometry

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Figure C-2. Enlargement of Figure C-1 Showing Rubber Stoppers and Fuel Assembly End Caps

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Figure C-3. Top View of Model Geometry

Figure C-4. Enlargement of Figure C-3

$r_1 = 0.5590$ cm	$r_6 = 2.3075$ cm
$r_2 = 0.6225$ cm	$r_7 = 2.9895$ cm
$r_3 = 1.4810$ cm	$r_8 = 3.0530$ cm
$r_4 = 1.5825$ cm	$r_4 = 1.5825$ cm
$r_5 = 2.2440$ cm	$r_{10} = 3.8100$ cm

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APPENDIX D: RESULTS OF SAMPLE CALCULATIONS**D.1 Computation Details**

The model described and illustrated in Appendix C was evaluated with the MCNP 4C code version using the ENDF/B-V cross-section set^{nnoo}. A total of 1000 neutrons per batch with 500 batches were used. The computing platform was a Dell Precision 360, Pentium 4 personal computer, running the Windows XPTM operating system. Isotopic and elemental composition for spent fuel at 2865 MWd/t was obtained from tabulations given in this document in Table 11 and by WIMS-D calculations.

D.2 Results of the Computations

Computations were carried out using the sample calculation model, first for green fuel and then for spent fuel. Calculations were performed at the same loading steps where measurements were taken. The information was plotted clearly showing some anticipated trends.

The calculated results behave very differently for highly subcritical loadings. As the test assembly is loaded, the difference between calculated and measured results decreases. At 54 tubes loaded, the agreement between kinetic distortion corrected values and measurement is very satisfactory (see Figure D-1). For irradiated fuel, Figure D-2, the agreement is even more pronounced for data corrected for kinetic distortions. It was possible to use MCNP4C to increase the number of fuel tubes to actual critical (not just the $k_{eff} = 0.97$ the experiment was limited to). Linear interpolation to $k_{eff} = 1.00$ was at 106 tubes for unexposed fuel. With exposed fuel at 2865 MWd/t, MCNP4C was not able to attain $k_{eff} = 1.00$. Even at 217 tubes, far beyond the real number of fuel tubes the experimental grid plate would hold, the result was $k_{eff} = 0.9858$.

Cases at high loadings of green, as well as spent fuel, were compared to calculated k_{eff} . Uncertainties were computed for both the experimental and analytic results (see Table D-1). The MCNP4C column contains the sigma from the Monte Carlo calculation itself. This sigma is combined quadratically with the calculational sensitivity to physical uncertainties and tolerances (which were 2.19 mk and 8.50 mk for unirradiated and irradiated cases, respectively). This pooled result was used in comparison with the measured k_{eff} s and their measured uncertainties in the final Z-Tests to determine if sufficient statistical overlap exists. Table D-1 contains measured values of k_{eff} resulting from the new analysis of the raw pulsed-neutron data. Applying statistical tasks to the results showed that the majority of cases passed the Z-Test.

ⁿⁿ Garelis, E. and J. L. Russell, Jr., *Theory of Pulsed Neutron Source Measurements*, Nuclear Science and Engineering, 1963, Vol. 16, pp. 263-270.

^{oo} Gozani, T., *A Modified Procedure for the Evaluation of Pulsed Source Experiments in Subcritical Reactors*, Nukleonik, 1962, Vol. 4, p. 348.

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Table D-1. Sample Calculation Results

Code (Cross Section Set) Case	MCNP4C (Continuous Energy $k_{\text{eff}} \pm \sigma$ ENDF/B-V)	Experimental $k_{\text{eff}} \pm \sigma$	Z-Test	Comment
51L Loading	0.93903 ± 0.00025	0.92602 ± 0.00419	2.75	Failed
54L Loading	0.94699 ± 0.00025	0.94909 ± 0.00380	0.48	Passed
54U Unloading	0.93010 ± 0.00026	0.94443 ± 0.00271	4.10	Failed
85L Loading	0.93881 ± 0.00024	0.94541 ± 0.01412	0.40	Passed
91L Loading	0.94364 ± 0.00023	0.95647 ± 0.00830	1.08	Passed
90U Unloading	0.93073 ± 0.00024	0.94585 ± 0.00965	1.18	Passed
78U Unloading	0.91899 ± 0.00023	0.92490 ± 0.01674	0.31	Passed

D.3 Explanation of Differences

The difference between calculations and measured data at low values of k_{eff} is based on the observation that the detectors are too far from the fueled region to realize any significant count rate. The calculation assumes that the neutron distribution is in the fueled region, thus resulting in a higher k_{eff} value. As the loading increases, the detectors are closer to fissile material and see a higher neutron flux region. When k_{eff} values approach 0.9 and higher the calculations and experimental values agree more closely. Some of the difference can be compensated for by the use of the spatial correction factor (see Section 2.6). Another factor is in the definition of k_{eff} itself. Static calculations of k_{eff} , whether Monte Carlo or deterministic, converge on the fundamental eigenmode of the problem. This flux distribution is the functional weighting appearing in both numerator and denominator of the estimate of k_{eff} . In highly subcritical systems, the spatial neutron distribution may be mostly the flux response from the source. Furthermore, the neutron response to a particular source (even in a weakly multiplying medium) is a different mathematical problem than the (mathematically) homogeneous eigenvalue problem. A very good discussion of the distinction between these two classes of problems is given by Cullen et al., in *Static and Dynamic Criticality: Are They Different?*^{PP}

D.4 Monte Carlo Simulation of Approach-to-Critical

The MCNP4C code was also processed in the neutron flux tally mode to replicate the count rates at the various detector locations during loading and unloading of the assembly thereby providing additional benchmark information. A match of the experimental count rates and the

^{PP} Cullen, D. E., C. J. Clouse, R. Procassini, and R. C. Little, *Static and Dynamic Criticality: Are They Different?*, 2003, UCRL-TR-201506, Lawrence Livermore National Laboratory.

calculated count rates permits a more realistic extrapolation to critical. Agreement between measured and calculated data further validates the model of the test assembly, the computer code, and the cross sections used in the computations. As such, the usefulness of the subcritical benchmark is extended. Figure D-3 shows the agreement between measured and calculated count rates for detector 3 during unloading of spent fuel; the calculated points were normalized to the detector readings. Applying this approach to both irradiated and unirradiated loadings and extrapolating beyond the last measured points, yielded new predictions as to when the test configuration would go critical. The prediction of the ratio of spent-to-green fuel critical masses remains at least a factor of 2. Extrapolating the unexposed approach-to-critical simulations, 108 tubes would be required for critical. This is very close to the interpolated critical number of tubes (106) from executing MCNP4C in the k_{eff} mode. Hence, MCNP4C (in k_{eff} mode) agrees very well with MCNP4C, when executing the code in source/response, or flux tally mode. This result is reassuring, as it supports the accuracy of eigenvalue calculations when $k_{\text{eff}} = 1.00$. For the irradiated cases, the comparison is not as clear-cut, because the cases were not carried out to an interpolated evaluation of critical (MCNP4C did not evaluate any loading with $k_{\text{eff}} > 1.0$). However, both k_{eff} mode and flux tally mode appear to agree that the critical loading is beyond 220 tubes.

D.5 Neutron Spectra

A WIMSD-5B⁹⁹ calculation was performed to obtain the spectrum for low-enriched concentric uranium metal fuel tubes in water. Figure D-4 shows the neutron spectra for a test configuration with green fuel at three locations. Neutron flux per unit lethargy is plotted as a function of neutron energy. The shape of the curve is representative of a thermal reactor system.

D.6 Summary and Conclusions

As a subcritical benchmark, this experiment is one of a kind. It provides valuable information to engineers working in burnup and depletion calculations. Some of the important features are listed below:

- Experiment carried out in a clean geometry amenable to computer modeling.
- An actual critical-mass measurement between green and spent fuel accomplished.
- A comparison between approach to critical and pulse neutron measurement achieved.
- Sensitivity to various computed parameters explored.
- Detector readings at various fuel loading steps completed.
- Experiment performed with identical fuel elements both green and spent.
- All spent fuel had similar exposure with uniform axial exposure distributions.
- Experiment extended with calculations to critical by using dose rate features of MCNP.
- Comparison of most higher k_{eff} cases between calculated and experimental results passed the statistical Z-Test.

⁹⁹ WIMSD5B – A Neutronics Code for Standard Lattice Physics Analysis, ANSWERS Software Service, AEA Technology, June 1997.

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Figure D-1. k_{eff} vs. Number of Tubes: MCNP4C vs. Experimental Data (unirradiated).

Figure D-2. k_{eff} vs. Number of Tubes: MCNP4C vs. Experimental Data (irradiated).

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Inverse Count Rate vs. Fuel Columns.
 1/Net cpm on 4/10/1975, Core Center.

Inverse Count Rate vs. Fuel Columns.
 1/Net cpm on 4/23/1975, Core Center.

Figure D-3. Inverse Count Rate vs Fuel Columns. 1/Net cpm on 4/10/1975 (Unirradiated) and 4/23/1975 (Irradiated)

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Figure D-4. Flux / Unit Lethargy for MKIA Fuel. Fuel at 3.2-inch Lattice Spacing

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APPENDIX E: APPROACH-TO-CRITICAL DISCUSSION AND DATA

E.1 Approach-To-Critical k_{eff} Determination

In an approach-to-critical measurement, primary neutrons emitted by a neutron source experience multiplication in the fuel. The multiplication of primary neutrons for systems close to critical is defined by the expression $1/(1-k_{\text{eff}})$. As more fuel is added to the assembly, the neutron multiplication becomes infinite and k_{eff} approaches 1.0.

Attempts will be made to correlate the experimentally determined multiplication with the assembly k_{eff} , however, such determinations will lack accuracy because the assembly will not be sufficiently close to critical.

A more realistic assignment of a k_{eff} value to substantially subcritical loading configurations will use the calculated relationship between k_{eff} and the number of fuel tubes in the assembly (Figures E-1 through E-8). As the system approaches the test limit, a better experimental prediction of the critical mass point becomes available. The calculated number of tubes to reach critical in Figure E-1 will be normalized to the experimental value, and the k_{eff} value for particular loading will be read from the normalized curve. The k_{eff} predictions will be correlated with values obtained from the pulsed-neutron source measurements.

E.2 Description of the Measurements

Fuel tubes that were prepared in other parts of the basin (see Figure 1) were moved to the test assembly via a chain hoist and crane system, and then lowered into pre-selected locations. The loading pattern was in successive concentric rings outward from the center position. At predetermined fuel loadings, neutron count-rate data and pulsed-neutron responses were gathered. The data were plotted as inverse count rates as a function of the number of fuel tubes loaded. As the multiplication increased, loading measurements were determined by the approach-to-critical guidance. Pulsed-neutron data were acquired at higher k_{eff} values.

During the measurements with green fuel, fission chambers were in locations 2, 3, 4, and 5 and a proportional counter was used in location 1 (see Figure 6). For spent fuel measurements, only fission chambers were used in all locations. During unloading, the central fuel tube was replaced with a detector. The fuel tube loading plans are shown in Figures 11 and 12. Unloading was the reverse of the loading sequence: last in, first out. The detector in the center position was employed to gather information essential for spatial correction factors. (See Section 1.4.2 of the main text.)

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The approach-to-critical data is plotted in the 8 figures in this appendix. The figures were taken out of Reference 3. For green, as well as spent fuel, these are the following plots: (there are two plots per page, even number is at the top, odd number is at the bottom)

E1. Green MKIA Fuel in Center Location	Inverse count rate vs. # of fuel tubes
E2. Green MKIA Fuel in Center Location	Number of fuel tubes divided by count rate vs. # of fuel tubes
E3. Green MKIA Detector 3 In center local	Inverse count rate vs. # of fuel tubes
E4. Green MKIA Detector 3 in center local	Number of Fuel Tube divided by count rate vs. # of fuel tubes
E5. Spent MKIA Fuel in Center Location	Inverse count rate vs number of fuel tubes
E6. Spent MKIA Fuel in Center Location	Number of fuel tubes divided count rate vs number of fuel tubes
E7. Spent MKIA Detector in Center Location	Inverse count rate vs number of fuel tubes
E8. Spent MKIA Detector in Center Location	Number of fuel tubes divided by count rate vs number of fuel tubes.

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Figure E-1. Unirradiated Fuel Approach to Critical Loading

Figure E-2. Unirradiated Fuel Approach to Critical Loading

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Figure E-3. Unirradiated Fuel Approach to Critical Unloading

Figure E-4. Unirradiated Fuel Approach to Critical Unloading

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Figure E-5. Irradiated Fuel Approach to Critical Loading

Figure E-6. Irradiated Fuel Approach to Critical Loading

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Figure E-7. Irradiated Fuel Approach to Critical Unloading

Figure E-8. Irradiated Fuel Approach to Critical Unloading

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The extrapolated curves clearly indicate that spent fuel requires significantly higher masses than green fuel for criticality.

The three-dimensional MCNP model replicated the test assembly. Each fuel tube was individually modeled containing two MKIA fuel assemblies. The fuel was composed of the isotopic compositions for green fuel listed in Table 1. Fuel, cladding, fuel tubes, and structural components incorporated the major impurity contributors. Detector and source positions were modeled to record detector responses as fuel tubes were loaded into the test assembly. The assembly was immersed in water to provide infinite reflection on top and on the sides. The bottom reflector consisted of water, a stainless steel plate, and 12-inches (30.48 cm) of concrete. Since the experiment was conducted in April, the water temperature would have been approximately 10 to 20°C. The water in the basin was circulated and filtered.

Most particulates and radioactive materials would have been removed or were in insignificant concentration levels; therefore, water was modeled as clean.

The approach-to-critical results support the conclusion that fuel elements exposed to 2865 MWd/t are much less reactive than unexposed fuel, and the critical mass for spent fuel is twice the amount of green fuel.

APPENDIX F: TECHNICAL DETAILS, PULSED-NEUTRON SYSTEM

This appendix contains a discussion of the Pulsed-Neutron Methodology and the tabulation and evaluation of the pulsed-neutron data, as described in References 1 and 3. All of the pulsed-neutron data, and graphic presentations, are taken from Reference 3. Other useful references are cited as footnotes. QuattroPro spreadsheet files are included on the distribution DVD, under the folder *Quattro*.

F.1 Pulsed-Neutron Source Theory And Measurement Technique

In the pulsed-neutron source measurements, a burst of neutrons was injected into the system in a fraction of a microsecond, and its effect on the neutron population at some point in the system was observed as a function of time following the burst. In actual practice, the measurements are made by repetitively injecting bursts of neutrons into the system and accumulating the observed time behavior of neutrons following each burst. The repetition rate, number of neutron bursts required, depends on the statistics of the accumulated data, which, in turn, depend on the particular system. A proportional counter, or a BF₃ tube, is generally used for the detection of the neutrons. For irradiated fuel, fission chambers are employed. A multi-channel analyzer was used as a series of sequentially gated scalars for the storage of the neutron counts as a function of time following each burst.

An ND2400 channel analyzer having 4096 channels was used for data accumulation in the Critical Mass Lab's (CML) system. The neutron source employed was a Kaman Nuclear A808 Pulsed-neutron Source, developed especially to CML specifications. It had a relatively high yield of 10⁹ n/sec, was repetitively pulsed up to 50 pulses per second, and was encased in a six-inch diameter cylinder 36-inches-long, to ease mobility.

To determine the degree of subcriticality of a system, the raw data obtained from a pulsed-neutron source measurement should, ideally, look something like that shown in Figure F-1. Each data point shown represents the total number of neutrons counted during a time interval positioned at that point in time between the neutron bursts. Since the neutron detector generally used is primarily sensitive to thermal neutrons, the count rate may be seen to increase immediately following the neutron burst. As the fast neutrons slow down and become thermalized, the count rate will reach a maximum and then begin to decrease exponentially. Once an asymptotic spatial distribution of neutrons is reached in the system, the space and time solutions become separable, and neutron point kinetics become a good model for predicting the time behavior of the system. The neutron point-kinetics equations can be written for one neutron energy group and one group of delayed-neutron precursors as

$$\begin{aligned}\frac{dn(t)}{dt} &= \frac{\rho - \beta}{\Lambda} n(t) + \lambda C(t) \\ \frac{dC(t)}{dt} &= \frac{\beta}{\Lambda} n(t) - \lambda C(t)\end{aligned}\tag{F.1}$$

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Figure F-1. Typical Pulsed-Neutron Source Measurement Data

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The analytical solution to this system can be written as a fundamental matrix solution, where the column vectors are the eigenvectors of the matrix, each of which is multiplied by an exponential function of time, with the time coefficient being the eigenvalue[†]. Both $n(t)$ and $C(t)$ can be expressed as a linear combination of these two exponential terms. The total neutron count rate, as a function of time following the neutron burst, can then be described as:

$$n(t) = A_1 e^{-\alpha_1 t} + A_2 e^{-\alpha_2 t} \quad (\text{F.2})$$

One term describes the time behavior of the prompt-neutrons, and the other describes the time behavior of the delayed-neutrons. Because the decay of the delayed-neutrons is very slow compared to that of the prompt-neutrons, and the measured delayed neutron level is actually a superposition of $C(t)$ arising from many previous bursts, Equation (F.2) is generally written as

$$n(t) = A_1 e^{-\alpha_1 t} + N_D, \quad (\text{F.3})$$

where N_D is a constant equal to the averaged delayed-neutron contribution and

$$A_1 e^{-\alpha_1 t}$$

is the prompt-neutron contribution. In the first point-kinetics equation (F.1), if the delayed-neutron term is assumed to be a constant, the equation can be solved approximately over a short time interval by using an integrating factor:

$$\int_0^t \frac{d}{dt} \left[e^{-(\rho-\beta)\frac{t}{\Lambda}} n(t) \right] dt = \int_0^t e^{-(\rho-\beta)\frac{t}{\Lambda}} \lambda C dt \quad (\text{F.4})$$

The solution may be written as

$$n(t) = n(0) e^{(\rho-\beta)\frac{t}{\Lambda}} + \frac{\Lambda}{\rho-\beta} \left(e^{(\rho-\beta)\frac{t}{\Lambda}} - 1 \right) \lambda C(0) \quad (\text{F.5})$$

The time coefficient in the prompt-neutron exponential, $-\alpha_1$, is therefore seen to be

$$-\alpha_1 = (\rho - \beta) / \Lambda = \left(\frac{k-1}{k} - \beta \right) \frac{k}{\ell} = \frac{(k-1-\beta k)}{\ell} = -\frac{1-k(1-\beta)}{\ell} \quad (\text{F.6})$$

After a period of time, the prompt-neutron portion of the total population that is due to the neutron burst will approach the background level preceding the burst, and only the delayed-neutrons will remain. Since the prompt-neutrons must all decay away before the delayed-neutron contribution can be determined, the prompt-neutron decay rate is a limiting factor in determining the pulse repetition rates and counting time intervals to be used. This rate of decay is dependent

[†] Braun, M., 1978, *Differential Equations and Their Applications – An Introduction to Applied Mathematics*, 2nd edition, Ch 3, Springer-Verlag, Flushing, New York.

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upon the system on which the measurements are being made. However, the faster the neutron spectrum is, the faster will the decay rate be. Also, the further subcritical a system is, the faster the decay rate will be. It is this ambiguous feature of the prompt-neutron decay which prevents it being used as a direct measurement of reactivity, although it is directly related to the reactivity of the given system. The rate of decay of the prompt-neutrons can, however, be used to obtain an effective multiplication constant, k_{eff} , for the system by the following relationship:

$$\alpha_0 = \frac{1 - k_{\text{eff}} (1 - \beta_{\text{eff}})}{\ell}, \quad (\text{F.7})$$

where α_0 is the decay rate of $-\alpha_t$, β_{eff} is the effective delayed-neutron fraction, and ℓ is the neutron lifetime in the system. However, this requires knowledge of ℓ and β_{eff} and ℓ is as difficult to calculate as k_{eff} . However, as will be seen later, Equation (F.7) is useful in analyzing those plant systems that defy any other type of analysis.

Equation (F.7) and the definition for reactivity,

$$\rho = \frac{k-1}{k}, \quad (\text{F.8})$$

can be combined to obtain an expression for the reactivity of the system in units of β , i.e., dollars.

$$\frac{\rho}{\beta} = -\frac{\alpha_0 - k \beta / \ell}{k \beta / \ell} \quad (\text{F.9})$$

If the neutron generation time Λ is assumed to be constant, then Equation (F.9) reduces to

$$\frac{\rho}{\beta} = -\frac{\alpha_0 - \alpha_c}{\alpha_c}, \quad (\text{F.10})$$

where α_c is the prompt-neutron decay rate at delayed critical ($k=1$).

However, the generation time varies significantly with the degree of subcriticality. Also α_c is a parameter not generally available for plant systems.

Three methods have been developed whereby the reactivity of a system can be determined directly from raw data such as that shown in Figure F-1. All three methods are similar: in each case, the reactivity of the system is related to the area under the curve generated by the pulsed-neutron source measurement. The earliest technique was suggested by Sjöstrand, who simply separated the data into a prompt-neutron component and a delayed-neutron component, as implied by Equations (F.2) and (F.3), and then approximated the reactivity by taking the ratio of the area under the prompt-neutron curve to the area under the delayed-neutron curve,

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$$\left(\frac{\rho}{\beta}\right)_{SJ} = -\frac{\text{Prompt Neutron Area}}{\text{Delayed Neutron Area}} \quad (\text{F.11})$$

Gozani improved on this by extrapolating the prompt-neutron fundamental mode of decay back to time zero following the burst of neutrons, and taking the reactivity as

$$\left(\frac{\rho}{\beta}\right)_{GO} = -\frac{\text{Prompt Neutron Fundamental Mode Area}}{\text{Delayed Neutron Area}}. \quad (\text{F.12})$$

Garelis and Russell developed a more rigorous theoretical model and showed that the prompt-neutron count rates multiplied by a factor involving the reactivity of the system must equal the total area under the pulsed-neutron source curve between time zero and $1/R$ following the burst of neutrons.

$$\int_0^{1/R} N_p e^{\frac{k_\beta}{\ell} t} dt = \int_0^{1/R} N_p dt + \frac{N_D}{R}, \quad (\text{F.13})$$

where, N_p is the prompt-neutron count rate at time t following the burst, N_D is the averaged delayed-neutron count rate, and R is the pulse repetition rate.

All the information needed to solve Equation (F.13) for $k\beta/\ell$ is available in the data obtained from the pulsed-neutron source measurement. Consequently, all the data needed to solve Equation (F.9) for the reactivity of the system is provided by the pulsed-neutron source data in the Garelis-Russell model. No other information is needed about the system.

It is possible to capture the neutron thermalization process within the prompt-neutron population by explicitly employing two neutron energy groups, fast and thermal. Starting from the time-dependent neutron diffusion equations, neutron point-kinetics equations can be written for two neutron energy groups and one group of delayed-neutron precursors as

$$\begin{aligned} \frac{dn_1}{dt} + v_1(\Sigma_{R1} + D_1 B^2) n_1 &= (1 - \beta) (v\Sigma_{f1} v_1 n_1 + v\Sigma_{f2} v_2 n_2) + \lambda C(t) \\ \frac{dn_2}{dt} + v_2(\Sigma_{a2} + D_2 B^2) n_2 &= v_1 \Sigma_{1 \rightarrow 2} n_1 \\ \frac{dC}{dt} &= \beta (v\Sigma_{f1} v_1 n_1 + v\Sigma_{f2} v_2 n_2) - \lambda C(t) \end{aligned} \quad (\text{F.14})$$

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This problem may be put into matrix form:

$$\begin{aligned} \frac{d}{dt} \begin{bmatrix} n_1 \\ n_2 \\ C \end{bmatrix} &= \begin{bmatrix} (1-\beta) v_1 \nu \Sigma_{f1} - v_1 (\Sigma_{R1} + D_1 B^2) & (1-\beta) v_2 \nu \Sigma_{f2} & \lambda \\ v_1 \Sigma_{1 \rightarrow 2} & -v_2 (\Sigma_{a2} + D_2 B^2) & 0 \\ v_1 \nu \Sigma_{f1} \beta & v_2 \nu \Sigma_{f2} \beta & -\lambda \end{bmatrix} \begin{bmatrix} n_1 \\ n_2 \\ C \end{bmatrix} \\ &= \bar{A} \begin{bmatrix} n_1 \\ n_2 \\ C \end{bmatrix} \end{aligned} \quad (\text{F.15})$$

The system buckling, B^2 , may be written for finite cylindrical geometry as

$$B^2 = \left(\frac{\pi}{\hat{H}} \right)^2 + \left(\frac{2.4048}{\hat{R}} \right)^2 \quad (\text{F.16})$$

The core radius, adjusted for reflector savings, depends upon the number of fuel tubes added to the core, given the 3.2-inch hexagonal lattice spacing, and the axial leakage corresponds to the height of two MKIA fuel assemblies.

This system of differential equations can be solved simultaneously by using the eigenvalue / eigenvector method. First, the characteristic polynomial of the matrix is found from

$$\det(\bar{A} - \alpha \bar{I}) = 0 \quad (\text{F.17})$$

This is a cubic polynomial that may be solved exactly using known solutions^{ss}. The three roots of this polynomial correspond to behavior associated with neutron thermalization, (i.e., fast prompt-neutron decay), thermal prompt-neutron decay, and delayed-neutron decay. Next, the three eigenvectors associated with each of the three roots are calculated. These vectors satisfy the equations

$$(\bar{A} - \alpha_{1,2,3} \bar{I}) \vec{e}_{1,2,3} = 0 \quad (\text{F.18})$$

Once the eigenvectors are found, the solution may be formally written, using the fundamental matrix solution:

$$\begin{bmatrix} n_1(t) \\ n_2(t) \\ C(t) \end{bmatrix} = e^{\bar{A}t} \begin{bmatrix} n_1(t_o) \\ n_2(t_o) \\ C(t_o) \end{bmatrix} = \bar{\Psi}(t) \bar{\Psi}^{-1}(t_o) \begin{bmatrix} n_1(t_o) \\ n_2(t_o) \\ C(t_o) \end{bmatrix} \quad (\text{F.19})$$

^{ss} CRC 1979, *Standard Mathematical Tables*, 25th edition, CRC Press, Inc., Arvida Executive Center, Boca Raton, Florida.

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The column vectors of the fundamental matrix solution are the eigenvectors of the system, multiplied by the exponential time functions:

$$\overline{\Psi}(t) = (\vec{e}_1 e^{\alpha_1 t} \quad \vec{e}_2 e^{\alpha_2 t} \quad \vec{e}_3 e^{\alpha_3 t}) \quad (F.20)$$

This solution was coded into a spreadsheet model and used to predict the thermal prompt-neutron decay rates and population curves as a function of time, for a given number of fuel tubes added to the core (which enters into the model through the radial buckling term). Two-group macroscopic cross sections were obtained from the WIMSD-5B lattice code. Figure F-2 shows the model predictions of thermal-neutron decay rate as a function of number of fuel tubes added for both unirradiated and irradiated fuel. (These are the alpha values documented in the old experimental writeup.) Figure F-3 shows the predicted neutron population curves for 50 unirradiated fuel tubes loaded. For 50 unirradiated tubes added, the three-equation model of Equation (F.19) predicts a thermal prompt neutron decay rate of -1278 s^{-1} . The model also predicts the rate of fast neutron loss. This eigenvalue is $-1.827 \times 10^6 \text{ s}^{-1}$.

Figure F-2. Analytical and Measured (Old) Thermal Neutron Decay Rates versus Fuel Tubes

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Figure F-3. Fast, Thermal Prompt, Delayed, and Total Neutrons vs. Time (50 Fuel Tubes Added)

Figure F-4. Analytical and Measured (New) Thermal Neutron Decay Rates versus Fuel Tubes Added

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Figure F-4 shows the analytical predictions for thermal neutron decay rate versus fuel tubes added, compared to the measured values, as obtained from the new analysis of the pulse-neutron data.

F.2 Pulsed Neutron Raw Data

The pulsed neutron data, associated with the measurements, is listed in this appendix. Key indicators are present so the data can be related to specific fuel loadings. The data lends itself for further analysis.

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DATE: 4-4-75.
 PULSE NUMBER: 4475-2.
 PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PNS TRIGGER.
 PULSE NEUTRON SOURCE: KAMAN 808 NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.
 DETECTOR LOCATION: EAST MIDPOINT (30'CLOCK).

LOADING: LATTICE EMPTY.

MICROSEC/CHANNEL: 20.

PRE-PULSE COUNT RATE: SCALER #1 303/80SEC.
 SCALER #2 55/80SEC.
 SCALER #3 0/80SEC.

TOTAL TIME: 400.2SEC . BUILDUP TIME: 0 SEC.

TOTAL PULSES: 8566.

ANALYSER CYCLES: 8571.

TOTAL COUNTS ALL CHANNELS: 117748

0										
1	4610	5033	5357	5474	5549	5267	5167	4879	4671	
4362	4196	4143	3851	3628	3533	3295	3107	2923	2746	
2521	2347	2259	2160	1999	1768	1658	1531	1499	1392	
1234	1166	1056	1059	985	865	770	721	670	610	
608	496	542	463	404	410	345	329	319	307	
246	217	242	218	180	162	150	152	121	120	
106	98	85	103	84	65	81	80	62	69	
48	57	39	47	29	41	39	31	24	26	
31	20	29	18	23	17	14	9	9	15	
11	9	9	11	7	6	6	6	8	9	
100										
3	7	6	5	3	4	5	4	1	4	
4	1	3	1	2	1	0	0	1	1	
1	0	4	0	2	3	0	1	0	0	
0	0	1	1	1	1	1	2	1	2	
1	2	0	1	0	1	2	1	0	1	
3	1	2	2	0	0	2	0	1	0	
2	0	1	2	1	1	1	0	0	1	
2	0	1	1	0	0	0	0	1	0	
3	0	0	0	1	3	0	0	0	0	
2	0	0	0	0	1	0	0	1	1	
200										
1	0	3	2	1	1	0	3	0	1	
2	2	0	2	1	1	0	1	1	0	
1	0	0	0	0	0	0	0	0	1	
3	0	0	0	1	3	1	1	1	0	
0	1	0	0	0	2	4	0	2	1	
1	0	0	2	0	2					

POST PULSE COUNT RATE: SCALER #1 339,299,290,321/80SEC .
 SCALER #2 56,44,61,59/80SEC .
 SCALER #3 0/80SEC .

BACKGROUND

COUNTING TIME: 407.5 SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 1483 .
 SCALER #2: 263 .
 SCALER #3 2 .

ANALYSER CYCLES: 6730.

TOTAL COUNTS ALL CHANNELS: 166

0

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DATE: 4-4-75.
PULSE NUMBER: 4475-3.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED: 20 MICROSECONDS FOLLOWING PNS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN 808 NO. 083.

DETECTOR: FISSION CHAMBER LASL #310.
DETECTOR LOCATION: WEST MIDPOINT (9 O'CLOCK).

LOADING: ASSEMBLY EMPTY.

MICROSEC/CHANNEL: 20.

PRE-PULSE COUNT RATE: SCALER #1 303/80SEC.
SCALER #2 55/80SEC.
SCALER #3 0/80SEC.

TOTAL TIME: 700.6 SEC .BUILDUP TIME: 0 SEC.

TOTAL PULSES: 13029.

ANALYSER CYCLES: 13034.

TOTAL COUNTS ALL CHANNELS: 7118

0									
5	143	214	249	231	278	285	278	270	269
262	236	210	225	216	217	167	167	162	161
167	141	135	150	114	113	105	97	106	98
87	80	68	63	53	58	48	49	50	48
41	41	25	29	28	24	23	24	30	25
24	18	21	16	19	12	12	11	18	9
17	7	11	7	6	5	9	2	3	7
4	10	6	6	1	11	8	3	6	5
4	8	6	6	1	5	2	3	2	3
2	5	2	6	2	4	3	4	1	2
100									
4	6	2	1	3	0	2	1	3	1
1	3	4	5	1	1	2	1	3	3
0	3	1	3	3	5	4	2	2	3
7	3	3	4	3	4	3	1	5	4
2	3	3	4	2	3	5	1	2	3
0	6	1	5	3	4	3	3	2	1
1	1	2	3	1	1	3	2	4	1
4	7	2	3	3	1	2	6	2	3
1	4	0	1	3	0	2	0	2	2
4	6	1	4	2	3	2	3	1	2
200									
1	1	2	1	0	3	2	5	5	2
1	3	3	0	1	6	1	2	3	2
1	4	4	2	4	3	2	0	2	3
4	2	5	4	4	3	2	0	1	5
1	0	6	2	1	3	1	1	3	2
1	1	2	0	3	1				

POST PULSE COUNT RATE: SCALER #1 DISCONNECTED FOR USE WITH HE COUNTER .
SCALER #2 55.64.61/80SEC .
SCALER #3 0/80SEC .

BACKGROUND

COUNTING TIME: 400.0 SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 N.A. .
SCALER #2: N.A. .
SCALER #3 0/80SEC .

ANALYSER CYCLES: 9376 (TOTAL TRIGGERS DURING THIS TIME WAS 7456
TIMER WAS RESET; HOWEVER ANALYSER MUST NOT HAVE BEEN, CONSEQUENTLY
TIME IS LONGER THAN 400SEC.)

TOTAL COUNTS ALL CHANNELS: 314

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DATE: 4-5-75.

PULSE NUMBER: 4575-1.

PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PNS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN 808 NO. 083.DETECTOR: B-PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: CENTER + 6 # 1 TUBES..

MICROSEC/CHANNEL: 20.

PRE-PULSE COUNT RATE: SCALER #1 309/80SEC.
SCALER #2 65/80SEC.
SCALER #3 0/80SEC.

TOTAL TIME: 2000.1 SEC .BUILDUP TIME: 0 SEC.

TOTAL PULSES: 4350.

ANALYSER CYCLES: 4351.

TOTAL COUNTS ALL CHANNELS: 36278

0									
0	1019	1505	1657	1718	1695	1774	1679	1582	1573
1520	1366	1335	1280	1184	1083	1010	942	897	870
760	730	672	581	601	534	506	459	407	424
374	316	325	309	261	233	244	203	212	187
168	145	136	121	113	120	99	87	85	87
89	65	43	73	65	48	40	47	31	41
37	34	26	46	27	32	25	16	14	10
16	16	15	14	10	11	15	12	11	5
14	3	4	7	3	8	4	3	4	6
4	4	3	2	2	0	4	2	2	2
100									
3	2	3	1	0	3	1	0	1	1
0	1	1	2	0	0	0	0	0	1
3	2	0	1	0	1	0	1	1	0
0	1	0	1	0	1	2	1	2	0
0	1	0	0	1	0	1	1	0	0
0	0	0	0	0	0	1	1	0	1
0	0	0	0	0	0	0	1	1	0
0	1	0	0	0	0	1	0	0	0
0	1	0	1	0	1	0	0	0	0
0	0	2	0	0	0	0	0	0	2
200									
0	1	0	0	0	0	0	0	0	1
2	1	1	0	1	0	1	0	0	0
0	0	1	0	2	0	1	0	0	0
0	1	0	1	1	0	0	1	0	1
0	0	1	0	0	0	0	1	0	0
0	0	1	0	0	0				

POST PULSE COUNT RATE: SCALER #1 322/80SEC .
SCALER #2 70/80SEC .
SCALER #3 5/80SEC .

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DATE: 4-5-75.

PULSE NUMBER: .4575-2

PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PNS TRIGGER.

PULSE NEUTRON SOURCE: KAMAN 808 NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.

DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 19 FUEL COLUMNS. CENTER + ALL 1'S, 2'S, AND 3'S..

MICROSEC/CHANNEL: 20

PRE-PULSE COUNT RATE: SCALER #1 457,472,472/80SEC.

SCALER #2 56,86,77/80SEC.

SCALER #3 0,0,1/80SEC .

TOTAL TIME: 1500.6 SEC . BUILDUP TIME: 0 SEC.

TOTAL PULSES: 3261.

ANALYSER CYCLES: 3262.

TOTAL COUNTS ALL CHANNELS: 32066

0									
2	762	1317	1389	1495	1445	1497	1436	1437	1362
1248	1173	1191	1100	1016	1011	892	865	777	712
668	676	615	536	532	496	462	416	366	377
354	308	263	270	270	252	228	195	211	171
146	154	150	115	130	104	114	93	77	77
69	71	70	54	60	56	42	49	26	33
39	38	26	30	34	26	21	19	19	13
10	22	11	14	12	13	16	13	6	8
11	3	11	7	6	6	5	4	4	10
4	7	4	4	8	5	3	1	2	2
100									
1	4	2	2	3	1	1	1	2	3
0	3	3	4	2	0	1	0	0	1
1	0	0	0	0	2	0	1	1	0
1	1	1	0	0	1	0	2	0	0
0	1	1	0	1	0	0	0	0	0
2	0	0	0	0	1	0	0	0	3
10	0	0	0	0	0	0	1	1	0
0	0	0	0	0	0	0	0	0	0
1	1	1	0	0	0	0	1	0	0
0	0	1	0	0	0	0	0	0	0
200									
1	1	0	1	1	0	0	0	0	0
0	0	1	0	0	0	0	0	1	0
2	1	0	0	0	0	0	1	0	1
1	0	0	1	0	2	0	0	0	0
0	0	0	2	1	1	0	1	0	1
0	0	0	1	0	1				

POST PULSE COUNT RATE: SCALER #1 445 PER 80 SEC .

SCALER#2 66 PER 80 SEC .

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DATE: 4-7-78.
PULSE NUMBER: 4775-1
PULSED NEUTRON SOURCE MEASUREMENTS - IN REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE, KAMAN SCS NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 99 37 FUEL COLUMNS .

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 6359,9433,9541/80SEC.
SCALER #2 145,123,123/80SEC.
SCALER #3 4.9,3/80SEC .

TOTAL TIME: 600.5 SEC .BUILDUP TIME: 120 SEC.

TOTAL PULSES: 11838.

ANALYSER CYCLES: 9472.

TOTAL COUNTS ALL CHANNELS: 821879

0	15711	24248	21015	18318	16143	13541	11447	9217	7696
100	5128	4300	3525	2994	2448	2063	1757	1440	1196
1057	893	751	667	565	496	439	350	349	320
281	235	216	215	180	158	173	150	136	142
100	123	117	102	106	90	117	92	100	101
109	94	120	106	99	106	82	81	104	98
94	88	92	120	84	109	96	89	96	92
90	106	80	95	93	101	103	94	81	91
111	106	112	97	98	108	99	89	106	92
99	99	74	109	99	84	100	95	77	106
94	84	90	86	88	107	90	90	115	100
83	111	85	108	104	88	87	110	86	94
104	82	108	108	104	93	89	95	86	105
75	112	79	80	86	101	110	83	108	114
90	102	98	87	88	93	90	104	108	107
99	110	96	87	91	105	88	97	95	88
98	84	87	100	95	94	83	86	85	99
105	86	101	95	100	97	113	89	106	81
106	101	85	86	96	88	94	89	109	100
106	102	96	99	102	99	102	99	86	87
93	92	84	95	84	100	102	87	101	121
94	95	96	94	88	89	95	112	97	95
92	85	100	101	97	104	104	91	102	97
95	116	87	96	102	87	81	98	91	108
104	87	81	89	75	87	109	101	104	93
80	108	84	101	105	98	81	92	96	101
88	106	95	84	89	104	87	107	106	106
73	113	99	87	102	111	80	95	98	107
76	98	102	91	107	97	99	102	84	108
109	102	103	76	110	86	100	85	92	92
100	102	89	98	100	101	99	97	94	80
91	84	109	95	84	80	123	108	80	89
101	103	68	80	83	102	108	89	96	103
95	101	87	103	93	99	92	92	83	90
92	100	96	99	97	73	90	106	97	88
95	112	101	96	99	91	90	86	85	104
117	91	103	109	91	89	93	89	90	71
113	120	103	94	90	111	94	119	81	102
114	92	99	109	91	103	97	99	94	81
107	95	94	95	88	97	93	105	79	85
92	70	110	90	109	116	104	111	88	99
103	110	99	91	116	117	76	90	103	86
97	106	101	80	88	94	100	86	95	81
106	102	83	102	101	90	75	95	113	79
86	93	94	119	89	94	90	80	103	98
99	94	82	117	91	93	88	96	109	95
96	96	85	83	98	111	114	107	114	95
102	115	95	93	103	96	112	105	99	82
102	94	92	104	79	106	89	86	101	108
91	115	89	98	101	87	102	87	96	103
500									
80	105	104	91	107	84	102	110	81	105
91	84								

POST PULSE COUNT RATE: SCALER #1 9542,9395,9529,9453/80SEC .
SCALER #2 126,152,118,133/80SEC .
SCALER #3 5.9,6.4/80SEC .

BACKGROUND

COUNTING TIME: 479.8 SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 56887 .
SCALER #2: 875 .
SCALER #3 27 .

ANALYSER CYCLES: 9475.

TOTAL COUNTS ALL CHANNELS: 46949

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DATE: 4-7-75.
PULSE NUMBER: 4775-3.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAHAN EOS NO. 063.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 43 FUEL COLUMNS.

MICROSEC/CHANNEL: 60

PRE-PULSE COUNT RATE: SCALER #1 92449.4 12602.12556.12181/60SEC.
SCALER #2 166.196.160/60SEC.
SCALER #3 6.8.10/60SEC.

TOTAL TIME: 720.43EC .BUILDUP TIME: 120SEC.

TOTAL PULSES: 14262.

ANALYSER CYCLES: 11817.

TOTAL COUNTS ALL CHANNELS: 347447

152	21335	32820	29915	26413	22907	19597	16775	13998	11962
10273	8750	7632	6419	5641	4880	4184	3683	3148	2584
2426	2144	1871	1625	1416	1304	1071	1025	894	790
664	631	598	500	414	419	374	331	337	293
294	267	242	229	222	217	235	225	200	170
173	183	181	172	200	165	153	146	163	175
174	146	165	149	162	153	170	172	167	170
153	134	139	164	146	170	156	149	171	145
143	129	165	149	143	163	141	153	146	148
159	165	164	150	181	174	151	162	151	159
100									
179	167	167	163	175	170	142	161	147	143
158	141	179	169	168	162	181	142	154	178
149	154	160	143	155	163	155	134	145	130
149	165	142	155	165	168	151	168	141	154
151	166	162	120	184	176	173	132	157	143
155	154	164	144	165	164	165	152	149	156
164	160	179	134	185	146	165	141	145	157
167	164	162	144	137	160	145	164	155	140
168	175	172	173	185	168	157	141	154	175
163	173	197	127	172	172	150	146	140	173
200									
177	153	170	177	138	155	163	172	162	163
165	177	157	149	137	181	152	171	148	148
154	162	164	143	159	140	153	160	144	156
149	172	136	149	153	159	156	148	152	161
147	165	147	154	159	159	153	148	169	164
141	135	166	146	160	150	166	143	155	152
156	176	169	152	154	154	148	155	162	162
159	150	177	153	162	169	152	159	159	169
148	145	142	161	154	161	140	127	169	149
176	147	143	176	166	142	153	163	143	161
300									
156	164	146	152	162	171	150	166	175	160
150	163	160	149	143	180	153	171	151	150
139	165	143	144	173	161	175	174	134	176
138	146	137	172	145	127	131	177	166	151
152	159	156	152	152	167	171	135	155	157
165	137	163	152	141	165	161	161	161	144
153	155	139	169	165	164	161	172	149	169
146	177	132	176	161	163	149	153	154	149
137	147	165	161	160	140	162	164	138	157
146	129	145	144	142	140	158	162	174	157
400									
142	154	173	137	160	128	132	164	151	157
151	156	154	164	154	157	163	154	173	147
155	151	157	142	166	136	136	169	161	160
164	152	163	166	147	162	146	136	145	131
159	146	160	179	157	163	153	147	142	155
141	165	166	171	152	163	169	152	175	149
136	168	154	157	163	149	155	156	149	161
158	173	136	169	146	151	121	165	164	174
151	135	143	172	165	147	156	155	169	157
151	154	162	146	145	164	156	166	164	161
500									
170	159	167	143	150	176	171	156	166	156
136	150								

POST PULSE COUNT RATE: SCALER #1 11923.11610.11538/SEC .
SCALER #2 166.186.191/60SEC .
SCALER #3 6.8.7/60SEC .

BACKGROUND

COUNTING TIME: 619.83EC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 90444 .
SCALER #2: 1430 .
SCALER #3 47 .

ANALYSER CYCLES: 12303.

TOTAL COUNTS ALL CHANNELS: 73767

HNF-29648, Rev. 1

DATE: 4-7-75.
PULSE NUMBER: 4775-4. DETECTOR MAY HAVE MOVED DURING 4775-3 MEASUREMENT.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN EOS NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 30'CLOCK.

LOADING: 43 FUEL ELEMENTS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 14971, 14946, 14681/80SEC.
SCALER #2 186, 190, 186/80SEC.
SCALER #3 8, 8, 10/80SEC.

TOTAL TIME: 520.15EC .BUILDUP TIME: 120SEC.

TOTAL PULSES: 11730.

ANALYSER CYCLES: 9021.

TOTAL COUNTS ALL CHANNELS: 281430

153	16823	24939	21860	19039	17113	15051	13043	11049	9625
5190	6976	5875	5229	4633	3949	3380	2933	2601	2229
2009	1705	1450	1325	1188	1006	992	749	705	657
589	512	475	443	362	353	318	317	306	263
247	234	242	220	204	199	168	170	190	167
178	160	172	163	162	166	127	144	165	140
154	167	155	146	146	167	167	173	149	143
132	178	151	135	142	147	140	146	154	141
129	128	143	168	135	166	149	163	160	155
153	164	155	146	150	164	170	156	160	130
100									
133	131	146	149	157	139	151	161	157	143
153	158	133	159	178	165	155	140	164	146
144	160	144	145	129	154	156	146	147	154
147	149	169	147	166	155	136	143	145	145
163	166	144	146	140	150	131	135	158	146
175	158	155	175	150	128	146	137	133	155
134	168	138	148	147	125	147	121	161	142
146	141	141	168	154	133	135	146	152	153
131	155	147	150	147	166	156	133	132	150
165	180	158	138	145	154	157	144	149	124
200									
126	147	125	156	141	167	153	145	147	146
138	163	148	153	164	168	142	146	141	156
160	154	133	134	136	145	133	161	140	147
157	149	127	144	155	140	163	137	161	141
139	155	162	132	157	152	151	152	153	154
131	132	132	144	161	154	166	144	149	138
140	127	133	127	143	174	175	174	153	145
130	120	140	139	155	149	164	153	138	144
134	143	181	154	139	150	152	154	136	145
166	157	149	136	141	165	155	161	145	162
300									
168	134	138	166	152	163	143	124	146	153
171	155	162	166	149	142	147	161	146	171
145	156	143	143	166	139	144	151	145	150
135	171	129	155	139	137	143	126	144	130
130	164	162	122	161	165	123	136	157	159
157	151	160	144	144	138	146	149	138	167
157	154	136	143	140	151	170	150	149	123
122	152	146	159	154	148	131	148	153	135
171	156	160	151	165	150	168	132	161	149
154	158	151	142	157	141	148	161	136	168
400									
161	160	134	139	148	150	153	126	146	146
164	146	153	134	155	146	155	142	160	151
129	134	169	143	158	145	157	149	151	157
140	154	170	138	154	143	147	144	147	143
153	142	139	134	154	127	138	160	130	140
151	137	137	174	160	138	153	142	141	173
137	160	157	140	146	135	135	173	166	144
146	145	150	139	170	131	167	152	154	158
152	146	153	155	157	141	143	159	137	135
146	130	141	135	170	136	152	155	125	166
500									
146	133	148	153	148	141	143	134	150	146
149	157								

POST PULSE COUNT RATE: SCALER #1 14711, 14899, 15063 /80SEC.
SCALER #2 196, 196, 170/80SEC.
SCALER #3 8, 7, 3/80SEC.

BACKGROUND

COUNTING TIME: 1400.0SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 73776.
SCALER #2: 972.
SCALER #3 28.

ANALYSER CYCLES: 9031.

TOTAL COUNTS ALL CHANNELS: 68878

HNF-29648, Rev. 1

DATE: 4-7-75.
PULSE NUMBER: 4775-8.
PULSED NEUTRON SOURCE MEASUREMENTS - IN REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

WD 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAHAN 805 NO. 053.

DETECTOR: PG LASH # 310.
DETECTOR LOCATION: 9 O'CLOCK.

LOADING: 43 FUEL COLUMNS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 1202, 1227, 1253/80SEC.
SCALER #2 11219, 11213, 11646/80SEC.
SCALER #3 11, 4, 6, /80SEC.

TOTAL TIME: 557.45EC . BUILDUP TIME: 120SEC.

TOTAL PULSES: 11360.

ANALYSER CYCLES: 8915.

TOTAL COUNTS ALL CHANNELS: 13950

0										
12	421	558	910	800	723	428	509	464	406	
357	301	271	218	205	179	136	130	110	111	
92	64	61	70	57	49	40	43	37	39	
29	26	23	23	16	23	13	14	14	17	
23	11	15	11	14	15	17	10	16	19	
3	19	10	11	14	11	10	12	10	4	
11	14	13	10	20	10	2	17	23	15	
10	13	10	10	16	13	11	15	8	9	
11	10	7	11	12	13	15	11	10	19	
7	9	10	12	11	10	8	18	15	10	
100										
10	13	17	11	11	15	9	11	13	14	
12	15	9	9	14	11	10	10	8	12	
12	24	14	10	13	8	10	11	9	14	
9	15	9	12	9	12	11	9	10	8	
10	9	8	9	5	5	15	14	8	11	
7	12	11	8	15	9	19	6	14	14	
5	13	11	14	11	11	5	16	15	10	
14	12	12	7	11	15	10	16	16	13	
13	11	11	14	10	11	15	6	11	13	
15	14	13	10	7	10	10	13	10	13	
200										
12	9	10	11	10	12	9	11	10	11	
10	8	14	11	15	15	9	12	9	7	
12	17	12	15	10	13	10	9	14	14	
13	12	12	10	13	7	17	8	15	2	
10	9	11	8	11	15	13	8	12	15	
13	13	15	14	14	13	11	18	12	6	
9	17	12	20	11	13	7	9	11	6	
12	6	6	16	13	10	12	10	20	15	
10	15	13	10	17	6	12	10	12	6	
10	10	12	14	11	8	9	11	6	11	
300										
15	6	11	15	9	10	13	12	12	14	
13	5	12	8	14	13	15	6	10	14	
11	14	10	13	15	13	10	5	7	9	
8	10	11	11	9	9	14	13	13	13	
7	10	13	9	8	13	9	12	7	12	
11	11	14	14	12	20	12	12	8	10	
9	12	15	18	7	12	13	11	15	15	
15	5	11	14	10	11	5	9	20	19	
14	14	14	9	18	3	8	14	7	19	
5	12	5	10	9	11	11	10	7	12	
400										
9	12	10	10	6	14	12	15	12	11	
10	11	12	12	11	13	14	10	15	8	
6	15	9	14	11	10	11	7	8	9	
12	15	10	13	11	10	14	9	9	9	
7	6	6	9	11	6	11	14	12	20	
12	10	11	15	7	8	14	14	14	14	
9	8	12	12	13	6	10	14	13	8	
21	8	9	11	8	9	8	7	12	12	
9	9	12	12	19	16	16	13	7	12	
14	14	12	9	12	15	14	15	4	11	
500										
10	9	11	10	11	9	7	7	7	13	
14	15									

POST PULSE COUNT RATE: SCALER #1 1193, 1183, 1199/80SEC.
SCALER #2 11425, 11213, 11343/80SEC.
SCALER #3 10, 8, 2/80SEC.

BACKGROUND

COUNTING TIME: 400.0SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1= 6016.
SCALER #2= 56704.
SCALER #3 26.

ANALYSER CYCLES: 8166.

TOTAL COUNTS ALL CHANNELS: 4999

HNF-29648, Rev. 1

DATE: 4-5-78.
PULSE NUMBER: 4875-3
PULSED NEUTRON SOURCE MEASUREMENTS - W REACTOR FUEL STORAGE BASIN.

3.8 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

WD 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN 808 NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 49 FUEL COLUMNS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 25769,26178,25596./80SEC.
SCALER #2 287,229,260/80SEC.
SCALER #3 9875,9988,10114/80SEC.

BUILDUP TIME: 1255SEC.

TOTAL TIME: 520.1 SEC.

TOTAL PULSES: 11093.

ANALYSER CYCLES: 8433.

TOTAL COUNTS ALL CHANNELS: 393674

254	22682	25749	24381	21971	19465	17526	15601	13942	11992
10713	9798	8723	7909	6892	6169	5529	4949	4475	4058
3590	3139	2847	2512	2245	2020	1757	1614	1465	1420
1189	1079	986	930	833	832	725	677	617	633
531	483	464	461	413	420	358	373	349	345
337	315	298	308	294	293	313	315	292	286
285	280	266	253	290	262	266	269	219	254
273	267	248	263	261	242	216	242	266	262
245	246	247	233	261	261	247	277	249	256
229	224	267	242	240	263	235	253	257	270
100									
239	263	245	249	264	259	245	241	236	222
249	234	256	245	238	255	235	253	271	255
244	248	275	241	247	249	277	238	252	261
229	270	220	243	251	241	266	261	248	242
235	224	254	249	236	225	247	255	240	248
249	230	267	242	246	263	248	257	233	259
222	278	238	247	272	256	242	242	238	252
265	245	257	261	267	238	217	252	251	263
217	249	248	269	240	264	233	254	236	241
233	250	271	236	228	226	256	225	216	231
200									
229	244	254	230	233	233	237	253	229	258
250	272	237	270	220	266	275	230	243	240
276	269	263	259	236	246	251	237	229	239
246	264	235	270	239	251	265	243	250	246
243	276	236	237	262	247	251	248	263	236
259	257	266	231	254	254	237	243	201	287
232	244	271	232	239	279	239	275	292	266
265	239	265	256	254	229	239	275	231	250
245	259	265	256	229	228	247	250	239	230
253	263	257	239	246	264	243	243	227	248
300									
249	297	272	244	259	240	256	241	241	272
277	234	257	239	215	241	243	257	268	248
250	237	256	255	246	268	275	253	242	233
265	263	250	233	252	227	243	241	244	239
233	230	233	246	231	273	260	265	227	231
247	265	249	247	252	262	259	262	253	267
254	231	221	240	244	235	253	274	248	261
255	231	276	233	247	255	261	247	265	258
232	251	254	234	238	242	215	252	237	249
260	227	221	253	245	249	238	225	273	266
400									
238	245	234	236	238	265	257	228	254	244
262	270	261	214	241	243	228	261	259	241
229	241	234	258	249	255	254	259	270	230
238	239	249	231	250	244	267	238	223	254
258	257	250	253	235	266	221	251	248	236
265	226	254	244	234	219	220	244	232	230
249	231	252	259	262	243	244	245	243	245
249	265	240	240	262	287	215	251	250	249
245	249	232	252	230	267	282	225	260	263
261	245	253	254	236	270	262	249	252	246
500									
245	231	248	239	229	253	236	262	256	252
237	241								

POST PULSE COUNT RATE: SCALER #1 25910,26127,25848/80SEC.
SCALER #2 252,271,253/80SEC.
SCALER #3 10134,10161,10045/80SEC.

BACKGROUND

COUNTING TIME: 400SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 130428.
SCALER #2: 1346.
SCALER #3 50512.

ANALYSER CYCLES: 8568.

TOTAL COUNTS ALL CHANNELS: 114549

HNF-29648, Rev. 1

DATE: 4-6-75.
PULSE NUMBER: 4875-4.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE KAHAN 808 NO. 083.

DETECTOR: FISSION CHAMBER LASL #310.
DETECTOR LOCATION: 9 O'CLOCK.

LOADING: 49 FUEL COLUMNS.

MICROSEC/CHANNEL: 50

PRE-PULSE COUNT RATE: SCALER #1 1734.1661.1672/50SEC.
SCALER #2 19211.19527.19222/50SEC.
SCALER #3 10088.10132.10064/50SEC.

BUILDUP TIME: 1205EC.

TOTAL TIME: 659.55EC

TOTAL PULSES: 13502.

ANALYSER CYCLES: 11044.

TOTAL COUNTS ALL CHANNELS: 20197

0									
20	359	926	907	943	765	691	636	560	516
453	421	350	380	285	274	248	209	188	202
147	167	164	118	106	104	93	81	62	65
64	49	60	55	51	46	41	35	39	36
46	19	36	32	19	24	22	22	28	28
18	23	17	20	22	25	23	21	17	15
19	23	21	25	19	9	24	23	22	27
13	16	14	15	24	21	13	17	19	13
17	20	17	20	14	15	22	22	10	12
21	14	22	27	12	16	27	18	12	20
100									
15	22	25	19	20	21	16	20	18	14
20	20	24	24	23	20	25	19	23	16
16	19	22	18	18	20	18	25	24	24
23	19	20	21	18	21	19	22	20	23
18	10	21	18	21	26	23	22	22	23
23	20	22	20	21	21	22	14	19	15
15	15	19	16	16	18	16	12	14	16
13	19	20	24	16	19	16	11	20	27
17	21	24	18	19	16	24	10	24	22
17	21	23	18	20	19	11	22	17	13
200									
21	18	14	23	18	18	13	20	15	13
24	18	17	21	24	19	19	15	20	23
17	22	17	20	25	16	26	14	24	17
15	16	21	17	23	19	25	22	23	26
13	11	21	26	21	25	21	13	21	19
15	24	16	20	21	20	25	22	10	19
20	19	13	17	15	20	26	20	20	14
17	21	14	20	19	14	25	20	22	22
17	18	22	14	14	17	16	20	23	20
20	18	24	22	14	29	23	22	13	20
300									
25	19	27	32	25	23	17	16	19	22
18	12	17	21	21	17	17	17	19	21
17	20	14	22	14	13	23	24	21	15
18	24	15	25	13	24	22	12	15	21
20	18	22	13	14	20	15	14	23	16
14	17	20	26	14	22	19	16	17	20
14	19	13	18	18	17	17	22	16	20
13	23	19	20	18	18	18	24	20	19
24	19	15	24	13	23	23	21	21	18
23	23	24	26	24	24	25	21	14	15
400									
22	21	16	17	13	11	17	21	27	22
13	21	11	16	25	18	23	16	24	24
19	13	22	20	19	22	12	19	18	22
14	19	21	20	21	28	20	25	20	16
19	19	15	13	17	14	21	24	23	20
17	19	17	24	22	19	18	21	20	13
25	18	20	23	22	22	15	22	10	17
15	22	17	13	17	16	16	20	15	15
17	11	17	19	20	22	19	33	22	22
28	22	21	23	21	23	18	19	23	20
500									
20	19	11	20	25	23	11	17	19	16
16	17								

POST PULSE COUNT RATE: SCALER #1 1600.1616/50SEC.
SCALER #2 19411.19324/50SEC.
SCALER #3 10001.9931/50SEC.

BACKGROUND

COUNTING TIME: 538.75EC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 11045.
SCALER #2 131514.
SCALER #3 67427.

ANALYSER CYCLES: 11045 (CHECKED).

TOTAL COUNTS ALL CHANNELS: 9250

0

HNF-29648, Rev. 1

X
DATE: 4-9-75.
PULSE NUMBER: 4975-1.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FULL STORAGE BASIN.

3.8 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 80 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE; KAMAN 808 NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 5) FUEL COLUMNS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 30415,30667,30687/80SEC.
SCALER #2 348,330,351/80SEC.
SCALER #3 12257,12223,12196/80SEC.

WILDFIRE TIME: 120 SEC.

TOTAL TIME: 590.0 SEC.

TOTAL PULSES: 10656.

ANALYSER CYCLES: 8194.

TOTAL COUNTS ALL CHANNELS: 439175

[illegible]

POST PULSE COUNT RATE: SCALER #1 30942, 31272, 30937/80SEC .
SCALER #2 329, 338, 357/80SEC .
SCALER #3 12369, 12176, 12211/80SEC .

BACKGROUND

COUNTING TIME: 400SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 155603 .
SCALER #2 1645 .
SCALER #3 60793 .

ANALYSE CYCLES: 0204.

TOTAL COUNTS ALL CHANNELS: 130852

HNF-29648, Rev. 1

DATE: 4-9-75.
PULSE NUMBER: 4975-2.
PULSED NEUTRON SOURCE MEASUREMENTS - M REACTOR FUEL STORAGE BASIN.

3.8 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

WD 2400 ANALYSER DELAYED 90 MICROSECONDS FOLLOWING PWS TRIGGER.
PULSE NEUTRON SOURCE: KAMAM 808 WD. 063.

DETECTOR: FISSION CHAMBER LAST #241.
DETECTOR LOCATION: RED 10 (ABOUT 1210 O'CLOCK).

LOADING: 51 FUEL COLUMNS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 22554, 22492, 22522/80SEC.
SCALER #2 327,350,303/80SEC.
SCALER #3 23911, 23749, 23676/80SEC.

BUILDUP TIME: 4800000 210SEC.

TOTAL TIME: 1522.0 SEC.

TOTAL PULSES: 11959.

ANALYSER CYCLES: 7141.

TOTAL COUNTS ALL CHANNELS: 141948

0										
198	294	435	1097	1462	2207	2572	2785	2952	3049	
2889	2932	2840	2494	2348	2300	2120	1910	1825	1641	
1541	1321	1261	1122	1019	911	879	810	730	654	
580	599	539	475	443	392	372	345	363	358	
345	322	295	260	275	243	243	243	241	241	
224	211	225	224	202	210	196	192	230	212	
214	189	185	182	187	229	175	196	177	193	
197	180	199	178	172	172	199	189	180	184	
168	167	180	186	177	206	165	173	202	191	
191	189	149	165	192	177	193	171	195	186	
100										
174	197	193	176	159	174	201	196	174	161	
167	183	183	170	193	175	168	157	170	184	
167	163	180	175	184	183	196	187	175	186	
207	188	167	193	162	161	180	195	176	168	
184	195	185	164	189	179	210	184	182	185	
195	172	173	178	195	158	147	171	175	186	
187	173	166	160	183	186	163	173	195	184	
196	171	187	195	192	186	207	196	167	206	
183	166	165	180	185	192	189	194	193	175	
179	176	176	181	195	168	177	183	191	163	
200										
176	178	177	175	164	192	180	181	162	177	
170	174	179	200	164	193	179	173	171	170	
180	184	183	193	189	179	168	171	183	160	
173	184	177	194	170	182	189	194	173	184	
167	174	173	170	175	180	183	169	163	175	
185	143	170	198	181	190	188	181	191	172	
175	195	172	163	200	171	187	182	187	189	
189	162	177	176	190	181	175	188	187	186	
214	179	176	157	199	156	167	179	189	184	
175	205	155	167	183	197	170	185	179	161	
300										
183	191	185	179	182	185	167	184	179	177	
192	189	174	201	202	190	186	170	162	198	
172	197	194	186	187	188	198	195	163	185	
183	187	184	184	183	181	193	185	179	185	
182	175	180	181	177	174	193	173	171	185	
166	174	164	188	161	183	198	200	169	174	
181	163	194	175	185	195	140	176	163	200	
186	170	190	187	166	183	178	178	161	189	
183	183	195	185	188	181	213	165	189	190	
188	191	171	176	176	175	166	197	187	188	
400										
179	185	167	183	214	175	192	156	177	168	
182	180	175	184	194	189	182	186	172	188	
189	156	177	187	187	181	195	177	185	179	
163	191	183	193	196	189	194	200	174	174	
167	196	182	165	181	182	203	191	188	193	
170	168	179	184	175	194	183	180	160	194	
194	172	175	181	169	174	184	156	156	187	
176	175	198	186	173	181	205	179	173	182	
179	177	177	169	165	175	203	169	189	177	
158	176	203	156	173	172	174	178	206	175	
500										
194	171	175	174	175	183	178	182	163	188	
169	176									

POST PULSE COUNT RATE: SCALER #1 22524, 22696, 22717/80SEC.
SCALER #2 322, 348, 344/80SEC.
SCALER #3 23645, 24009, 23698/80SEC.

BACKGROUND

COUNTING TIME: 317.5 SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 90304.
SCALER #2 1285.
SCALER #3 95177.

ANALYSER CYCLES: 7284.

TOTAL COUNTS ALL CHANNELS: 84988

HNF-29648, Rev. 1

DATE: 4-10-75.
PULSE NUMBER: 41075-1.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

6.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-IA FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN SOB NO. 083.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 4 54 FUEL COLUMNS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 48226.44327/80SEC.
SCALER #2 OFF.
SCALER #3 OFF.

BUILDUP TIME: 120SEC.

TOTAL TIME: 522.2 SEC.

TOTAL PULSES: 11266.

ANALYSER CYCLES: 8672.

TOTAL COUNTS ALL CHANNELS: 538056

470	19378	22235	21500	19909	18471	17025	15951	14691	13675
12422	11672	10774	10215	9249	8705	7766	7143	6728	6003
5553	5090	4716	4311	3941	3584	3347	3040	2812	2608
2455	2236	2049	1916	1729	1660	1563	1389	1411	1268
1191	1080	1013	1047	913	900	951	831	828	791
770	735	675	649	662	626	618	607	593	588
561	558	546	570	571	553	527	548	546	531
518	510	457	513	516	481	465	496	503	484
471	486	473	455	503	483	454	442	509	470
477	504	476	493	473	477	414	478	422	478
100									
470	455	432	505	458	426	454	450	450	422
464	456	433	445	472	444	477	431	473	448
473	459	464	463	444	435	476	461	484	416
456	472	443	465	428	453	411	455	466	485
434	441	429	437	424	456	464	494	424	477
454	417	483	435	446	438	448	500	457	469
456	468	426	439	411	478	437	466	472	455
461	414	457	447	414	427	464	456	405	441
454	469	442	467	437	434	446	475	454	429
445	432	451	448	418	463	442	449	487	409
200									
431	443	421	480	476	435	424	493	470	463
440	414	447	460	431	462	473	453	444	451
410	445	479	478	478	450	489	426	458	441
436	457	399	483	434	469	487	469	442	426
434	458	448	472	461	445	446	458	447	429
427	440	449	423	459	478	444	447	461	458
479	461	459	458	446	454	484	506	448	444
456	463	466	475	483	425	454	468	406	415
462	480	463	487	455	469	505	421	447	426
461	451	475	458	445	439	460	478	468	454
300									
429	461	481	480	498	476	438	484	456	519
503	461	493	466	479	460	515	465	491	518
411	457	500	484	487	486	461	459	459	489
451	493	469	455	474	481	430	471	456	472
444	466	422	505	485	443	479	451	464	449
438	498	453	453	482	458	449	476	487	505
483	476	497	450	461	478	485	470	490	429
495	477	449	463	473	467	498	490	481	491
464	517	472	467	485	508	522	455	462	493
516	449	464	457	497	441	474	482	448	451
400									
435	460	444	457	465	473	495	471	478	484
453	483	494	458	476	490	455	452	499	480
497	446	452	473	484	459	482	425	484	488
425	474	487	465	503	468	512	509	476	492
480	482	518	480	480	490	465	466	456	521
497	439	447	486	454	467	455	486	450	492
479	496	471	478	443	461	474	535	485	476
451	488	427	488	489	475	493	483	473	489
469	467	466	509	475	466	498	427	497	471
497	516	461	469	458	483	471	523	497	507
500									
473	525	484	465	515	461	488	484	532	450
505	463								

POST PULSE COUNT RATE: SCALER #1 48249.48327 .
SCALER #2 OFF .
SCALER #3 OFF .

BACKGROUND

COUNTING TIME: 400SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 243196 .
SCALER #2: OFF .
SCALER #3 OFF .

ANALYSER CYCLES: 8639. TRIGGERS: 8640.

TOTAL COUNTS ALL CHANNELS: 215200

HNF-29648, Rev. 1

DATE: 4-10-75.
PULSE NUMBER: 41075-3.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN S08 NO. 137.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 54 FUEL COLUMNS.

MICROSEC/CHANNEL: 60

PRE-PULSE COUNT RATE: SCALER #1 48151, 48031, 47985/80SEC.
SCALER #2 477, 493, 482/80SEC.
SCALER #3 13656, 13795, 13614/80SEC.

BUILDUP TIME: 120SEC.

TOTAL TIME: 524.1 SEC.

TOTAL PULSES: 11636.

ANALYSER CYCLES: 8969.

TOTAL COUNTS ALL CHANNELS: 483996

0									
452	10715	19600	19362	17840	16424	14617	13517	12419	11414
10311	9729	8934	8165	7507	6787	6043	5592	5250	4728
4497	4120	3434	3302	3133	2836	2668	2446	2339	2120
2003	1797	1623	1518	1490	1360	1319	1264	1107	1024
992	956	869	785	808	845	839	730	730	829
670	651	657	657	601	586	578	556	564	584
533	578	505	522	563	518	494	512	441	490
477	489	516	485	487	507	469	485	496	484
468	486	467	471	510	442	483	472	459	456
520	491	461	492	475	496	514	467	473	441
100									
473	517	493	475	463	465	505	442	456	453
477	447	441	460	488	476	432	466	462	487
454	476	465	487	453	466	465	487	481	491
523	476	503	467	467	465	485	493	481	453
478	468	419	498	422	444	484	476	468	451
486	450	472	452	474	434	443	485	459	453
490	460	489	472	454	467	469	500	437	484
462	462	509	472	493	438	507	424	485	431
511	482	492	454	481	443	443	465	456	438
446	461	470	462	459	505	463	451	486	450
200									
471	481	499	451	481	448	498	457	457	464
520	445	506	460	486	487	486	471	455	491
428	465	466	473	469	497	430	467	413	470
499	489	451	458	469	468	483	471	460	
500	467	439	448	482	488	404	454	463	463
471	480	491	437	448	487	515	475	508	491
469	507	486	504	453	478	471	450	486	508
521	465	479	470	473	476	467	457	478	449
485	488	452	452	459	487	467	442	499	444
463	469	471	505	480	495	461	447	469	479
300									
504	450	450	466	454	480	513	449	444	487
477	447	498	486	495	485	499	467	446	464
455	490	481	462	454	460	475	463	452	465
428	483	517	481	461	462	454	460	461	468
484	490	471	462	502	466	518	496	477	451
504	469	414	456	444	468	518	454	474	509
472	434	461	473	519	477	474	463	481	441
468	503	458	507	492	500	484	494	497	476
456	456	459	472	476	460	459	503	488	475
473	461	487	448	477	491	466	483	440	463
400									
457	474	471	447	480	470	472	472	478	480
475	458	460	519	499	490	474	529	458	458
470	495	454	474	451	470	466	474	475	448
454	478	481	490	485	482	454	494	491	503
462	462	458	448	427	488	469	452	469	492
468	475	479	470	485	495	505	456	492	425
456	482	484	475	499	454	514	461	497	471
451	488	465	505	470	474	542	478	448	497
445	506	467	491	446	439	508	500	466	508
485	468	471	503	482	485	460	460	465	486
500									
494	462	500	485	503	490	462	495	497	482
492	483								

POST PULSE COUNT RATE: SCALER #1 48385, 48102, 48213/80SEC.
SCALER #2 461, 514, 515/80SEC.
SCALER #3 13523, 13731, 13550/80SEC.

BA+[[[[[-----

COUNTING TIME: 400SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 240966.
SCALER #2: 2534.
SCALER #3 68489.

ANALYSER CYCLES: 8890. TRIGGERS: 8890.

TOTAL COUNTS ALL CHANNELS: 219802

0

HNF-29648, Rev. 1

DATE: 4-10-75.
PULSE NUMBER: 41075-4.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
ELSE NEUTRON SOURCE: KAMAN 137.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 5A FUEL COLUMNS: CENTER POSITION CONTAINS NO FUEL.
FISSION CHAMBER LAST #241 IN CENTER. BOTTOM: 13" FROM FLOOR OF BASIN.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 358,356,329/80SEC. ALL
SCALER #2 3,3,4/80SEC. SOURCES REMOVED
SCALER #3 187,206,189/80SEC

BUILDUP TIME: 120 SEC.

TOTAL TIME: 520.4 SEC.

TOTAL PULSES: 11748.

ANALYSER CYCLES: 9037.

TOTAL COUNTS ALL CHANNELS: 197658

0	7798	15920	15927	14297	12716	11505	10348	9548	8549
7697	6859	6324	5836	5248	4694	4090	3664	3218	3008
2785	2514	2218	1953	1772	1566	1386	1263	1123	952
864	837	736	683	581	520	492	431	370	375
340	274	229	200	175	152	135	116	106	142
109	118	93	100	92	71	70	68	74	63
62	60	66	42	52	43	44	46	42	37
41	39	43	36	44	35	32	25	33	29
34	30	34	28	26	27	34	31	36	32
27	35	33	28	32	25	26	39	30	34
100	26	36	34	28	31	36	31	31	25
31	30	22	34	28	27	30	34	34	35
29	24	31	27	29	29	40	27	33	28
24	35	29	34	34	25	31	31	27	25
31	30	31	28	28	24	35	28	28	23
26	25	27	26	22	33	31	36	36	34
37	34	26	28	30	28	35	37	39	38
32	28	29	32	29	30	27	28	20	22
36	26	22	25	26	34	33	47	34	25
26	29	34	40	44	28	39	30	30	29
200	31	35	31	27	30	26	20	28	36
37	29	25	33	21	32	25	35	32	29
34	26	18	18	28	25	29	21	31	25
33	37	36	36	28	22	33	41	33	24
22	21	32	29	22	34	24	26	36	26
28	24	25	40	28	24	42	41	27	34
28	26	29	21	31	34	23	36	29	35
32	27	33	42	32	34	33	34	33	34
37	41	28	34	32	36	32	32	43	28
31	17	33	31	27	25	24	33	26	27
300	30	30	30	37	24	30	38	25	21
37	36	22	22	30	36	22	30	31	24
34	37	32	30	27	42	38	26	34	21
33	22	24	24	39	33	30	30	27	33
40	37	27	21	29	33	25	38	28	37
22	29	20	27	33	40	25	25	36	26
25	30	31	28	31	30	20	31	28	30
19	31	34	32	38	21	38	34	39	39
31	24	37	28	27	31	26	35	20	28
31	15	35	30	37	30	36	28	27	31
400	24	22	22	36	32	26	26	29	41
26	43	24	30	19	36	27	41	35	32
40	34	29	25	25	32	28	38	35	29
29	28	34	23	27	26	24	30	31	39
35	29	25	19	22	40	36	32	35	34
25	41	38	34	37	30	29	29	24	22
37	43	28	25	40	27	36	21	25	35
20	24	33	26	37	35	32	31	29	43
40	22	30	20	23	35	27	25	32	24
37	32	33	28	33	38	32	28	28	30
500	21	26	33	30	35	22	23	29	26
21	21	26	33	30	35	22	23	29	26
33	26								

POST PULSE COUNT RATE: SCALER #1 375,312,324/80SEC .
SCALER #2 2,2,2/80SEC
SCALER #3 201,186,202/80SEC .

BACKGROUND

COUNTING TIME: 400SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 1566 .
SCALER #2: 13 .
SCALER #3 964 .

ANALYSER CYCLES: 9036.

TRIGGERS 9037 .

TOTAL COUNTS ALL CHANNELS: 1438

HNF-29648, Rev. 1

DATE: 4-10-79.
PULSE NUMBER: 41075-5.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.

WD 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN 137.

DETECTOR: FISSION CHAMBER LASL #241.
DETECTOR LOCATION: CENTER, 13" ABOVE BASIN FLOOR.

LOADING: 54 FUEL COLUMNS. (CENTER FUEL TUBE REMOVED)

MICROSEC/CHANNEL: 50

PRE-PULSE COUNT RATE: SCALER #1 357,364,395/80SEC. ALL SOURCES
SCALER #2 1,1,1/80SEC. REMOVED
SCALER #3 235,225,260/80SEC.

BUILDUP TIME: 120 SEC.

TOTAL TIME: 527.5 SEC.

TOTAL PULSES: 11269.

ANALYSER CYCLES: 8709.

TOTAL COUNTS ALL CHANNELS: 171189

POST PULSE

171189

0										
35	2012	6935	9857	11037	11044	10703	10129	9215	8456	
7349	6514	6243	5539	4860	4444	3975	3486	3091	2803	
2572	2250	2001	1780	1545	1412	1325	1186	1045	908	
862	741	697	605	567	481	459	411	344	322	
280	241	234	209	224	168	176	148	141	129	
129	114	100	110	85	83	76	77	77	58	
60	72	56	52	62	59	46	59	59	60	
50	49	46	39	49	44	35	52	42	47	
37	42	42	47	37	43	50	43	42	41	
30	39	37	44	46	44	45	43	46	49	
100										
45	40	46	42	55	45	38	33	34	52	
35	50	35	36	52	40	52	47	42	32	
35	55	41	45	39	42	37	40	31	34	
39	41	49	39	42	45	39	36	39	31	
34	41	36	47	42	42	38	43	38	46	
56	27	46	46	42	51	44	41	39	43	
44	45	39	46	42	27	45	44	33	46	
38	41	55	32	37	44	44	41	32	36	
47	59	35	35	40	41	29	37	44	37	
40	44	48	49	45	38	55	40	45	46	
200										
37	46	41	38	45	42	44	39	33	50	
35	39	37	31	45	42	35	44	48	51	
44	44	46	45	43	35	37	46	45	44	
32	39	44	42	41	32	30	51	47	33	
40	50	50	46	38	35	43	38	43	45	
30	37	36	44	31	38	34	40	45	46	
51	43	50	42	41	41	46	47	46	35	
36	46	31	32	51	46	50	42	41	39	
37	40	34	42	42	42	38	40	38	44	
33	39	33	36	43	49	46	54	41	35	
300										
45	37	40	30	55	37	36	37	47	33	
42	37	41	40	30	29	47	45	39	45	
34	34	45	36	43	39	48	44	38	34	
42	32	49	34	51	35	45	43	50	49	
36	41	45	39	47	43	63	34	46	36	
43	42	36	33	44	32	34	41	33	40	
47	43	39	29	52	36	34	34	30	36	
39	27	44	39	35	39	35	38	35	45	
46	29	42	40	43	39	39	33	39	45	
37	39	39	45	40	38	45	33	39	51	
400										
45	47	54	33	43	44	35	43	26	40	
44	40	45	28	34	43	40	45	40	46	
56	50	43	43	42	37	46	50	53	46	
46	29	46	39	40	48	37	36	40	38	
55	40	41	47	49	40	59	41	45	42	
35	38	46	39	38	44	45	49	43	42	
42	30	43	42	42	36	46	49	36	44	
45	51	45	38	30	33	45	33	40	35	
37	47	35	46	33	45	49	41	55	55	
44	34	42	37	44	30	37	41	43	35	
500										
35	35	38	46	44	32	37	44	35	56	
42	48									

COUNT RATE: SCALER #1 391,436,364/80SEC.
SCALER #2 3,3,2,7/80SEC
SCALER #3 255,264,256/80SEC.

BACKGROUND

COUNTING TIME: 400SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 1929.
SCALER #2: 13.
SCALER #3: 1257.

ANALYSER CYCLES: 8545.

TRIGGERS: 8545.

TOTAL COUNTS ALL CHANNELS: 1693

HNF-29648, Rev. 1

DATE: 4-11-75.
PULSE NUMBER: 41175-1.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.
ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PWS TRIGGER.
ULSE NEUTRON SOURCE: KAMAN 137.

DETECTOR: BORON PROPORTIONAL COUNTER.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 48 TUBES (ALL NOS. 10 ARE OUT. CENTER TUBES OUT ALSO..

MICROSEC/CHANNEL: 60

PRE-PULSE COUNT RATE: SCALER #1 213.192.172/80SEC. ALL SOURCES REMOVED.
SCALER #2 P.O. 0/80SEC
SCALER #3 195.189.153/80SEC

BUILDUP TIME: 1205SEC.

TOTAL TIME: 521.6 SEC.

TOTAL PULSES: 11606.

ANALYSER CYCLES: 8932.

TOTAL COUNTS ALL CHANNELS: 132735

0	7320	14309	13888	12142	10584	9114	7984	6794	5822
10	4455	3754	3310	2843	2463	2185	1949	1751	1503
20	1137	972	858	750	657	518	470	404	301
30	269	254	209	169	152	150	112	117	99
40	68	81	64	67	50	56	46	45	32
50	19	27	18	13	21	23	23	19	14
60	15	18	17	19	16	10	13	4	13
70	12	11	13	16	18	10	10	10	11
80	14	15	9	11	14	4	22	7	14
90	14	10	10	9	16	11	10	11	15
100	13	10	7	4	8	11	12	12	7
110	11	10	13	11	14	11	11	16	13
120	9	12	14	11	18	10	10	12	15
130	16	9	13	8	9	10	9	17	10
140	14	11	17	9	8	10	10	7	9
150	16	11	10	9	7	10	9	18	15
160	11	15	8	12	19	10	10	5	15
170	8	9	6	7	7	16	13	13	16
180	10	10	11	14	11	8	11	4	12
190	13	9	10	9	7	13	14	13	4
200	13	10	15	9	7	9	15	5	10
210	17	15	12	13	7	9	16	7	17
220	10	14	11	14	9	11	14	12	12
230	18	12	8	14	9	13	10	9	11
240	10	12	7	13	11	13	13	11	11
250	19	11	11	10	11	14	14	6	10
260	11	13	8	14	9	12	10	5	9
270	10	15	12	17	11	9	15	11	16
280	27	10	8	18	9	13	14	12	12
290	6	12	14	11	7	16	8	11	14
300	4	7	13	8	17	10	9	15	12
310	11	12	15	9	9	15	14	8	16
320	12	13	9	12	9	14	12	17	6
330	16	9	12	15	10	13	19	12	14
340	11	13	17	11	15	11	14	9	9
350	10	10	11	12	7	16	8	16	11
360	13	14	7	11	13	7	7	12	8
370	4	7	22	11	10	10	7	15	12
380	12	19	15	10	13	11	7	21	15
390	13	7	9	8	15	14	10	19	12
400	10	17	15	16	9	11	16	12	17
410	9	10	16	13	17	13	16	18	10
420	19	15	11	12	14	17	19	16	14
430	15	11	12	6	6	15	17	8	12
440	12	12	5	16	9	10	10	15	9
450	11	8	12	10	18	9	10	8	15
460	17	14	18	23	13	16	13	21	11
470	10	6	16	11	21	11	16	15	16
480	13	12	9	14	9	13	22	10	13
490	12	9	12	13	13	15	14	10	10
500	7	10	11	8	9	11	12	15	13
510	20	5							

POST PULSE COUNT RATE: SCALER #1 174.187.169/80SEC.
SCALER #2 4.2.2/80SEC
SCALER #3 185.170.180/80SEC

BACKGROUND

COUNTING TIME: 400SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 923.
SCALER #2: 4.
SCALER #3 773.

ANALYSIS CYCLES: 8910.

TRIGGERS 8911.

TOTAL COUNTS ALL CHANNELS: 848

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DATE: 4-11-75.
 PULSE NUMBER: 41175-2.
 PULSED NEUTRON SOURCE MEASUREMENTS - W REACTOR FUEL STORAGE BASIN.
 3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.
 ND. 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PNS TRIGGER.
 PULSE NEUTRON SOURCE: KAMAN 137.
 DETECTOR: BORON PROPORTIONAL COUNTER.
 DETECTOR LOCATION: 3 O'CLOCK.
 LOADING: 42. (CENTER FUEL TUBE REMOVED)
 MICROSEC/CHANNEL: 50
 PRE-PULSE COUNT RATE: SCALER #1 120.105.134/80SEC ALL SOURCES
 SCALER #2 0.1.0/80SEC. OUT
 SCALER #3 146.152.125/80SEC
 BUILDUP TIME: 120 SEC.
 TOTAL TIME: 520.3 SEC.

TOTAL PULSES: 10480.

ANALYSER CYCLES: 6021.

TOTAL COUNTS ALL CHANNELS: 56168

0	3577	11700	11111	9316	7794	6418	5291	4256	3422
9	2541	1998	1799	1471	1252	1043	865	744	665
2979	438	396	309	303	232	191	172	147	115
543	108	71	61	59	57	43	40	25	37
114	24	17	16	11	8	15	13	14	9
33	2	5	10	3	4	4	5	4	1
10	3	1	3	2	4	6	6	2	3
3	3	3	4	4	4	4	5	2	7
2	10	4	11	2	7	6	5	2	3
6	0	2	5	3	9	6	2	1	3
0	1	3	3	3	1	2	6	5	2
100	3	3	3	4	11	5	3	3	3
4	6	5	8	0	1	3	4	3	6
3	5	6	1	6	8	9	5	2	5
5	5	4	5	4	6	4	6	1	8
2	3	3	2	4	2	7	4	4	3
3	3	3	4	4	4	3	5	6	3
3	4	5	6	4	7	6	6	4	10
4	4	4	10	4	4	5	4	4	7
200	4	6	3	7	3	7	4	2	6
4	4	3	4	3	7	5	0	7	5
6	4	4	3	3	1	5	5	7	4
2	2	5	4	5	6	3	5	6	7
7	4	6	8	6	1	2	2	4	4
5	9	3	4	4	5	0	7	2	6
5	4	4	7	3	5	3	5	3	7
2	8	4	3	4	4	7	2	8	6
5	5	4	2	5	3	4	5	4	4
4	5	2	6	7	5	3	3	6	7
300	5	4	6	4	3	5	3	5	1
5	2	4	2	4	4	1	5	3	4
5	3	7	3	5	5	4	2	4	2
1	1	4	3	6	7	6	3	1	5
4	7	6	3	6	3	4	1	3	1
0	5	2	6	8	6	7	6	2	12
0	3	6	2	2	3	9	3	2	4
3	4	5	6	9	3	3	5	5	6
400	3	3	6	3	3	3	3	5	3
3	10	11	6	3	6	1	6	1	4
5	2	13	4	5	4	6	6	3	4
6	8	2	4	6	3	4	3	2	3
6	5	7	3	4	2	5	6	1	4
1	4	10	3	4	4	6	6	3	2
5	4	3	6	4	5	4	1	3	7
4	2	7	3	7	5	6	5	4	4
5	6	5	8	6	6	1	7	3	4
500	3	7	4	5	4	7	4	4	1
3	8	5	5	4	7	4	7	4	1

POST PULSE COUNT RATE: SCALER #1 102.101.91/80SEC .
 SCALER #2 0.1.1/80SEC
 SCALER #3 117.141.114/80SEC .

BACKGROUND

COUNTING TIME: 400.0SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 533 .
 SCALER #2: 2 .
 SCALER #3 671 .

ANALYSER CYCLES: 6064.

TRIGGERS 8065 .

TOTAL COUNTS ALL CHANNELS: 433

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DATE: 4-11-75.
 PULSE NUMBER: 41175-3.
 PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.
 3.2 INCH HEXAGONAL LATTICE OF UNIRRADIATED MARK-1A FUEL ELEMENTS.
 ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
 PULSE NEUTRON SOURCE: KAMAN 137.

DETECTOR: BORON PROPORTIONAL COUNTER.
 DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 36. (CENTER FUEL TUBE REMOVED)

MICROSEC/CHANNEL: 60

PRE-PULSE COUNT RATE: SCALER #1 74.55.61/80SEC. ALL
 SCALER #2 3.1.0/80SEC SOURCES
 SCALER #3 136.103.125/80SEC . REMOVED

BUILDUP TIME: 2205SEC.

TOTAL TIME: 620.25EC SEC.

TOTAL PULSES: 14231.

ANALYSER CYCLES: 11478.

TOTAL COUNTS ALL CHANNELS: 100857

0	6112	15518	14480	12078	9880	7810	6284	4936	3788
3189	2511	1944	1595	1326	1049	922	677	544	471
409	301	255	229	174	142	125	107	83	75
41	45	50	35	32	33	21	14	14	13
12	7	13	4	11	5	5	0	7	4
6	0	5	4	1	3	4	4	6	3
7	2	3	4	5	3	4	1	3	2
2	4	3	2	4	7	1	4	2	2
3	3	3	1	6	4	1	3	3	1
0	3	2	3	1	3	2	4	6	2
100	3	4	7	7	2	4	2	3	4
1	2	3	2	7	5	1	0	2	2
4	5	5	1	3	2	2	3	3	4
1	3	6	5	2	5	3	2	1	1
3	2	2	4	2	1	4	2	2	2
2	2	4	0	7	2	4	5	1	2
0	2	6	1	2	3	4	4	4	1
5	3	1	4	1	4	6	2	3	2
1	4	2	3	1	3	4	3	2	5
200	4	2	3	1	3	3	5	5	5
5	2	2	3	5	3	6	2	4	3
1	1	2	6	3	2	4	4	0	4
4	2	4	5	2	3	5	0	4	1
2	2	1	4	2	1	3	5	3	5
6	1	0	2	0	4	2	2	2	3
1	5	4	1	3	6	3	2	2	4
5	2	3	2	3	0	2	6	2	2
3	2	2	5	4	4	4	3	6	5
4	2	3	5	2	1	3	6	7	4
300	4	2	7	5	1	1	1	0	5
4	5	2	0	4	3	0	3	2	3
0	1	3	4	2	3	3	2	4	6
2	2	3	2	2	3	5	1	4	4
5	3	2	2	3	3	2	0	6	6
3	8	2	2	3	0	4	3	2	6
1	3	2	3	2	4	2	3	1	2
3	3	4	0	2	6	1	2	2	3
1	4	4	0	3	3	3	4	3	2
2	6	3	1	3	2	2	1	3	2
400	7	8	3	4	4	0	5	3	0
5	2	2	1	3	4	3	2	2	6
2	6	2	5	4	2	3	2	3	3
3	1	4	3	0	5	3	2	1	4
1	3	2	4	3	3	4	2	3	4
5	3	6	7	4	2	2	3	3	1
3	2	3	2	2	5	3	3	7	0
1	6	1	3	3	2	2	2	0	1
3	3	6	7	3	1	3	3	1	4
7	4	1	2	5	3	4	3	6	3
500	5	2	5	6	2	2	2	4	3
2	3	2	5	6	2	2	2	4	3
7	3	2	5	6	2	2	2	4	3

POST PULSE COUNT RATE: SCALER #1 66.72.61/80SEC .
 SCALER #2 0/80SEC
 SCALER #3 468.68 106.102.113/80SEC .

BACKGROUND

 COUNTING TIME: 499.85EC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 431 .
 SCALER #2: 2 .
 SCALER #3 647 .

ANALYSER CYCLES: 11464.

TRIGGERS: 11485 .

TOTAL COUNTS ALL CHANNELS: 400

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DATE: 4-16-78.
 PULSE NUMBER: 41675-1.
 PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.
 3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.
 ND 2400 ANALYSER DELAYED 65-NSEC 20 MICROSECONDS FOLLOWING PMS TRIGGER.
 PULSE NEUTRON SOURCE: KAMAN 808 NO. 137.
 DETECTOR: FISSION CHAMBER L&S 367.
 DETECTOR LOCATION: 8 O'CLOCK.
 LOADING: 67 FUEL COLUMNS.
 MICROSEC/CHANNEL: 80
 PRE-PULSE COUNT RATE: SCALER #1 1262.1177.1209/80SEC. EXTERNAL
 SCALER #2 824.532.486/80SEC. SOURCE
 SCALER #3 6972.7032.7191/80SEC. IN.

TOTAL TIME: 662.2 SEC .BUILDUP TIME: 120 SEC.

TOTAL PULSES: 14507.

ANALYSER CYCLES: 11847.

TOTAL COUNTS ALL CHANNELS: 10881

0									
11	201	540	529	504	495	440	436	340	304
323	233	229	183	156	154	145	103	103	88
56	66	67	60	61	42	36	29	35	37
32	19	22	16	22	18	18	19	10	23
10	9	17	17	11	18	4	12	15	15
10	10	7	13	12	11	6	7	13	9
11	11	13	14	15	9	7	6	11	10
11	9	12	8	9	7	6	10	16	10
3	8	11	5	11	6	12	12	7	15
9	11	9	4	6	14	10	8	9	3
100	8	17	7	11	8	6	9	8	12
11	8	9	12	5	6	14	12	10	19
11	11	2	15	6	10	10	10	7	10
7	10	8	5	8	14	7	10	5	12
6	10	10	15	12	7	8	12	9	13
15	8	13	9	10	16	6	12	9	13
10	11	9	11	8	9	10	12	12	6
7	7	7	5	16	10	9	13	9	12
5	11	5	8	14	9	9	9	11	12
9	9	5	9	9	3	10	5	9	4
200	10	11	15	9	12	13	8	9	7
7	19	12	10	6	11	11	6	9	13
14	8	7	11	7	10	8	13	11	19
18	6	11	14	7	8	5	10	8	14
12	12	3	12	13	11	7	11	13	6
16	8	12	11	6	6	9	8	10	9
13	12	10	11	9	14	8	8	12	14
7	8	13	9	10	7	9	9	12	19
5	8	8	12	13	15	10	13	6	3
9	8	15	11	5	11	9	14	15	19
300	11	15	12	8	6	7	13	12	10
11	8	8	11	4	12	14	10	16	12
16	11	6	4	12	9	10	13	10	12
9	15	10	5	8	16	11	5	10	10
5	7	12	5	11	10	13	14	7	11
9	7	5	11	11	11	11	4	6	10
13	10	6	9	13	12	10	13	9	7
12	7	10	12	13	4	7	12	13	9
9	11	7	11	6	11	12	9	11	9
7	11	8	10	9	14	8	11	7	8
400	5	10	8	17	7	8	19	5	8
5	10	15	6	9	11	3	4	13	8
14	8	8	9	10	8	9	11	11	9
6	7	7	6	7	9	14	9	10	7
14	5	9	9	4	9	8	14	11	14
10	10	10	12	9	10	12	10	11	17
8	10	17	8	6	10	11	18	10	16
9	4	11	9	6	10	9	6	9	11
10	12	16	12	10	12	9	9	12	8
14	8	7	8	4	11	9	9	10	15
500	4	7	10	13	5	9	13	8	12
10	8								

POST PULSE COUNT RATE: SCALER #1 803.516.783.791/80SEC. EXTERNAL
 SCALER #2 395.436.783.791/80SEC. SOURCE
 SCALER #3 5181.5167.8093.5162/80SEC. OUT.

BACKGROUND

COUNTING TIME: 539.7 SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 5273.
 SCALER #2: 2874.
 SCALER #3 34041.

ANALYSER CYCLES: 11845. TRIGGERS: 11846.

TOTAL COUNTS ALL CHANNELS: 4746

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DATE: 4-17-75.
 PULSE NUMBER: 41775-1.
 PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.
 3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.
 ND 2400 ANALYSER DELAYED 45-NDSE 20 MICROSECONDS FOLLOWING PMS TRIGGER.
 PULSED NEUTRON SOURCE: KAMAN 608 NO. 137.
 DETECTOR: FISSION CHAMBER LASEL 357.
 DETECTOR LOCATION: 3 O'CLOCK.
 LOADING: 73 FUEL COLUMNS.
 MICROSEC/CHANNEL: 20
 PRE-PULSE COUNT RATE: SCALER #1 1052,969,943/40SEC. CF SOURCE OUT.
 SCALER #2 423,611,633/60SEC.
 SCALER #3 7453,7298,7266/80SEC

TOTAL TIME: 710.8 SEC .WUDDUP TIME: 180 SEC.

TOTAL PULSES: 15885.

ANALYSER CYCLES: 11866.

TOTAL COUNTS ALL CHANNELS: 21522

0	427	933	1095	1101	1052	961	943	871	878
741	674	595	534	481	412	353	304	303	273
234	187	183	180	127	121	116	109	85	72
75	83	65	52	57	54	34	47	32	34
32	57	17	31	20	21	17	19	15	13
23	26	20	14	17	17	0	16	16	15
18	17	14	14	17	12	17	21	12	11
15	8	13	12	11	15	12	0	8	16
13	20	15	12	13	10	17	18	13	7
20	17	20	12	17	9	14	24	0	11
100	14	10	13	11	17	10	13	20	17
8	21	15	11	11	18	17	17	11	14
8	14	8	16	11	14	22	15	17	14
9	17	11	16	14	24	15	19	10	13
13	12	14	17	8	10	13	18	12	12
11	16	12	20	11	14	18	15	10	17
12	16	0	9	16	7	15	20	22	10
17	15	0	11	20	12	19	13	12	18
14	13	10	13	13	17	6	19	9	17
23	8	16	13	14	0	23	14	16	14
200	11	12	11	14	11	14	9	13	17
8	13	16	14	22	12	14	11	12	13
13	12	16	15	12	12	14	17	18	8
14	9	18	12	20	14	18	10	14	15
14	10	14	13	17	13	11	6	10	12
11	13	12	15	10	22	10	16	15	14
10	12	13	13	13	12	10	16	16	15
22	14	15	6	18	17	14	14	13	21
14	13	8	12	21	10	14	17	13	15
7	6	0	16	16	14	22	6	11	10
300	13	25	15	16	13	14	12	11	13
10	17	14	16	12	10	11	0	11	16
14	12	15	21	8	0	14	12	14	15
12	17	13	13	14	17	17	20	13	12
17	15	15	13	11	27	7	10	11	19
22	17	15	8	10	15	14	19	21	15
20	10	17	16	13	11	15	13	20	21
15	15	13	17	6	24	13	15	15	15
15	12	7	14	11	15	12	16	11	17
16	16	19	13	21	12	15	17	14	7
400	18	14	11	7	19	7	11	14	12
9	13	20	13	13	13	10	13	19	9
15	16	24	13	14	25	17	18	8	12
15	16	13	15	11	18	21	13	15	17
8	14	16	13	12	13	6	10	17	14
18	12	10	17	19	10	15	16	18	11
11	6	14	0	13	8	13	0	16	10
15	16	17	10	10	16	6	10	11	12
15	12	7	7	14	15	14	10	16	8
23	18	16	15	11	14	17	10	9	14
500	16	15	19	11	13	13	20	10	18
11	19								

POST PULSE COUNT RATE: SCALER #1 1075,1058,1034/40SEC .
 SCALER #2 605,572,558/60SEC .
 SCALER #3 7075,7140,7143/80SEC .

BACKGROUND

COUNTING TIME: 532.3 SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 4828 .
 SCALER #2: 3070 .
 SCALER #3 47534 .

ANALYSER CYCLES: 11902, TRIGGERS: 11903.

TOTAL COUNTS ALL CHANNELS: 6241

0

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DATE: 4-21-75.
PULSE NUMBER: 42175-1.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.
3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.
WD 2400 ANALYSER DELAYED 80 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE; KAMAN 808 NO. 137.
DETECTOR: FISSION CHAMBER LSL 387.
DETECTOR LOCATION: 3 O'CLOCK.
LOADING: 79 FUEL COLUMNS.
MICROSEC/CHANNEL: 80
PRE-PULSE COUNT RATE: SCALER #1 4P 899444899888 1731.1756/80SEC. CF
SCALER #2 88882888 906.917/80SEC. SOURCE
SCALER #3 10008,9993/80SEC. OUT.

TOTAL TIME: 570.25EC.
BUILDUP TIME: 1708EC.

TOTAL PULSES: 12862.

ANALYSER CYCLES: 8987.

TOTAL COUNTS ALL CHANNELS: 19569

[illegible]

POST PULSE COUNT RATE: SCALER #1 1729,1721,1757/60SEC .
SCALER#2 941,939,911/60SEC .
SCALER #3 9922,9870,10036/60SEC .

BACKGROUND

COUNTING TIME: 400.0SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 4668.00 8653 .
SCALER #2 4332 .
SCALER #3 50063 .

TRIGGERS: 9037.

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DATE: 4-22-75.
PULSE NUMBER: 48875-1.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSED NEUTRON SOURCE: KAMAN 608 NO. 137.

DETECTOR: FISSION CHAMBER LASL 387.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 85 FUEL COLUMNS.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 8554, 2555, 2545/80SEC. CF
SCALER #2 1487, 1514, 1517/80SEC. SOURCE
SCALER #3 12587, 12761, 12571/80SEC. OUT .

TOTAL TIME: 557.5 SEC.
BUILDUP TIME: 120 SEC.

TOTAL PULSES: 11915.

ANALYSER CYCLES: 9362.

TOTAL COUNTS ALL CHANNELS: 37560

0										
30	323	575	1117	1235	1312	1364	1422	1391	1291	
1217	1202	1122	1019	887	762	754	657	575	574	
469	456	437	375	370	298	292	264	221	216	
155	155	149	160	169	107	97	95	81	92	
✓56	79	70	65	66	52	66	50	54	50	
43	43	49	42	46	40	36	✓30	43	39	
39	39	31	30	27	35	27	29	40	25	
24	24	32	22	✓24	34	30	39	31	32	
31	27	26	32	23	24	32	23	22	32	
30	✓27	29	26	41	31	32	30	29	22	
100										
26	15	19	33	29	37	28	32	26	38	
32	31	34	17	29	28	35	15	34	34	
22	25	26	17	26	29	30	33	35	37	
26	26	24	24	29	31	30	27	20	28	
25	26	✓23	23	21	29	28	23	23	25	
21	35	26	25	31	29	30	21	30	33	
33	36	28	23	33	31	21	17	31	26	
41	27	42	32	32	35	29	14	26	30	
26	29	15	27	14	32	32	25	25	27	
25	32	30	✓19	25	31	33	22	26	20	
200										
35	30	28	27	27	35	20	26	24	30	
✓31	28	20	32	21	32	25	34	34	33	
30	34	30	37	21	35	14	✓22	19	28	
20	31	36	33	33	23	34	24	30	26	
22	30	27	40	✓30	20	25	33	25	26	
32	35	36	24	30	26	23	26	28	26	
29	✓40	32	27	33	35	27	31	36	20	
27	27	33	28	30	22	28	30	✓24	42	
23	32	27	29	33	26	25	34	28	26	
32	38	19	22	24	✓27	34	21	18	23	
300										
25	19	29	30	27	26	21	25	26	29	
40	20	✓26	27	20	28	39	23	23	37	
26	28	35	27	28	28	25	35	28	✓30	
31	35	33	23	29	32	29	23	30	24	
24	27	27	22	24	23	✓27	23	27	27	
36	26	27	32	21	29	33	30	19	32	
23	22	23	✓28	23	36	34	26	31	18	
28	28	20	32	35	28	25	31	21	26	
✓27	38	23	35	27	33	30	26	24	25	
21	24	26	37	19	26	27	✓20	27	33	
400										
37	39	24	24	29	32	42	26	29	38	
20	30	17	29	✓18	23	26	40	24	29	
28	27	21	21	22	32	31	25	24	27	
27	✓34	31	19	22	21	23	32	38	25	
23	26	35	23	28	30	26	25	✓30	25	
28	33	24	32	36	22	35	31	27	37	
30	30	34	24	30	✓30	32	28	28	33	
30	21	36	30	26	20	22	30	22	14	
27	23	✓20	34	31	27	32	36	29	33	
30	19	25	29	36	30	23	31	29	✓36	
500										
30	31	32	28	27	19	27	29	26	31	
40	32									

POST PULSE COUNT RATE: SCALER #1 2581, 2573, 2493/80SEC .
SCALER #2 1548, 1443, 1537/80SEC .
SCALER #3 12774, 12693, 12680/80SEC .

BACKGROUND

COUNTING TIME: 436.9 SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 13718 .
SCALER #2 8216 .
SCALER #3 69121 .

TRIGGERS: 9360.

ANALYSER CYCLES: 9359.

TOTAL COUNTS ALL CHANNELS: 12026

0

HNF-29648, Rev. 1

DATE: 4-22-75.
 PULSE NUMBER: 42275-2.
 PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.
 3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.
 ND 2400 ANALYSEP DELAYED 20 MICROSECONDS FOLLOWING PNS TRIGGER.
 PULSED NEUTRON SOURCE: KAHAN 808 NO. 137.

DETECTOR: FISSION CHAMBER LAST 387.
 DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 88 FUEL COLUMNS.

MICROSEC/CHANNEL: 60

PRE-PULSE COUNT RATE: SCALER #1 3877,3880,3874/80SEC CF
 SCALER #2 2258,2261,2256/80SEC SOURCE
 SCALER #3 899** OUT

TOTAL TIME: 320.1 SEC.
 BUILDUP TIME: 120 SEC.

TOTAL PULSES: 10578.

ANALYSER CYCLES: 8126.

TOTAL COUNTS ALL CHANNELS: 47294

0	32	324	804	1056	1264	1462	1616	1531	1618	1507
1499	1403	1327	1215	1109	1053	890	828	791	732	
600	608	543	506	437	407	379	362	298	250	
235	236	256	224	187	166	171	148	148	141	
112	112	103	109	90	80	88	79	86	81	
62	67	65	54	42	42	59	58	57	48	
45	46	38	48	52	38	43	40	46	47	
39	39	43	49	33	42	41	31	41	42	
34	47	57	34	36	33	29	33	38	45	
48	34	41	37	40	34	31	36	29	42	
100										
36	30	32	34	48	40	32	33	29	39	
43	34	40	37	40	38	31	37	48	49	
38	34	42	44	35	35	45	42	36	46	
33	43	51	34	34	35	41	42	43	30	
29	38	43	27	43	38	38	43	37	38	
47	38	33	30	47	41	36	36	29	41	
34	46	33	37	39	37	27	36	40	29	
33	24	41	32	29	40	32	35	41	31	
32	46	42	39	46	41	37	33	44	48	
38	43	41	39	28	47	34	40	32	44	
200										
44	31	37	42	33	31	46	37	28	43	
45	46	44	42	36	45	40	36	49	32	
44	36	42	40	42	47	45	33	37	28	
50	46	34	44	40	43	39	39	45	42	
35	42	45	33	34	37	40	33	35	33	
33	43	36	31	36	38	38	35	33	45	
40	31	38	42	41	39	41	38	32	33	
33	34	40	38	39	46	29	33	37	43	
27	28	34	28	45	44	19	34	38	43	
29	42	42	33	42	30	45	38	28	35	
300										
51	36	39	38	39	42	42	35	40	42	
41	38	44	31	26	31	39	36	29	43	
44	35	34	41	34	44	43	45	40	44	
39	47	41	33	30	34	42	44	43	40	
38	37	44	47	35	43	45	36	39	37	
32	37	33	38	30	31	44	32	45	41	
39	47	32	34	37	27	42	38	37	39	
47	50	43	38	32	47	46	30	38	47	
40	38	41	39	42	47	33	33	24	35	
31	44	40	41	39	45	49	34	49	36	
400										
40	44	32	32	43	39	28	37	36	37	
45	33	40	34	44	38	43	31	44	39	
36	34	36	26	39	44	43	40	30	51	
46	31	43	52	55	41	41	39	35	37	
36	44	25	38	36	31	36	30	35	35	
36	41	35	43	35	52	30	36	37	47	
37	43	41	35	33	42	36	37	38	49	
32	31	23	43	40	35	38	28	30	32	
34	36	25	43	36	34	38	38	45	38	
40	38	27	36	38	31	41	37	44	42	
500										
35	43	24	48	37	47	40	33	34	39	
40	47									

POST PULSE COUNT RATE: SCALER #1 3815,3787,3874/80SEC .
 SCALER #2 2293,2249,2282/80SEC .
 SCALER #3 OUT .

BACKGROUND

COUNTING TIME: 400.0SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 19407 .
 SCALER #2 11142 .
 SCALER #3 OUT .

RIGGERS: 8148.

ANALYSER CYCLES: 8147.

TOTAL COUNTS ALL CHANNELS: 16229

ROUGH ESTIMATE OF REACTOR

TOTAL COUNTS ALL CHANNELS: 16229

HNF-29648, Rev. 1

DATE: 4-23-75.
 PULSE NUMBER: 42375-1.
 PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.
 3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.
 ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
 PULSE NEUTRON SOURCE: KAMAN 808 NO. 137.
 DETECTOR: FISSION CHAMBER LASTL 241.
 DETECTOR LOCATION: CENTER OF ASSEMBLY.
 LOADING: 90 FUEL COLUMNS (CENTER FUEL COLUMN OUT).
 MICROSEC/CHANNEL: 80
 PRE-PULSE COUNT RATE: SCALER #1 50316, 50908, 50280/80SEC . CF SOURCE OUT
 SCALER #2 1932, 1956, 1942/80SEC.
 SCALER #3 12, 16, 17/80SEC .

TOTAL TIME: 520.25SEC..
 BUILDUP TIME: 120SEC..

TOTAL PULSES: 10167.

ANALYSER CYCLES: 7821.

TOTAL COUNTS ALL CHANNELS: 513214

476	2720	5956	13965	16661	17863	18269	17641	16837	15687
14715	13385	12283	11174	10183	9391	8354	7682	6987	6492
5777	5353	4798	4484	3949	3587	3291	3032	2762	2566
2382	2095	2021	1781	1694	1520	1461	1308	1273	1162
1025	988	954	918	903	870	798	784	800	682
744	653	609	589	606	592	522	539	619	561
547	563	522	480	535	491	518	509	499	484
463	473	465	485	450	435	499	491	511	474
499	487	493	466	503	472	466	436	489	450
460	479	463	453	467	486	498	456	454	450
100									
467	453	480	464	483	460	459	485	482	499
455	427	473	456	464	474	463	442	418	466
457	461	443	482	487	437	441	475	478	452
461	454	466	446	443	457	498	435	486	470
501	514	469	454	475	437	482	482	422	467
442	437	495	484	468	491	473	451	437	483
478	460	445	463	459	505	466	483	501	490
457	452	442	487	503	465	457	469	470	464
477	469	497	453	433	438	436	469	435	463
467	467	502	455	492	469	420	460	471	458
200									
450	452	458	483	462	487	492	456	472	444
441	449	462	498	445	458	459	482	473	470
494	508	499	510	469	454	511	491	468	479
460	499	449	478	503	480	448	488	472	489
461	497	465	521	480	448	442	442	439	460
446	447	459	433	453	472	447	493	482	458
438	469	440	482	491	459	442	488	454	487
458	446	471	460	527	456	437	474	493	505
441	457	419	466	475	512	500	428	453	512
453	437	494	469	526	445	489	451	465	449
300									
493	443	501	465	448	476	426	497	480	466
466	455	448	427	483	478	509	462	423	434
432	462	434	429	477	447	461	509	451	491
443	462	469	459	431	476	488	450	433	484
457	462	529	487	451	434	445	450	454	453
446	497	451	462	489	468	478	494	508	513
503	489	435	471	473	445	456	474	510	477
454	467	487	433	496	518	429	507	470	490
440	420	519	483	465	484	458	469	463	493
447	479	464	464	422	421	471	441	480	471
400									
456	420	470	436	458	467	456	457	458	469
470	516	510	495	471	477	484	471	469	490
497	491	479	492	419	435	462	483	484	486
489	525	473	464	471	492	477	449	447	467
471	447	437	470	453	487	488	492	480	516
459	477	474	446	491	466	466	489	480	464
459	526	458	475	483	443	490	504	475	499
447	470	458	476	495	449	483	487	503	489
492	482	475	485	486	477	470	512	483	470
480	439	476	472	480	456	468	487	483	465
500									
456	429	479	457	481	490	465	461	482	486
456	447								

POST PULSE COUNT RATE: SCALER #1 50875, 50572, 50568/80SEC .
 SCALER #2 2026, 1987, 1949/80SEC .
 SCALER #3 17, 15, 15/80SEC .

BACKGROUND

COUNTING TIME: 400.0SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 259631 .
 SCALER #2: 9712 .
 SCALER #3 61 .

TRIGGERS: 7824.

ANALYSER CYCLES: 7823.

TOTAL COUNTS ALL CHANNELS: 502295

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DATE: 4-23-75.
PULSE NUMBER: 42375-8
PULSED NEUTRON SOURCE MEASUREMENTS - IN REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAMAN 505 NO. 137.

DETECTOR: FISSION CHAMBER LAST 387.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 78 FUEL COLUMNS (CENTER COLUMN OUT)

MICROSEC/CHANNEL: 50

PRE-PULSE COUNT RATE: SCALER #1 1474.1452, 1487/80SEC.
SCALER #2 768.783, 771/80SEC. CF. SOURCE OUT
SCALER #3 29980.29044, 29461/80SEC.

TOTAL TIME: 754.2 SEC.
BUILDUP TIME: 120 SEC.

TOTAL PULSES: 15299.

ANALYSE CYCLES: 12984.

TOTAL COUNTS ALL CHANNELS: 29381

0									
18	388	877	1101	1239	1298	1255	1246	1196	1081
963	935	827	724	659	613	539	477	385	376
303	273	263	230	210	180	140	120	110	100
103	80	79	55	76	63	48	39	31	50
42	53	46	36	31	27	26	44	33	31
20	19	26	24	32	22	22	33	25	33
26	26	25	25	21	19	16	23	18	17
27	22	14	25	26	19	22	23	29	19
17	21	34	23	20	21	27	23	22	23
25	21	23	18	29	15	21	21	25	18
100									
24	12	27	20	28	21	22	23	21	14
14	19	24	21	18	16	19	17	19	25
22	11	20	15	15	26	25	21	17	17
23	20	25	17	25	18	24	27	18	19
30	24	26	21	17	16	16	27	13	25
18	28	24	22	21	14	23	29	22	28
26	17	27	24	17	22	22	22	19	18
15	29	25	20	26	19	16	16	24	28
23	18	20	15	18	20	25	25	21	18
28	19	22	27	14	27	20	24	21	28
200									
21	21	24	19	21	15	16	27	22	29
25	27	17	27	24	14	15	26	25	29
23	29	18	20	22	26	24	17	21	26
13	20	25	27	20	24	19	18	27	16
16	19	25	14	12	23	18	30	27	24
22	21	31	29	25	27	24	17	14	21
28	20	26	14	17	16	20	16	29	36
26	24	25	20	22	17	18	24	24	21
18	24	25	20	23	17	19	23	15	18
300									
23	19	27	20	21	24	21	25	26	20
23	23	19	21	18	20	19	29	21	17
17	31	30	24	19	26	26	18	22	23
14	28	21	29	16	28	22	22	20	25
19	27	21	27	17	21	19	13	28	27
20	21	27	18	30	19	25	23	22	27
20	23	25	14	19	27	23	17	20	27
21	33	19	23	16	19	22	23	21	20
15	16	21	17	22	23	17	19	15	18
21	21	15	21	21	18	11	25	17	27
400									
14	29	20	15	27	33	26	26	18	27
22	17	24	28	20	29	22	22	13	17
22	16	23	20	23	27	18	21	15	16
17	17	27	19	23	15	26	25	21	31
22	23	25	20	20	15	23	19	22	20
31	23	19	21	26	15	18	20	17	16
30	15	24	22	16	26	16	34	22	32
22	18	21	25	18	24	20	21	25	28
25	18	19	25	24	27	16	21	17	21
18	23	16	20	36	22	21	25	21	22
500									
20	23	21	19	22	24	23	22	21	24
23									

[illegible]

BACKGROUND

COUNTING TIME: 663.2 SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 12156 .
SCALER #2: 6394 .
SCALER #3 241281

THIGGERS: 12957.

ANALYSER CYCLES: 12954.

TOTAL COUNTS ALL CHANNELS: 9733

HNF-29648, Rev. 1

DATE: 4-23-75.
PULSE NUMBER: 42375-3.
PULSED NEUTRON SOURCE MEASUREMENTS - W REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
PULSE NEUTRON SOURCE: KAHAN 808 NO. 137.

DETECTOR: FISSION CHAMBER LAST 241.
DETECTOR LOCATION: CENTER.

LOADING: 75 FUEL COLUMNS. CENTER OUT.

MICROSEC/CHANNEL: (80

PRE-PULSE COUNT RATE: SCALER #1 39192, 39442/80SEC. CP
SCALER #2 808, 755/80SEC. SOURCE
SCALER #3 3/80SEC. OUT.

TOTAL TIME: 400.25EC.
BUILDUP TIME: 120SEC.

TOTAL PULSES: 8495.

ANALYSER CYCLES: 5946.

TOTAL COUNTS ALL CHANNELS: 297673

0									
228	1987	5577	10100	11931	12187	11957	11538	10635	9606
229	1840	7055	6399	5667	5057	4541	3998	3563	3223
230	2459	2311	2059	1804	1630	1397	1287	1179	1055
231	292	231	241	1495	652	601	547	521	522
232	474	417	413	405	397	356	341	327	342
233	315	313	327	318	293	295	315	235	271
234	276	275	279	261	295	291	254	246	232
235	269	249	315	274	305	239	263	258	283
236	280	243	256	266	290	261	257	277	260
237	280	272	266	261	250	249	255	264	300
238	100								
239	252	247	290	268	265	289	270	262	250
240	290	265	292	279	255	243	273	265	269
241	294	282	271	256	264	270	256	253	266
242	226	273	256	284	281	274	259	255	271
243	290	265	240	324	282	294	277	248	255
244	271	260	265	290	266	270	263	252	267
245	235	301	266	246	260	243	261	270	270
246	277	294	264	259	242	255	270	273	247
247	260	290	280	283	276	307	267	274	277
248	264	286	259	259	277	294	265	260	286
249	200								
250	258	260	296	276	278	281	296	260	261
251	278	253	290	293	270	270	262	266	263
252	303	259	285	279	296	291	282	282	297
253	276	272	292	274	263	283	282	283	255
254	285	253	244	304	266	268	270	257	272
255	277	282	271	283	275	253	262	261	275
256	294	273	254	269	286	267	268	270	261
257	261	267	267	274	249	271	307	256	269
258	283	263	279	283	276	280	250	279	252
259	282	214	254	272	284	299	266	258	267
260	300								
261	257	294	266	278	259	252	296	282	234
262	267	283	312	290	285	278	274	252	255
263	299	257	293	267	289	273	257	252	258
264	298	289	297	260	307	295	265	272	259
265	288	264	281	295	271	286	275	263	264
266	282	263	285	253	266	259	245	277	261
267	282	279	278	256	289	269	267	285	265
268	294	249	256	279	263	266	262	243	263
269	261	254	276	287	268	291	265	274	277
270	265	296	232	250	252	281	280	259	254
271	400								
272	284	260	277	263	273	253	257	273	256
273	271	275	275	297	294	248	248	276	312
274	296	254	270	259	290	240	277	275	269
275	273	253	244	274	265	257	271	281	266
276	257	271	273	296	246	283	263	261	266
277	247	282	223	247	279	255	253	248	276
278	265	232	270	256	298	238	282	266	277
279	251	293	256	276	272	269	283	267	265
280	306	273	2503	254	285	296	258	256	245
281	257	281	269	272	242	259	270	272	317
282	300								
283	252	290	242	215	244	244	261	261	263
284	295	271							266

POST PULSE COUNT RATE: SCALER #1 39400, 39743, 39392, 39169/80SEC.
SCALER #2 750, 813, 807, 775/80SEC.
SCALER #3 3.3, 2.2/80SEC.

BACKGROUND

COUNTING TIME: 331.05EC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 163890.

SCALER #2: 3199.
SCALER #3 10.

FIGGERS: 7053

ANALYSER CYCLES: 7053.

TOTAL COUNTS ALL CHANNELS: 142624

HNF-29648, Rev. 1

DATE: 4-23-78.
PULSE NUMBER: 42378-4.
PULSED NEUTRON SOURCE MEASUREMENTS - W REACTOR FUEL STORAGE BASIN.

3.2 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.

WD 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PMS TRIGGER.
ELSE NEUTRON SOURCE: KAMAN 808 NO. 137.

DETECTOR: FISSION CHAMBER LASTL 387.
DETECTOR LOCATION: 3 O'CLOCK.

LOADING: 60 FUEL COLUMNS. CENTER OUT.

MICROSEC/CHANNEL: 80

PRE-PULSE COUNT RATE: SCALER #1 9447,452,481/80SEC. CF
SCALER #2 447,452,481/80SEC. SOURCE
SCALER #3 20329,20340,20340/80SEC. OUT.

TOTAL TIME: 326.65EC.
BUILDUP TIME: 120SEC.

TOTAL PULSES: 11458.

ANALYSER CYCLES: 8844.

TOTAL COUNTS ALL CHANNELS: 8857

0	0	0	0	0	0	0	0	0	0
301	280	632	664	664	576	514	449	341	351
60	288	244	198	155	131	110	112	85	78
16	40	53	40	43	42	18	22	20	16
7	20	12	14	6	10	6	8	9	9
5	6	8	13	5	5	7	6	1	3
1	7	4	16	2	1	2	4	5	5
4	4	5	5	5	4	6	3	6	4
3	3	7	7	7	1	3	1	1	1
3	4	2	3	4	6	6	4	3	3
100	5	9	4	4	6	9	3	3	5
6	5	6	1	4	4	3	6	6	4
3	4	14	3	8	3	4	3	6	5
2	1	5	5	7	4	4	9	10	3
7	6	6	4	4	10	3	2	4	2
6	5	9	2	7	2	3	3	3	8
6	3	5	2	2	2	4	4	4	9
2	9	2	7	5	3	5	2	4	5
6	3	9	5	1	6	5	2	3	5
3	15	1	7	4	7	7	3	6	2
200	4	1	6	3	2	9	5	4	6
4	5	8	2	5	4	5	2	6	4
10	3	5	5	5	2	7	7	2	7
9	3	6	5	5	2	4	6	2	6
6	7	1	5	4	7	3	4	5	4
12	4	4	3	6	9	8	4	6	9
5	7	3	6	6	1	3	3	8	5
0	4	5	4	5	3	4	6	5	4
300	2	6	5	5	3	5	6	4	8
3	7	4	6	4	6	5	0	5	1
8	4	3	5	5	5	4	9	5	3
2	6	4	5	4	3	4	3	5	3
5	3	6	5	1	2	4	6	5	7
3	3	3	5	2	5	5	7	3	3
5	5	5	5	5	7	6	3	7	3
4	2	4	8	6	4	4	6	4	2
7	11	4	2	5	3	4	4	6	4
400	4	4	4	3	9	7	6	6	6
2	4	4	4	1	2	6	5	2	7
3	3	1	4	6	3	3	6	7	4
6	2	7	5	3	5	7	9	4	1
1	5	4	2	8	5	7	6	6	9
4	5	4	4	3	2	2	4	6	3
5	10	10	4	3	1	9	4	6	3
4	7	7	3	3	5	10	3	2	3
1	5	5	5	2	7	3	3	5	2
4	2	3	6	3	5	7	4	4	3
5	6	5	3	3	5	3	0	5	5
500	5	6	2	6	7	6	5	10	7
5	2	6	2	6	7	6	5	10	7

POST PULSE COUNT RATE: SCALER #1 460,453,462/80SEC.
SCALER #2 253,263,298/80SEC.
SCALER #3 20321,20494,20174/80SEC.

BACKGROUND

COUNTING TIME: 400.0SEC.

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 2357.
SCALER #2: 1387.
SCALER #3 102264.

RITGENS: 898444 5718

ANALYSER CYCLES: 8717.

TOTAL COUNTS ALL CHANNELS: 2089

HNF-29648, Rev. 1

DATE: 4-23-75.
PULSE NUMBER: 42375-6.
PULSED NEUTRON SOURCE MEASUREMENTS - N REACTOR FUEL STORAGE BASIN.

3.8 INCH HEXAGONAL LATTICE OF IRRADIATED MARK-1A FUEL ELEMENTS.

ND 2400 ANALYSER DELAYED 20 MICROSECONDS FOLLOWING PWS TRIGGER.
ELSE NEUTRON SOURCE: KAMAN 808 NO. 137.

DETECTOR: FISSION CHAMBER LASL 241.
DETECTOR LOCATION: CENTER.

LOADING: 84 FUEL COLUMNS. CENTER OUT.

MICROSEC/CHANNEL: 50

PRE-PULSE COUNT RATE: SCALER #1 11944,12038,12214/40SEC. CF SOURCE OUT.
SCALER #2 75,85,65/40SEC.
SCALER #3 ZERO.

TOTAL TIME: .525.65EC
BUILDUP TIME: 125.05EC

TOTAL PULSES: 10739.

ANALYSER CYCLES: 8182.

TOTAL COUNTS ALL CHANNELS: 211993

[illegible]

POST PULSE COUNT RATE: SCALER #1 12045,11032,12196/40SEC .
SCALER#2 75,65,77/40SEX .
SCALER #3 ZERO .

BACKGROUND

સાથે જોડેલા સ્થાનના નામો આપવાના છે.

COUNTING TIME: 400 SEC .

TOTAL COUNTS DURING BACKGROUND COUNTING TIME: SCALER #1 120686 .
SCALER #2: 767 .
SCALER #3 3 .

TRIGGERS: 8184.

ANALYSEN CYCLES: 6185.

TOTAL COUNTS ALL CHANNELS: 101364

HNF-29648, Rev. 1

Pulse Neutron Data

Pulse Designation	GRRHO	GOZRHO	ALPHA	Delayed Neutron Level	Background Level
4-4-75 Pulse 2	0.00	0.00	4058.14	0.00	0.00
4-4-75 Pulse 3	0.00	0.00	3810.35	0.00	0.00
4-5-75 Pulse 1	0.00	0.00	3954.20	0.00	0.00
4-5-75 Pulse 2	0.00	0.00	3756.30	0.00	0.00
4-7-75 Pulse 1	-57.98	-48.00	2139.18	6.35	89.72
4-7-75 Pulse 3	-30.05	-27.82	1823.88	18.74	138.38
4-7-75 Pulse 4	-31.62	-29.74	1824.95	15.91	133.21
4-7-75 Pulse 5	-24.56	-26.19	1830.02	0.70	10.66
4-8-75 Pulse 3	-21.60	-21.38	1546.33	28.71	220.20
4-8-75 Pulse 4	-20.88	-20.52	1440.48	1.09	18.12
4-9-75 Pulse 1	-18.56	-18.93	1438.73	34.89	255.20
4-9-75 Pulse 2	-9.70	-12.83	1393.37	17.27	162.73
4-10-75 Pulse 1	-15.29	-16.22	1229.15	44.53	421.87
4-10-75 Pulse 3	-13.43	-13.92	1257.00	41.85	432.52
4-10-75 Pulse 4	-16.14	-16.58	1413.96	27.36	2.81
4-10-75 Pulse 5	-10.15	-10.86	1442.25	37.75	3.37
4-11-75 Pulse 1	-29.43	-29.74	1763.43	10.14	1.66
4-11-75 Pulse 3	-104.11	-84.85	2391.30	2.31	0.78
4-16-75 Pulse 1	-68.56	-73.16	1676.43	0.21	9.27
4-17-75 Pulse 1	-21.87	-24.51	1573.55	1.69	12.17
4-21-75 Pulse 1	-17.84	-22.49	1494.83	1.50	15.55
4-22-75 Pulse 1	-13.37	-15.92	1342.47	4.43	23.48
4-22-75 Pulse 2	-10.67	-12.59	1260.54	6.51	31.62
4-22-75 Pulse 3	-12.61	-14.44	1355.85	5.11	31.41
4-23-75 Pulse 1	-9.26	-10.35	1360.98	75.27	395.20
4-23-75 Pulse 2	-15.80	-19.60	1626.18	2.74	19.01
4-23-75 Pulse 3	-12.51	-14.33	1633.91	36.71	234.84
4-23-75 Pulse 4	-36.63	-43.48	2047.49	0.44	4.14
4-23-75 Pulse 6	-21.23	-25.47	2284.97	13.76	197.90