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PHYSICS VERIFICATION PROGRAM

PART III, TASK 9

— Quarterly Technical Report —

April-June 1973

by

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Physics Verification Program, Part III, Task 9
Quarterly Technical Report, April-June 1973

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ABSTRACT

An experimental program is in progress to measure the fission rate in UO_2 fuel at selected positions within a critical assembly and to compare it with the neutron absorption rate in the boron of a simulated lumped burnable poison material.

CONTENTS

	Page
1. INTRODUCTION.	1-1
2. DESCRIPTION OF ASSEMBLY	2-1
2.1. Facility	2-1
2.2. Fuel Rods	2-1
2.3. Poison Rods	2-1
2.3.1. Thick-Walled Poison Pins.	2-1
2.3.2. Thin-Walled Poison Pins	2-2
2.3.3. Sample Pins.	2-2
2.4. Core Description	2-2
2.5. Moderator Boron	2-2
3. THEORY.	3-1
4. SUMMARY	4-1
5. REFERENCES.	5-1
DISTRIBUTION	A-1

List of Figures

Figure

2-1. Poison Sample Pin.	2-4
2-2. Core XIII Loading Diagram	2-5

1. INTRODUCTION

In April of 1973, Babcock & Wilcox (B&W) began Part III, Task 9, of the Physics Verification Program, entitled Capture Measurements. The Program is jointly sponsored by B&W and the U.S. Atomic Energy Commission. The objective of the task is to measure the fission rate in UO_2 fuel at selected positions within a critical assembly, and to compare it with the neutron absorption rate in the boron of a simulated lumped burnable poison (LBP).

This report summarizes the work completed during the second calendar quarter of 1973, which has consisted principally of planning and preparing for Task 9. Completed phases of the program are summarized in references 1 through 8.

2. DESCRIPTION OF ASSEMBLY

2.1. Facility

The facility, located at B&W's Lynchburg Research Center, is a critical assembly within a 5-foot-diameter by 6.5-foot-high core tank with 1/2-inch-thick aluminum walls. The water-handling equipment, instrumentation, control system, and other auxiliaries are conventional.

2.2. Fuel Rods

The fuel rods are 0.475 inch in diameter by 61.59 inches long. The cladding is 0.032-inch-thick, type-6061 aluminum, and the UO_2 fuel is enriched to 2.46 wt % ^{235}U . The fuel is described further in reference 6.

A special fuel rod, which can be disassembled so that a sample fuel pellet can be removed, is also available. Sample fuel pellets irradiated in this rod will be used to determine local power densities accurately.

2.3. Poison Pins

Two sizes of poison pins have been prepared. The bottom end of each pin is sealed with a tight-fitting, 1/2-inch-thick aluminum plug sealed in place with epoxy, and each top is sealed with a removable cork plug. Powdered sodium tetrafluoroborate (NaBF_4) serves as the poison material and forms a column in each pin of sufficient length to span the full core height.

2.3.1. Thick-Walled Poison Pins

The thick-walled poison pins are 0.438-inch-OD by 0.120-inch-wall tubes of 6061 aluminum. Each pin contains $55.9 \pm 1.1\text{g}$ of powdered NaBF_4 which constitutes a column 174.0 cm in height.

2.3.2. Thin-Walled Poison Pins

The thin-walled poison pins are 0.438-inch-OD by 0.035-inch-wall tubes of 6061 aluminum. Each pin contains 195.3 ± 1.5 g of powdered NaBF_4 , which constitutes a column 174.0 cm in height.

2.3.3. Sample Pins

Special poison pins of both the thick- and thin-walled varieties have been prepared; these pins can be disassembled so that a sample of poison material can be easily removed. This type of pin is illustrated in Figure 2-1.

2.4. Core Description

Top and bottom grid plates align the fuel rods in the core tank. The top and bottom egg-crate grid plates are 1-inch-wide, slotted aluminum strips, which are interlocked to form a square matrix. The 0.160-inch-thick strips are spaced on centers of 0.644 ± 0.001 inch. The fuel rods are supported on a 2-inch-thick aluminum base plate, which is raised about 1 inch above the bottom of the 5-foot-diameter tank. The intervening space is filled with moderator. Moderator levels are measured from a reference plane at the bottom of the active fuel.

The six safety blades are 8-inch-wide boral. Each blade is thin enough to pass between rows of fuel rods, so that uniform lattice spacing can be maintained. During measurements, the safety blades are held in a cocked position to minimize their contribution to core perturbations. Criticality is achieved by raising the level of moderator. Boric acid in the moderator controls excess reactivity.

As shown in Figure 2-2, a loading diagram of the core, imaginary lines divide the core into nine elements. At the center position of each element is a 1/4-inch-OD by 0.035-inch-wall Inconel-600 tube, which is sealed at the bottom to exclude water. Type-304 stainless steel sleeves (0.475-inch OD) at the bottom and near the top of the tubes center the tubes in the grid positions and thus maintain proper pitch.

2.5. Moderator Boron

Excess reactivity will be controlled by dissolving boric acid (H_3BO_3) in the demineralized water moderator. The boron concentration, in parts

per million (ppm), is defined as the grams of natural boron per 10^6 cm³ of moderator at 25C. The boron concentration will be determined by titration against a standard KOH solution. The measured concentrations have an absolute standard deviation of ± 3 ppm.

Figure 2-1. Poison Sample Pin

Figure 2-2. Core XIII Loading Diagram

3. THEORY

The objective of this task is to measure the fission rate in UO_2 fuel at selected positions within the critical assembly and to compare it with the neutron absorption rate in a simulated LBP material placed in other selected locations. The method being applied to this work relates the ^{99}Mo fission product activity in a fuel sample to the fission rate, and the ^{24}Na activity in a simulated LBP material to the total neutron absorption rate in that material. Absolute counting is avoided by measuring under identical counting conditions the ^{99}Mo and ^{24}Na activity from samples of identical composition irradiated in a highly thermal neutron flux for which the ratio of fission rate to absorption rate is known.

Sodium tetrafluoroborate (NaBF_4) was chosen to simulate an LBP material for several reasons:

1. The sodium of the compound will activate and thus provide a quantitative measure of absorption rate.
2. The sodium and boron are uniformly dispersed throughout the material.
3. The absorption cross section curves for sodium and boron have a similar shape up to a neutron energy of several keV.

If we designate the fission rate in a specified fuel sample by F , then

$$F = K'M \frac{1 + \delta_{238}}{\left(1 + \frac{Y_{238}}{Y_{235}} \delta_{238}\right)} \quad 3-1)$$

where K' = a proportionality constant,
 M = relative saturated activity of ^{99}Mo in the irradiated fuel sample,
 Y_{235} = the ^{99}Mo yield from ^{235}U fission,
 Y_{238} = the ^{99}Mo yield from ^{238}U fission,
 δ_{238} = ^{238}U to ^{235}U fission ratio.

Fission rate is almost directly proportional to the production rate of the fission product ^{99}Mo . The slight deviation from direct proportionality is easily corrected for since approximate values for Y_{235} , Y_{238} , and δ_{238} are available:⁹⁻¹¹

$$\frac{Y_{238}}{Y_{235}} = 1.03 \pm 0.03$$

And in the experimental lattice,

$$\delta_{238} = 0.07 \pm 0.01 \text{ (ref 1).}$$

If we designate the ^{10}B neutron absorption rate in a specified sodium tetrafluoroborate sample by A , then

$$A = K''\theta S \quad (3-2)$$

where K'' = proportionality constant,
 S = relative saturated activity of ^{24}Na in the irradiated LBP sample,
 θ = calculated correction factor (approximately unity) which adjusts the results for the slight difference in absorption cross section shape of sodium and ^{10}B .

Combining equations 3-1 and 3-2,

$$\frac{F}{A} = \frac{KM (1 + \delta_{238})}{\theta S (1 + \frac{Y_{238}}{Y_{235}} \delta_{238})} \quad (3-3)$$

where K is an overall proportionality constant.

The constant K can be accurately evaluated from ^{99}Mo and ^{24}Na activities resulting when small, infinitely dilute samples of fuel and LBP are irradiated in a highly thermal neutron flux. For a highly thermal flux equation 3-3 reduces to

$$\left(\frac{F}{A}\right)_\phi = K \left(\frac{M}{S}\right)_\phi \quad (3-3a)$$

The subscript ϕ is used to designate parameters referring to or obtained from the thermal irradiations.

The thermal neutron reaction rates can be calculated from accurately known thermal cross sections. Thus,

$$\left(\frac{F}{A}\right)_\phi = \left(\frac{\bar{\Sigma}_{\text{fiss}}^{\text{U235}}}{\bar{\Sigma}_{\text{abs}}^{\text{B10}}} \right)_\phi \quad (3-4)$$

where $\bar{\Sigma}_{\text{fiss}}^{\text{U235}}$ = average macroscopic thermal cross section for U235 fission in the fuel sample,
 $\bar{\Sigma}_{\text{abs}}^{\text{B10}}$ = average macroscopic thermal cross section of B10 in the LBP samples.

Combining equations 3-3a and 3-4 gives

$$K = \left(\frac{\bar{\Sigma}_{\text{fiss}}^{\text{U235}}}{\bar{\Sigma}_{\text{abs}}^{\text{B10}}} \right)_\phi \left(\frac{S}{M} \right)_\phi \quad (3-5)$$

If equation 3-5 is substituted into equation 3-3 and the subscript R is used to designate the experimental lattice irradiations, then

$$\left(\frac{F}{A}\right)_R = \left[\frac{(1 + \delta_{238}) \bar{\Sigma}_{\text{fiss}}^{\text{U235}}}{\theta (1 + \frac{Y_{238}}{Y_{235}} \delta_{238}) \bar{\Sigma}_{\text{abs}}^{\text{B10}}} \right]_\phi \frac{(M/S)_R}{(M/S)_\phi} \quad (3-6)$$

The last term in equation 3-6 will be the experimentally measured quantity; the other terms will be evaluated theoretically or obtained from the literature.

4. SUMMARY

Work during the quarter has consisted of planning and preparing for experimental Task 9. Experimental measurements will be conducted during the third quarter.

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