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Reactivity Measurements on an Experimental Assembly of 4.31 Wt% ²³⁵U Enriched UO₂ Fuel Rods Arranged in a Shipping Cask Geometry

S. R. Bierman

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Office of Civilian Radioactive Waste Management
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**Pacific Northwest Laboratory
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REACTIVITY MEASUREMENTS ON AN EXPERIMENTAL
ASSEMBLY OF 4.31 WT% ^{235}U ENRICHED UO_2 FUEL
RODS ARRANGED IN A SHIPPING CASK GEOMETRY

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SUMMARY

A research program was initiated for the U.S. Department of Energy (DOE) Sandia National Laboratory Transportation Systems Development Department in 1982 to provide benchmark type experimental criticality data in support of the design and safe operations of nuclear fuel transportation systems. The overall objective of the program is to identify and provide the experimental data needed to form a consistent, firm, and complete data base for verifying calculational models used in the criticality analyses of nuclear transport and related systems. A report, PNL-6205, issued in June 1988 (Bierman 1988) covered measurement results obtained from a series of experimental assemblies (TTC-1, 2, 3 and 4) involving neutron flux traps. The results obtained on a fifth experimental assembly (TTC-5), modeled after a calculational problem of the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group, are covered in this report. The experiments were performed at the U.S. Department of Energy Hanford Critical Mass Laboratory, operated by Pacific Northwest Laboratory of Battelle Memorial Institute, with funds jointly provided by The Office of Civilian Radioactive Waste Management and the Defense Program.

Five problems were prepared by the CSNI Workgroup as part of an initial exercise in comparing criticality computational techniques and capabilities of the member nations. The first three problems consisted of a series of critical experiments and the last two consisted of hypothetical fuel transport casks that had characteristics similar to existing cask design. No experimental data existed on either of these hypothetical cask arrangements and conditions. Thus, although intercomparison between calculational results could be made, direct comparison of the calculations with experiments was not possible. The TTC-5 assembly described in this report was modeled after the CSNI Problem 5 to provide data on an experimental assembly having the complex geometrical features of the CSNI Problem 5.

The experimental assembly consisted of a seven unit array of 4.31 wt% ²³⁵U enriched UO₂ fuel rods arranged in water similar to the CSNI Problem 5. As in Problem 5, each unit was encased in a borated aluminum sleeve.

Neutronically the TTC-5 assembly conditions differ from those of Problem 5 in that to achieve a desired k_{eff} of 0.9 or greater, fuel rods were removed from some of the TTC-5 fuel units increasing the water-to-fuel ratio, whereas in Problem 5 the water density between units was reduced. Measurements of k_{eff} were made using the pulse neutron source method. The TTC-5 assembly with 1264 fuel rods had a k_{eff} of 0.920 ± 0.005 and provided data for validating calculations on systems having the complex geometrical features of the CSNI Problem 5.

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REACTIVITY MEASUREMENTS ON AN EXPERIMENTAL ASSEMBLY
OF 4.31 WT% ^{235}U ENRICHED UO_2 FUEL RODS
ARRANGED IN A SHIPPING CASK GEOMETRY

1.0 INTRODUCTION

A research program was initiated for the U.S. Department of Energy (DOE) Sandia National Laboratory Transportation Systems Development Department in 1982 to provide benchmark type experimental criticality data in support of the design and safe operations of nuclear fuel transportation systems. The overall objective of the program is to identify and provide the experimental data needed to form a consistent, firm, and complete data base for verifying calculational models used in the criticality analyses of nuclear transport and related systems. A report, PNL-6205, issued in June 1988 (Bierman 1988) covered measurement results obtained from a series of experimental assemblies (TTC-1, 2, 3 and 4) involving neutron flux traps. The results obtained on a fifth experimental assembly (TTC-5), modeled after a calculational problem of The Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group, are covered in this report.

The CSNI is an international committee of scientists and engineers established in 1973 to develop and coordinate the NEA's work in nuclear safety matters. As the result of a CSNI meeting convened to investigate setting up international projects on nuclear fuel cycle safety, a working group was created to develop an acceptable method of intercomparison and validation of criticality computations used in nuclear fuel transport. The objective of the working group was to provide increased assurances that the certification process of an originating member country could be examined and approved by another member country in which the transport was to occur.

Five problems were prepared by the CSNI Workgroup as part of an initial exercise in comparing criticality computational techniques and capabilities of the member nations. The first three problems consisted of a series of critical experiments and the last two consisted of hypothetical fuel transport casks that had characteristics similar to existing cask design. No

experimental data existed on either of these hypothetical cask arrangements and conditions. Thus, although intercomparison between calculational results could be made, direct comparison of the calculations with experiments was not possible. The objective of the measurements covered in this report is to provide experimental data on an assembly of fuel similar to that identified as Problem 5 in this initial CSNI Exercise, (CSNI Working Group 1982).

The experiments were performed at the U.S. Department of Energy Hanford Critical Mass Laboratory, operated by the Pacific Northwest Laboratory of Battelle Memorial Institute with funds provided by The Office of Civilian Radioactive Waste Management. The experimental measurements are described in detail in the sections that follow. Briefly, however, the experimental assembly consisted of a seven unit array of 4.31 wt% ^{235}U enriched UO_2 fuel rods arranged in water as illustrated in the photograph shown in Figure 1.1. Each unit of fuel was encased in a 0.384 ± 0.002 cm thick B-Al metal sleeve. The number of fuel rods and spacing in each unit of fuel was varied to obtain a k_{eff} in the range calculated by the CSNI working group for Problem 5 ($0.900 < k_{\text{eff}} < 0.955$). The maximum number of fuel rods used in the seven unit array ranged up to 1372. The pulse neutron source method was used in the reactivity determinations.

FIGURE 1.1. Photograph of Experimental Assembly TTC-5

FIGURE 1.1. Photomicrograph of a polyethylene (PE) film.

2.0 DESCRIPTION OF EXPERIMENTAL ASSEMBLY

The experimental system and materials associated with the measurements covered by this report are described in this section.

2.1 GENERAL

An existing experimental system used previously in fuel element array studies at the Critical Mass Laboratory was used in performing the measurements covered in this report. This system, the Fuel Element Array System (FEAS), consists of a 1.8 m x 3 m x 2.1 m deep, open-top, carbon-steel tank provided with a moderator/reflector dump valve, a control blade drive, a safety blade drive, a water deionizer, and associated electronic detection and interlock devices. An overall photograph of the system is shown in Figure 2.1.

The control and safety blade drive systems shown in Figure 2.1 were modified to permit replacing the blades with rod-type devices. The experimental assembly shown in Figure 1.1 was positioned in the large tank beneath these drive systems. As shown in Figure 1.1, the experimental assembly consisted of seven units of fuel. Each unit of fuel consisted of 4.31 wt% ^{235}U enriched UO_2 rods enclosed in identical B-Al metal sleeves. The sleeves were positioned equal distance (4.95 ± 0.04 cm) from each other and provided with two lattice plates each to permit loading up to 196 fuel rods into each sleeve on a 1.891 ± 0.001 cm center-to-center square spacing. Each B-Al sleeve covered the active length of the fuel rods. A layout arrangement of the experimental assembly, giving dimensions and relative locations, is shown in Figure 2.2. An overhead photograph view of the experimental assembly is shown in Figure 2.3.

The experimental assembly was fully reflected by at least 15 cm of water on all sides and essentially free of any structural materials in the fuel region. All of the instrument thimbles were located outside the fuel area except for the boron lined proportional detectors used to accumulate neutron count rate data during the measurements. These detectors were located at various locations in the assembly, as indicated in Figure 2.2, to obtain information on spatial dependencies in the data. As will be discussed

FIGURE 2.1 Experimental System Modified for Use in Experiments

FIGURE 2.2 TTC-5 Experimental Assembly Arrangement

FIGURE 2.3 Overhead Photograph of Experimental Assembly TTC-5

in the measurements section to follow, the effect of these detectors on the assembly reactivity was evaluated and found to be negligible. Except for the two 1.2 cm diameter T6061 aluminum guide tubes for the safety-control rods, all other material (lattice and support plates) in the fuel region have neutronic properties similar to the water moderator.

2.2 FUEL RODS

The fuel rods used to construct the experimental assembly were 4.31 wt% ^{235}U enriched UO_2 rods previously obtained by downloading stainless steel clad rods, originally fabricated for Core II of the N. S. Savannah (Katz 1969), and reloading the pellets into type 6061 aluminum tubes (see Appendix A for American Society for Testing Materials, T6061 Al Specifications). A complete description of these rods is given in Figure 2.4. The uranium assay (1059.64 ± 4.80 g/rod) and the ^{235}U enrichment ($4.306 \pm 0.013\%$) shown in Figure 2.4 for these rods are the average of six assays and six spectrographic analyses made on fuel pellets chosen at random during the reloading. The oxide density (10.40 ± 0.06 g UO_2/cm^3) given in Figure 2.4 is based on individual volume displacement measurements with 20 pellets selected at random during the reloading operations. The mass of UO_2 per rod (1203.38 ± 4.12 g) is the average mass of the 1865 rods of this type available for use in the experiments. The fuel diameter (1.265 ± 0.003 cm) given in Figure 2.4 was checked repeatedly during the reloading operations and found to agree with that quoted in the document characterizing Core II of the N. S. Savannah (Katz 1969). The rubber end cap density (1.321 g/ cm^3) quoted in Figure 2.4 is the result of a single mass-volume measurement with six end caps selected at random. The composition of the end caps is the result of four analyses on randomly selected end caps.

2.3 MODERATOR-REFLECTOR

As indicated previously, each assembly was moderated and fully reflected with water. Impurities analyses of a water sample taken during the experiment are given in Appendix B.

Cladding: 6061 Aluminum Tubing

Loading

Enrichment - $4.306 \pm 0.013\%$ ²³⁵U
 Oxide Density - 10.40 ± 0.06 g/cm³
 UO₂ - 1203.38 ± 4.12 g/Rod
 U - 1059.64 ± 4.80 g/Rod

Uranium Composition:

²³⁴U - 0.022 ± 0.002
²³⁵U - 4.306 ± 0.013
²³⁶U - 0.022 ± 0.002
²³⁸U - 95.650 ± 0.017

End Cap:

C- 58 ± 1 wt% S- 1.7 ± 0.2 wt%
 H- 6.5 ± 0.3 wt% O- 22.1 wt% (Balance)
 Ca- 11.4 ± 1.8 wt% Si- 0.3 ± 0.1 wt%

Notes:

1. Error limits are one standard deviation
2. End Cap Density is 1.321 g/cm³

FIGURE 2.4 Description of 4.31 wt% ²³⁵U Enriched UO₂ Fuel Rods

2.4 LATTICE PLATES AND SUPPORTS

The B-Al encased fuel units were supported on a 5.08 cm thick acrylic plate (1.185 g/cm^3 containing 8 wt% H, 60 wt% C and 32 wt% O) mounted off the FEAS tank walls. The elevation of the acrylic support plate, relative to components in the experimental assembly, is shown in Figure 2.5.

Two 1.23 ± 0.01 cm thick polypropylene plates, having a C_3H_6 molecular structure and a density of 0.90 g/cm^3 and supported off the FEAS tank wall, were used to position the B-Al sleeves as indicated in Figure 2.2. One plate was located on the acrylic support plate and one about 25 cm below the top of the B-Al sleeves. Typical trace impurity levels for the polypropylene plates are given in Appendix C.

Each B-Al sleeve was provided with two 1.23 ± 0.01 cm thick polypropylene lattice plates containing 196 holes, 1.448 ± 0.002 cm in diameter, on a 1.891 ± 0.001 cm center-to-center spacing and arranged in a 14×14 pattern as shown in Figure 2.2. The lattice plates were positioned in each B-Al sleeve at the same levels as the polypropylene support plates discussed in the previous paragraph. Trace impurity levels for the polypropylene plates are given in Appendix C.

2.5 B-Al METAL SLEEVES

The metal sleeves enclosing each unit of fuel were obtained from Eagle-Picher Industries, Inc., Quapaw, Oklahoma. Two experimental heats of an AlB_2 alloy were remelted and mixed into five smaller pours from which five AlB_2 plates were rolled. A neutron radiograph of a plate section chosen at random indicated a homogeneous distribution of boron in the alloy. The seven sleeves used in the TTC-5 experimental assembly were fabricated from these five AlB_2 plates.

Each of the seven AlB_2 sleeves were 93.8 ± 0.3 cm in length and had a plate thickness of 0.384 ± 0.002 cm. The AlB_2 alloy contained 3.41 ± 0.57 wt% boron enriched to 98.71 wt% in boron-10 and had an average density of $2.64 \pm 0.03 \text{ g/cm}^3$. The AlB_2 alloy contained the following impurities:

Si	0.16 wt%
Fe	0.07 wt%
Zn	0.02 wt%
Na	0.20 wt%
K	0.77 wt%
Others	<0.01 wt%

FIGURE 2.5 Experimental Assembly Elevations

Complete chemical analytical results for the AlB_2 alloy are presented in Appendix D.

2.6 CONTROL AND SAFETY BLADES

The FEAS control and safety drives were provided with gadolinium rods such that each experimental assembly could be constructed in compliance with operating regulations and procedures. These rods and their associated Type 6061 aluminum guides were located in the fuel regions of the experimental assembly such that the rods fall by gravity into the assembly should an out-of-range condition exist in the control channels shown in Figure 1.1. The control and safety rods are not described in detail since all of the data reported herein were obtained with these rods fully withdrawn from the experimental assembly. ASTM specifications for the aluminum sleeves are given in Appendix A. The aluminum sleeves extended from the acrylic base plate shown in Figure 2.5 to above the top reflector level at locations shown in the loading diagrams in Appendix F. The sleeves had a 1.283 cm inside diameter and a wall thickness of 0.066 cm.

Thermal chemical analytical results for the A152 alloy are presented in

Appendix D.

2.2. CONTROL AND SAFETY RESULTS

The A152 control and safety devices were provided with guidelines rods such that each experimental assembly can be constructed in compliance with operating regulations and procedures. These rods and their associated rod alignment guides were located in the fuel regions of the experimental assembly such that the rods fall by gravity into the assembly should an out-of-control condition exist in the control channels shown in Figure 1.1. The control and safety rods are not depicted in detail since all of the data reported herein were obtained with these rods fully withdrawn from the experimental assembly. A152 specifications for the aluminum sleeves are given in Appendix A. The aluminum sleeves extended from the active base plate shown in Figure 2 to above the top reflector level as shown. The needed diameter in Appendix A. The sleeves had a 1.583 cm inside diameter and a wall thickness of 0.125 cm.

3.0 EXPERIMENTAL MEASUREMENTS AND RESULTS

The experimental measurements performed and the results obtained are covered in this section.

3.1 MEASUREMENT TECHNIQUES

As discussed in the introduction, the purpose of the measurements was to obtain data on an experimental assembly similar to the CSNI Problem 5 having a k_{eff} value between about 0.90 and 0.95. The pulse neutron technique was used to determine k_{eff} in the TTC-5 assembly as fuel loading conditions were varied to obtain a value in this range.

In pulse neutron source measurements, a burst of neutrons is injected into a system in a fraction of a microsecond and its effect on the neutron population at some point in the system is 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 following each burst. The repetition rate and number of neutron bursts required depends on the statistics of the accumulated data, which, in turn, is dependent on the particular system. A proportional counter or a BF_3 tube is generally used for the detection of the neutrons, and a multichannel analyzer is used as a series of sequentially gated scalars for the storage of the neutron counts as a function of time following each burst.

Since the pulse neutron source characteristically produces epithermal neutrons and since the neutron detector generally used is primarily sensitive to thermal neutrons, the count rate will reach a maximum and then begin to decrease exponentially as the epithermal neutrons slow down immediately following a neutron burst and become thermalized. Once an asymptotic distribution of neutrons is reached in a neutron multiplying system, the count rate, as a function of time following the neutron burst, can be described as the sum of two exponential terms:

$$N(t) = A_1 e^{-\alpha_1 t} + A_2 e^{-\alpha_2 t}. \quad (1)$$

One term describes the time behavior of the prompt neutrons and the other describes the time behavior of the delayed neutrons from fissions caused by the pulse neutron source. Since the decay of the delayed neutrons is very slow compared to that of the prompt neutrons, Equation (1) is generally written as:

$$n(t) = A_1 e^{-\alpha_1 t} + N_D \quad (2)$$

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 decreasing exponentially with time at a rate α_1 .

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 upon the system on which the measurements are being made. However, the faster the neutron spectrum, the faster the decay rate will be. Also, the further subcritical a system is, the faster the decay rate will be. Therefore it is possible to obtain the same decay rate for different degrees of subcriticality. It is this ambiguous feature of the prompt neutron decay that 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:

$$\alpha_1 = \frac{1 - k_{eff}(1 - \beta_{eff})}{\ell} \quad (3)$$

where α_1 is the prompt neutron decay rate, β_{eff} the effective delayed neutron fraction, and ℓ the neutron lifetime in the system. However, this requires a knowledge of ℓ and β_{eff} , and ℓ is as difficult to calculate as k_{eff} . However, Equation (3) and the definition for reactivity:

$$\rho \equiv \frac{k - 1}{k}, \quad (4)$$

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

$$\frac{\rho}{\beta} = - \frac{\alpha_1 - k\beta/\ell}{k\beta/\ell}. \quad (5)$$

If the neutron generation time is assumed to be constant, Equation (5) can be reduced to:

$$\frac{\rho}{\beta} = - \frac{\alpha_1 - \alpha_c}{\alpha_c}, \quad (6)$$

where α_c is the prompt neutron decay rate at delayed critical ($k=1$). However, the generation time does vary significantly with the degree of subcriticality. Also, α_c is a parameter not generally available for fuel transport or storage systems.

Three methods have been developed whereby the reactivity of a system can be determined directly from the pulse neutron measurements if the data can be obtained essentially free of energy and spatial effects. The earliest technique was suggested by Sjostrand (1956) who simply separated the data into a prompt neutron component and delayed neutron component, as implied by Equations (1) and (2), 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.

$$\left(\frac{\rho}{\beta}\right)_{\text{SJ}} = - \frac{\text{Prompt Neutron Area}}{\text{Delayed Neutron Area}} \quad (7)$$

Gozani (Gozani et al. 1965) 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 essentially as:

$$\left(\frac{\rho}{\beta}\right)_{GO} = - \frac{\text{Prompt Neutron Fundamental Mode Area}}{\text{Delayed Neutron Area}} \quad (8)$$

Garelis and Russell (1963) developed a theoretical model showing 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.

$$\int_0^{1/R} A_1 e^{-\alpha_1 t} e^{\frac{k\beta}{\ell} t} dt = \int_0^{1/R} A_1 e^{-\alpha_1 t} dt + \frac{N_D}{R} \quad (9)$$

Where $A_1 e^{-\alpha_1 t}$ 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 (9) for $K\beta/\ell$ is available in the data obtained from the pulse neutron source measurement. Consequently, all the data needed to solve Equation (5) for the reactivity of the system is provided by the pulse neutron source data in the Garelis-Russell model.

If energy-and space-independent kinetics data are not obtained in the measurements, correction factors must be applied to Equations (7), (8), and (9) to account for any harmonic effect and kinetic distortion in the prompt neutron fundamental mode of decay. Although harmonic effects can be due to prompt, delayed, and other complicated forms of harmonics, prompt harmonics are generally the dominant harmonic factor (Gozani 1972). Consequently the extrapolated area or Gozani method of analysis (Eq. 8) is, in general, least affected by harmonics. For this reason, and the fact that the Garelis-Russell (GR) and Gozani (GO) methods of analyses (Eq. 8 and 9) affect the obtained reactivities in opposite ways (Gozani 1972, 1965), both the Gozani and the Garelis-Russell methods were used in analyzing each of the pulse neutron measurements covered in this report.

3.2 DESCRIPTION OF PULSE NEUTRON SYSTEM

The pulse neutron source measurements were made with a Kaman A808 pulsed neutron source capable of being repetitively pulsed up to 50 pulses per second and having a total neutron yield of about 10^9 n/sec. The source is encased in a six-inch diameter oil filled cylinder, 36 inches long, and can be moved about. In all of the measurements the source was located outside the lattice plates at least 15 cm from the fuel rods as indicated in loading diagrams shown in Appendix F. The source was positioned such that its tritium target was located approximately at the horizontal midplane of the assembly fuel region.

Data were simultaneously accumulated in each measurement using up to four neutron detectors and a Nuclear Data, Inc. ND66 channel analyzer operating in the time mode with a 0.25 microsecond dead time between the counting channels. The resolving time for the data accumulations system was one microsecond. Each detector was a Reuter Stokes RS-P7-0812-117 boron-10 lined proportional counters having a 2.62 cm diameter by 35 cm long active region. The active region of each detector was centered at the horizontal midplane of the assembly fuel region. The location of the detectors in each measurement is identified in the respective assembly loading diagrams given in Appendix F. A diagram of the pulsed neutron source control and data acquisition system is shown in Figure 3.1.

3.3 EXPERIMENTAL MEASUREMENTS AND RESULTS

As indicated in Section 3.1, simultaneous sets of measurement data were obtained from multiple detectors to monitor and obtain data on spatial effects. Three detectors were positioned symmetrically around the center fuel unit. A fourth detector was positioned at different radial distances from the center unit as indicated in Figure 2.2 and the data tables.

To provide a degree of confidence in the measurement results, measurements were made with three different fuel loading arrangements to obtain data as a function of reactivity in the assembly. The prompt decay rates as observed by the three detectors located around the center fuel unit are shown in Figure 3.2 as a function of fuel loading. Fuel loading diagrams

FIGURE 3.1 Pulse Neutron Source and Data Acquisition System

FIGURE 3.2 Observed Prompt Neutron Decay Rates About the Center Unit of the TTC-5 Assembly

for each of the measurements are shown in Appendix E. There appears to be no spatial dependencies in these decay constants. Corresponding average k_{eff} values are shown in Figure 3.3. Although, not unexpectedly, the k_{eff} values determined by the two analyses techniques differ, both sets of measurements result in uniformly increasing k_{eff} values for the assembly as the reactivity is increased by varying the fuel loading arrangement. (It should be noted that the changes in reactivity in the assembly are due to changes in neutron moderation - not the amount of fuel in the assembly). As normally observed in these type of measurements, the difference in k_{eff} between the Gozani and the Garelis-Russell analyses decreases as the loading arrangement approaches the critical condition. The measurement results obtained for each data point shown in Figures 3.2 and 3.3 are tabulated in Table 3.1. The measurement data, on which these results are based, are given in Appendix F.

With the most reactive fuel loading arrangement of 1264 fuel rods, measurement data were accumulated simultaneously with neutron detectors positioned radially outward 16.17 cm, 32.55 cm and 50.33 cm (Locations 3, 4 and 5 shown in Figure 2.2). The prompt neutron decay constants and k_{eff} values measured at these locations are given in Table 3.1. The average of these data are shown graphically in Figure 3.4. As can be seen in Figure 3.4, the Gozani analyses of these measurement data appears to be more spatially dependent than the Garelis-Russell analyses, especially as the center of the assembly is approached. Consequently an extrapolation of the Garelis-Russell analyses to the assembly center, where kinetic distortion should be minimal, probably yields the better value of k_{eff} for the assembly. This extrapolation of the Garelis-Russell analyses yields a k_{eff} value of 0.92 for the assembly compared to 0.91 obtained by a similar extrapolation of the Gozani analyses. Consequently, even if the measurement data contains some harmonic contamination, k_{eff} for the assembly lies somewhere between 0.91 and 0.92 since the Gozani and the Garelis-Russell methods of analyses have a bracketing effect with respect to harmonics.

Since the boron lined proportional counters used in the measurements could have an effect on the reactivity of the assemblies, measurements were made using the detector at Location 3 with the assembly containing 1264 fuel rods and detector at Location 1 removed from the assembly. Essentially no

FIGURE 3.3. Average of k_{eff} Values Determined About the Center Fuel Unit of the TTC-5 Assembly

TABLE 3.1. Measurement Results and Summary Description
of TTC-5 Assembly Conditions

Detector \ Fuel Locations \ Loading (a)	1372 Rods (b)			1329 Rods (b)			1264 Rods (b)		
	Prompt Neutron Decay Rate $\alpha \text{ sec}^{-1} \pm 1\sigma$	Garells- Russell k (c)	Gozani k (c)	Prompt Neutron Decay Rate $\alpha \text{ sec}^{-1} \pm 1\sigma$	Garells- Russell k (c)	Gozani k (c)	Prompt Neutron Decay Rate $\alpha \text{ sec}^{-1} \pm 1\sigma$	Garells- Russell k (c)	Gozani k (c)
1	6299 $\pm$ 66	0.877	0.830	4298 $\pm$ 46	0.903	0.887	3716 $\pm$ 97	0.914	0.902
	6514 $\pm$ 177	0.861	0.769	4375 $\pm$ 38	0.899	0.882	3732 $\pm$ 84	0.915	0.903
2	6078 $\pm$ 136	0.882	0.813	4054 $\pm$ 44	0.912	0.907	3540 $\pm$ 35	0.918	0.904
	—	—	—	4217 $\pm$ 67	0.915	0.900	3628 $\pm$ 46	0.915	0.901
3	6202 $\pm$ 188	0.901	0.789	4117 $\pm$ 72	0.907	0.866	3696 $\pm$ 94	0.919	0.886
	6159 $\pm$ 159	0.873	0.795	4238 $\pm$ 89	0.900	0.843	3702 $\pm$ 58 (d)	0.919	0.833
4	—	—	—	4006 $\pm$ 159	0.886	0.814	3643 $\pm$ 123 (d)	0.912	0.869
5	—	—	—	—	—	—	3447 $\pm$ 190 (d)	0.861	0.755
	—	—	—	—	—	—	3197 $\pm$ 133 (d)	0.908	0.866

- (a) As shown in Figure 2.2
 (b) Refer to appropriate appendix for fuel rod arrangement in each fuel unit
 (c) Calculated from dollars of reactivity using 0.0065 for β
 (d) Detector moved from Location 1 to Location 5

FIGURE 3.4. Observed Spatial Dependence of k_{eff} and Prompt Neutron Decay Rate in the TTC-5 Assembly With the Most Reactive Fuel Loading Arrangement

change was observed in the reactivity of the assembly with and without a detector at Location 1. The prompt neutron decay rate measured at Location 3 was $3696 \pm 94 \text{ sec}^{-1}$ and $3707 \pm 58 \text{ sec}^{-1}$ respectively with and without a detector at Location 1. Garelis-Russell analyses of the measurement data resulted in the same k_{eff} value of 0.919 with and without the detector at Location 1 and the Gozani analyses gave values of 0.886 and 0.883 respectively.

FIGURE 3.12. Prompt neutron decay rate in the TUC-2 assembly, 1968. The Garelis-Russell and Gozani analyses are shown for comparison.

4.0 CONCLUSIONS

The TTC-5 experimental assembly arrangement with a 1264 fuel rod loading provides a set of experimental conditions that have a k_{eff} of 0.92 within about 0.005 in k_{eff} . This reactivity level was achieved by removing fuel rods in some of the fuel units to increase the neutron moderation. Although this differs from the CSNI Working Group's Problem 5 in which the moderator density between fuel units was reduced to achieve a k_{eff} above 0.9, the TTC-5 experimental assembly provides the necessary conditions for validating calculations on systems having the complex geometrical features of the CSNI Problem 5.

Since replacement type measurements demonstrated that the neutron detectors located in the fuel region have no observable effect on the measurement data, the detectors can be ignored when modeling the TTC-5 experimental assembly in computer code validations. Also, in the design of the experiments it was concluded that the biological shield in CSNI Problem 5 would have no reactivity effect on the TTC-5 assembly. It should be noted, however, that experimental criticality data exists on the reactivity effect of these biological materials in close proximity to arrays of fuel rods (Bierman 1979, 1981).

5.0 ACKNOWLEDGEMENTS

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- G. E. Whitesides of Oak Ridge National Laboratory for his assistance in planning this and other experiments in this program.

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APPENDIX A

Density and Chemical Composition of Type 6061 Aluminum

APPENDIX A

Density and Chemical Composition of Type 6061 Aluminum

Density and American Society for Testing Materials (ASTM) chemical specifications for the 0.066 cm thick, 1.283 cm inside diameter, aluminum safety and control rod guide sleeves in the experimental assemblies are presented in this appendix.

TABLE A.1. ASTM Standard B210-78 Specifications for Type 6061 Aluminum

<u>Chemical Composition</u>	
<u>Element</u>	<u>Wt%</u>
Si	0.40-0.80
Fe	0.7 (Maximum)
Cu	0.15-0.40
Mn	0.15 (Maximum)
Mg	0.8-1.2
Cr	0.04-0.35
Zn	0.25 (Maximum)
Ti	0.15 (Maximum)
Al	Remainder
	97.36 96.15

<u>Maximum Impurities</u>	
<u>Element</u>	<u>Wt%</u>
Each	0.05
Total	0.15

Density: 2.69 g/cm³

(not part of standard - measured by volume displacement)

APPENDIX B

Trace Impurities Present in Water
Used as Moderator-Reflector in Experiments

APPENDIX B

Trace Impurities Present in Water Used as Moderator-Reflector in Experiments

Samples were taken during the experiments of the water moderator-reflector. The results of the sample analyses are presented in this appendix. All analyses were performed in accordance with Standard Methods for the Examination of Water and Waste Water, 15th ed., American Public Health Association, Washington, DC. The analysis was performed by the Hanford Environmental Health Foundation, Richland, Washington.

TABLE B.1. Analysis of Water Sample

	<u>Sample Number</u> <u>TTC-5</u>
pH	7.6
Total alkalinity mg/liter as CaCO ₃	58
Bicarbonate Alkalinity mg/liter as CaCO ₃	54
Carbonate alkalinity mg/liter as CaCO ₃	<0.5
Total dissolved solids mg/liter	91
Sulfate mg/liter	1.1
Nitrate (as N) mg/liter	0.13
Fluoride mg/liter	<0.2
Chloride mg/liter	14.5
Cadmium mg/liter	0.0008
Copper mg/liter	<0.05
Chromium mg/liter	<0.005
Iron mg/liter	0.07
Lead mg/liter	<0.005
Manganese mg/liter	<0.01
Zinc mg/liter	0.21
Boron mg/liter	<50
Gadolinium mg/liter	<40

APPENDIX C

Impurities Measured in Polypropylene Plate Material

APPENDIX C

Trace Impurities Measured in Polypropylene Plate Material

Trace impurity levels are presented in this appendix for the polypropylene plate material used in these experiments. The analytical results are from a spark source mass spectrographic analysis of samples taken from, and considered representative of, the polypropylene plates. The spectrographic analysis were performed by the Hanford Engineering Development Laboratory, Richland, Washington.

TABLE C.1. Trace Impurities Present in Polypropylene Support Lattice Plates (Parts per Million by Weight)

<u>Element</u>	<u>Sample Number</u> <u>TTC-5</u>
Li	7
Be	1
B	0.04
F	-----
Na	20
Mg	10
Al	30
Si	8
P	5
Cl	1
K	60
Ca	-----
Ti	100
V	10
Cr	50
Mn	2
Fe	-----
Cu	100
Zn	10
Ge	-----
As	20
Rb	-----
Y	-----
Zr	-----
Mo	-----
Sn	-----
Br	-----

APPENDIX D

AlB₂ Alloy Sample Analyses

APPENDIX D

AlB₂ Alloy Sample Analyses

Analytical results obtained from samples of the AlB₂ alloy used to fabricate the seven B-Al sleeves used for these experiments are presented in this appendix.

TABLE D.1. Results of Sample Analyses
for Wt% B in the AlB₂ Alloy

<u>Sample</u>	<u>E-P</u> ⁽¹⁾	<u>KBI</u> ⁽²⁾
R8702AI 1-A	3.15	2.7
R8702AJ 1-B	3.70	3.4
R8702AK 1-D	3.15	3.7
R8702AL 2-T	3.44	2.6
R8702AM 2-B	2.85	3.5
R8702AN 2-D	2.95	3.5
R8702AO 3-A	3.44	3.0
R8702AP 3-B	2.53	3.1
R8702AQ 3-D	3.21	3.8
R8702AR 4-T	3.86	3.7
R8702AS 4-B	3.48	4.4
R8702AT 4-D	3.49	4.2
R8702AU 5-T	4.53	2.1
R8702AV 5-B	3.13	3.9
R8702AW 5-D	3.40	4.4

⁽¹⁾ Eagle-Picher Research Laboratory, Miami, Oklahoma
⁽²⁾ Cabot Corporation Laboratory, Robards, Kentucky

TABLE D.2. Results of Sample Analyses for Impurities
in the Two Experimental Heats of AlB₂⁽¹⁾

<u>Impurity</u>	<u>Heat 1 (Wt%)</u>	<u>Heat 2 (Wt%)</u>
Si	0.19	0.13
Fe	0.08	0.05
Zn	0.02	0.02
Na	0.22	0.17
K	0.83	0.71
Others	<0.01	0.01

⁽¹⁾ Cabot Corporation Laboratory, Robards, Kentucky

TABLE D.3. Results of Sample Analyses for Isotopic
Composition of Boron Used in the AlB₂ Alloy

<u>Sample</u> ⁽¹⁾	<u>B-10</u>		<u>B-11</u>	
	<u>Atomic %</u>	<u>Weight %</u>	<u>Atomic %</u>	<u>Weight %</u>
R8609AE	98.82	98.71	1.18	1.29

⁽¹⁾ Eagle-Picher Research Laboratory, Miami, Oklahoma

APPENDIX E

Loading Diagrams of TTC-5 Assembly

APPENDIX E

Loading Diagrams of TTC-5 Assembly

Measurements were made with five fuel/detector arrangements in the TTC-5 experimental assembly. These arrangements of the TTC-5 experimental assembly and the locations of the control/safety rod guide tubes are presented in this appendix. Each arrangement can be related to the measurement results discussed in the main body of the report by the number of fuel rods loaded in the assembly.

TTC-5 Assembly Fuel Loading Arrangement

③

Assembly Number : TTC-5
 Lattice Pitch : 1.891 ± 0.001 cm
 Safety Rods : Out of System
 Control Rods : Out of System
 Fuel Rods : 1372
 Fuel : 4.31 wt% ^{235}U Enriched UO_2

Legend

② Detector Locations

○ Locations with Fuel

TTC-5 Assembly Fuel Loading Arrangement

Assembly Number : TTC-5
 Lattice Pitch : 1.891 ± 0.001 cm
 Safety Rods : Out of System
 Control Rods : Out of System
 Fuel Rods : 1329
 Fuel : 4.31 wt% ^{235}U Enriched UO_2

Legend

- ② Detector Locations
- Locations with Fuel
 - Locations without Fuel
 - ▲ Control & Safety Rod Guide Tubes

TTC-5 Assembly Fuel Loading Arrangement

③

①

②

Assembly Number : TTC-5
 Lattice Pitch : 1.891 ± 0.001 cm
 Safety Rods : Out of System
 Control Rods : Out of System
 Fuel Rods : 1264
 Fuel : 4.31 wt% ^{235}U Enriched UO_2

Legend

② Detector Locations

○ Locations with Fuel

● Locations without Fuel

▲ Control & Safety Rod Guide Tubes

TTC-5 Assembly Fuel Loading Arrangement

Assembly Number : TTC-5
 Lattice Pitch : 1.891 ± 0.001 cm
 Safety Rods : Out of System
 Control Rods : Out of System
 Fuel Rods : 1264
 Fuel : 4.31 wt% ^{235}U Enriched UO_2

Legend

- ② Detector Locations
- Locations with Fuel
 - Locations without Fuel
 - ▲ Control & Safety Rod Guide Tubes

APPENDIX F

Computerized Plots of Pulse Neutron Source Measurement Data

APPENDIX F

Computerized Plots of Pulse Neutron Source Measurement Data

The observed time behavior of the neutron flux in each of the pulsed neutron source measurements is presented in this appendix as photo reproductions of computer plots obtained following data reduction and analysis. Each set of data are identified by the assembly number (TTC-5), the experiment number (S232) and the location of the detector used in obtaining the measurement data. For example, 3D1 is the data obtained during measurement number 3 with the detector at Location 1. Missing measurement and detector location numbers are associated with equipment testing, partial loss of measurement data during transmission, data not obtained at location, measurement interruption, etc.

In each of the measurements, 256 data channels were used in accumulating and analyzing the time dependent data. All of the data have been corrected for coincidence losses. The least-square calculated fit of the data is also shown in each of the plots. The data indicated by an "X" were not considered in the least-squares fitting process; however, these data were included in the (Garelis and Russell 1963) reactivity determinations. The first data channel used in the least-squares fitting is identified as start channel in the legend.

The fundamental mode and the delayed neutron plus background level, as determined by the least-square fitting of the data, are shown as solid lines in each figure. The slope of the fundamental mode curve is tabulated in the legend of each figure as α in sec.^{-1} . Also included in the legend of each figure are the Garelis-Russell (GRRHO) and the Gozani (GOZRH0) determined reactivities in dollars, the neutron background level during each measurement, the delayed neutron level as determined by the least-squares fit, and the data accumulating channel widths in microseconds.

Both the Gozani and the Garelis-Russell methods of data analyses require that a fundamental mode of decay be established. To determine if a single mode of neutron decay eventually existed in each set of measurement data, a technique developed by Pfeiffer, Brown and Marshall (Pfeiffer 1974)

in a Fort St. Vrain startup test was used. The technique makes use of the standard deviation obtained on the fitted value for the prompt neutron decay rate, α , and the random fluctuation of the fitted value of α within the standard deviation as measurement data immediately following the neutron burst is successively ignored in the fitting process. If it can be demonstrated that a decay rate, and the associated standard deviation, fluctuates within the standard deviation as data are deleted, it can be concluded that a single time rate of decay predominates and is the fundamental mode of neutron decay for that experimental assembly.

TTC-5 S232-3D1 ASSEMBLY FULLY LOADED WITH FUEL.
DETECTOR AT LOCATION I. 10SEPT87. 22C.

GRRHO	=	-21.61
GOZRHO	=	-31.59
ALPHA	=	6.29936E+03
DELAYED NEUTRON LEVEL	=	5.74515E+00
BACKGROUND LEVEL	=	2.65493E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	9

TTC-3 S232-3D3 ASSEMBLY FULLY LOADED WITH FUEL.
DETECTOR AT LOCATION 3. 11SEPT87. 22C.

TTC-5 S232-4D1 ASSEMBLY FULLY LOADED WITH FUEL.
 DETECTOR AT LOCATION 1. 21SEPT87. 22C.

TTC-5 S232-4D3 ASSEMBLY FULLY LOADED WITH FUEL
DETECTOR AT LOCATION 3. 21SEPT.88. 22C.

GRRHO	=	-16.95
GOZRHO	=	-41.13
ALPHA	=	6.20186E+03
DELAYED NEUTRON LEVEL	=	2.56412E+00
BACKGROUND LEVEL	=	2.99219E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	14

TTC-5 S232-4D2 ASSEMBLY FULLY LOADED WITH FUEL.
DETECTOR AT LOCATION 2. 21SEPT87. 22C.

TTC-5 S232-5D1 ASSEMBLY FULLY LOADED LESS 25 IN 7 AND 9 EACH IN 3&6.
 DETECTOR AT LOCATION 1. 23SEPT87. 22C.

GRRHO	=	-16.45
GOZRHO	=	-19.60
ALPHA	=	4.29765E+03
DELAYED NEUTRON LEVEL	=	3.52895E+00
BACKGROUND LEVEL	=	2.68463E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	9

TTC-5 S232-5D2 ASSEMBLY FULLY LOADED LESS 25 IN SUBASSEMBLY 7 & 9 RODS IN 3&6
DETECTOR 2 AT LOCATION 2.23SEPT88. 22C.

F.9

TTC-5 S232-5D3 ASSEMBLY FULLY LOADED LESS 25 IN 7 AND 9 EACH IN 3&6.
DETECTOR AT LOCATION 3. 23SEPT.87 22C.

TTC-5 S232-5D4 ASSEMBLY FULLY LOADED LESS 25 RODS IN SUBASSEMBLY 7 & 9 RODS
 DETECTOR AT LOCATION 4. 10OCT87. 22C.

GRRHO	=	-19.69
GOZRHO	=	-35.23
ALPHA	=	4.00572E+03
DELAYED NEUTRON LEVEL	=	8.48581E-01
BACKGROUND LEVEL	=	3.92028E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	17

TTC-5 S232-6D1 ASSEMBLY FULLY LOADED LESS 25 IN 7 AND 9 EACH IN 3&6.
 DETECTOR AT LOCATION 1.23SEPT87. 22C.

GRRHO	=	-17.21
GOZRHO	=	-20.68
ALPHA	=	4.37481E+03
DELAYED NEUTRON LEVEL	=	4.08687E+00
BACKGROUND LEVEL	=	2.47892E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	8

TTC-5 S232-6D2 ASSEMBLY FULLY LOADED LESS 25 RODS IN 7 & 9 EACH IN 3&6.
 DETECTOR AT LOCATION 2. 23SEPT87. 22C.

GRRHO	=	-14.28
GOZRHO	=	-18.06
ALPHA	=	4.21661E+03
DELAYED NEUTRON LEVEL	=	4.41593E+00
BACKGROUND LEVEL	=	2.55119E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	13

TTC-5 S232-6D3 ASSEMBLY FULLY LOADED LESS 25 IN 7 AND 9 EACH IN 3&6.
DETECTOR AT LOCATION 3. 23SEPT.87. 22C.

TTC-5 S232-7D1 ASSEMBLY FULLY LOADED LESS 36 RODS IN 3, 6&7 SUBASSEMBLIES.
DETECTOR AT LOCATION 1. 25SEPT87. 22C.

TTC-5 S232-7D2 ASSEMBLY FULLY LOADED LESS 36 RODS IN 3.6&7 SUBASSEMBLIES.
 DETECTOR AT LOCATION 2. 25SEPT87. 22C.

CRRHO	=	-13.80
GOZRHO	=	-16.34
ALPHA	=	3.54018E+03
DELAYED NEUTRON LEVEL	=	3.75696E+00
BACKGROUND LEVEL	=	2.61394E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	10

F.16

TTC-5 S232-7D3 ASSEMBLY FULLY LOADED LESS 36 RODS IN 3, 6&7 SUBASSEMBLIES.
 DETECTOR AT LOCATION 3. 25SEPT87. 22C.

TTC-5 S232-8D1 ASSEMBLY FULLY LOADED LESS 36 RODS IN 3,6&7 SUBASSEMBLIES.
DETECTOR AT LOCATION 1. 25SEPT87. 22C.

GRRHO = -14.34
GOZRHO = -16.58
ALPHA = 3.73179E+03
DELAYED NEUTRON LEVEL = 3.73882E+00
BACKGROUND LEVEL = 2.31780E+00
CHANNEL WIDTH = 5.02500E-05
START CHANNEL = 17

TTC-5 S232-8D2 ASSEMBLY FULLY LOADED LESS 36 RODS IN 3, 6&7 SUBASSEMBLIES.
 DETECTOR AT LOCATION 2. 25SEPT87. 22C.

TTC-5 S232-11D2 ASSEMBLY FULLY LOADED LESS 36 RODS IN 3.6&7 SUBASSEMBLIES.
DETECTOR 1 AT LOCATION 5. 28SEPT.87. 22C.

GRRHO	=	-15.46
GOZRHO	=	-19.41
ALPHA	=	3.69725E+03
DELAYED NEUTRON LEVEL	=	4.69166E+00
BACKGROUND LEVEL	=	5.52081E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	17

TTC-5 S232-11D3 ASSEMBLY FULLY LOADED LESS 36 RODS INSUBASSEMBLIES 3,6&7.
 DETECTOR AT LOCATION 3. 28SEPT87. 22C.

TTC-5 S232-11D4 ASSEMBLY FULLY LOADED LESS 36 RODS IN SUBASSEMBLIES 3, 6&7
DETECTOR AT LOCATION 4. 28SEPT87. 22C.

TTC-5 S232-11D5 ASSEMBLY FULLY LOADED LESS 36 RODS IN SUBASSEMBLIES 3,6&7.
 DETECTOR 1 MOVED TO LOCATION 5.28SEPT87. 22C.

GRRHO	=	-24.81
GOZRHO	=	-49.94
ALPHA	=	3.44878E+03
DELAYED NEUTRON LEVEL	=	4.02652E-01
BACKGROUND LEVEL	=	4.33051E+00
CHANNEL WIDTH	=	5.02500E-05
START CHANNEL	=	20

F.23

TTC-5 S232-12D5 ASSEMBLY FULLY LOADED LESS 36 RODS IN SUBASSEMBLIES 3, 6 & 7.
 DETECTOR 1 AT LOCATION 5 & DETECTOR 3 AT LOCATION 6. 29SEPT87. 22C.

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