Journal of the Korean Nuclear Society
Volume 10, Number 2, June, 1978

《**Technical Report**》

Multigroup Calculations for TRIGA-type Reactor Analysis

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(Received June 3, 1978)

Abstract

Multigroup constant calculation system for TRIGA-type reactor analysis was provided. Calculations for initial criticality, temperature coefficient, flux and power distributions of TRIGA-Mark III reactor were carried out by using diffusion code CITATION. And some of results were compared with the values of start-up experiments and design values. It could be confirmed that the prepared computation system is very useful for TRIGA-type reactor analysis.

요 약

TRIGA형 원자로를 해석하기 위한 다군 군정수 계산 조직을 마련하고 확산코드 CITATION에 의한 2차원 계산을 통해 TRIGA-Mark III 원자로의 초기임계치, 온도계수, 중성자속 및 출력분포 등을 계산하였다. 일부 계산결과를 임계실험 및 설계치와 비교 검토하였다. 마련된 계산체계가 만족한 결과를 얻을 수 있음이 입증되었다.

I. Introduction

One of the basic characteristics of TRIGA-type research reactors is the use of homogeneous U-Zr hydride solid fuel-moderator elements with a large prompt negative temperature coefficient of reactivity. The zirconium hydride pocesses a basic neutron spectrum-hardening mechanism. And additional advantage includes the fact that the zirconium hydride has a good heat capacity. Because of these fuel mechanisms, it is well-known[1] that few group calculations such as one thermal group are not suitable for accurate analysis of TRIGA-type reactor core. To obtain reasonable values of temperature coefficients and other parameters of TRIGA-type reactor, multigroup treatment is necessary. Four thermal groups are commonly used for these[1,2].

The purposes of this paper are to provide neutronics calculation system for TRIGA-type reactors and confirm the accuracy. For these purposes, a computer code system was prepared for producing scattering kernels of the neutrons with the chemically bound moderator atoms, multigroup constants of the spatially dependent spectrum averaged thermal group and the other related group constants. And several library data of thermal range were prepared with them.

The code system was checked for its applicability by comparing the calculated result of

TRIGA-Mark III reactor parameters with the design values[3].

II. Calculation System for Multigroup Constants

1. Scattering Kernel

Scattering kernels are used to describe the interactions of the neutrons with the chemically bound moderator atoms. The bound hydrogen kernels for hydrogen in water were generated by GAKER code[4], while those for hydrogen in zirconium hydride were generated by UNCLE code[5] (equivalent to SUMMIT code[6]). And all other scattering transfers within the cell were based on a free gas model scattering kernel. These data were stored on library tape to calculate thermal group constants.

- **H_2O Model**

A scattering kernel for protons bound in water has been formulated by Nelkin[7]. The motion of hydrogen atoms in water is considered in terms of the H_2O molecule at the basic dynamical unit. The physical model which GAKER code used is that of a translator of mass 18, a hindered rotational oscillator of mass 2.32 and energy 0.06 eV, a vibrational oscillator of mass 5.84 and energy 0.205 eV, and a vibrational oscillator of mass 2.92 and energy 0.48 eV.

- **ZrH Model**

The scattering of slow neutrons by zirconium hydride can be described by a model in which the hydrogen atom motion is treated as a isotropic harmonic oscillator with energy transfer quantized in multiples of a value in the range of 0.13~0.145 eV[8]. More precisely, UNCLE code uses a frequency spectrum with two branches, one for the optical modes for energy transfer with the bound proton, and the other for the acoustical modes for energy transfer with the lattice as a whole. In the present work, the optical modes were represented as a broad frequency band centered at 0.13 eV and with the width 0.03 eV. And the low frequency acoustical modes were assumed to have a Debye spectrum with a cut-off of 0.02 eV and a weight determined by an effective mass of 360.

2. Thermal Group Constants

In the processes of TRIGA-Mark III design[3], thermal group constants were obtained from GATHER section of GGC-3 code[9]. GATHER code computes the thermal spectrum for the homogenized (space-independent) cell by solving the B-1 thermalization equation and produces spectrum weighted broad group averaged cross sections. But according to the recent results of TRIGA-type reactor core analysis, it is known[1] that heterogeneous cell calculations are more accurate than homogeneous cell calculations. Recently, thermal group constants of the fuel region are generated by the multigroup cross section code which computes the spatially dependent thermal spectra at each mesh point in the cell by solving transport equation[2]. Therefore, thermal group constants of the fuel region were generated by the multigroup cross section code THERMOS-MUG[10], which is a modified version of the ANL-revised THERMOS[11] code. The THERMOS-MUG code can prepare scattering matrix of neutrons in the thermal energy range. The THERMOS code computes the spatially dependent thermal spectra at each mesh points in the cell by solving the integral transport equation numerically. Cell averaged multigroup constants were obtained to both homogeneous and heterogeneous systems by averaging the fine group cross sections over the space dependent spectrum. On the other hand, thermal group cons-

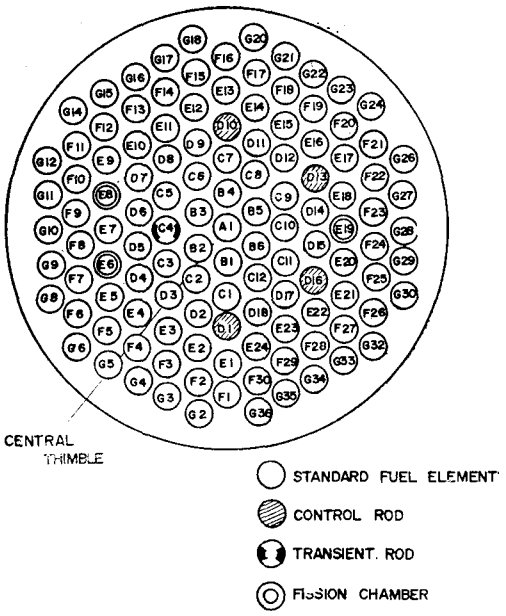

Fig. 1. Horizontal Core Diagram of TRIGA-Mark III Reactor.

tants of other regions were generated by GATHER section of GGC-4 code[12], which is an updated version of the GGC-3 code. The energy range of GGC-4 basic nuclear data is divided into a number of fast groups (up to 99) and thermal groups (up to 101).

3. Fast Group Constants

All fast group constants (>1.125 eV) were generated by the GAM section of GGC-4 code, where fine group (99 group) cross sections are averaged over a spatially independent flux derived by solution of the B-1 equations for each discrete reactor region composition.

III. Calculations

Horizontal core of TRIGA-Mark III reactor consists of hexagonal geometry as shown in Fig. 1. The active section of a standard fuel -moderator element in the core is 38 cm (15 in.) long and 3.63 cm (1.43 in.) in diameter and contains 8.5 w/o uranium enriched to 20% in U-235. The hydrogen to zirconium atom ratio is approximately 1.6 (in the face-centered cubic delta phase). To facilitate hydriding, a small hole (~0.25 in. in diameter) is drilled through the center of the active fuel moderator section and a zirconium rod is inserted in this hole after hydriding is complete. Each element is clad with a 0.051 cm (0.02 in.) thick stainless steel tube. The control rods are sealed stainless steel tubes approximately 109 cm (43 in.) long and by 3.43 cm (1.35 in.) in diameter in which the uppermost 16.5 cm (6-1/2 in.) section is an air void and the next 38 cm (15 in.) is the neutron absorber (borated graphite). Immediately below the neutron absorber is a fuel follower section consisting of 38 cm (15 in.) of U-$ZrH_{1.6}$ fuel. The safety-transient rod is a sealed, 93.35 cm (36.7 in.) long by 3.18 cm (1-1/4 in.) diameter aluminum tube containing a solid rod of borated graphite as a neutron absorber. Below the absorber is an air-filled follower section.

In the processes of TRIGA-Mark III start-up experiments, the first cold-clean critical core consists of 68 fuel elements including 4 fuel follower elements. The standard fuel-moderator elements contain 38 g U-235 per each element. But 8 fuel moderator elements of 39 g U-235 were used, which positioned all of B ring and C9 and C3 of C ring as shown in Fig. 1. On account of all control rod out, control rod regions are substituted by control follower regions. The control follower elements also contain 32 g U-235 per each element. And the transient rod region and fission chamber region is substituted by the air-filled region. The initial operating core consists of 100 fuel elements including 4 fuel follower elements. The fuel cell model in the core is shown in Fig. 2.

Group constants for the homogeneous cells were generated by the GGC-4 code. And

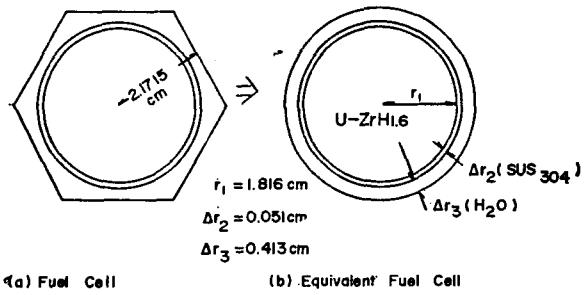

Fig. 2. Fuel Cell Configuration of TRIGA-Mark III Reactor

thermal group constants for the fuel cell and the control rod follower cell were generated by the THERMOS-MUG code for the heterogenseou cells. The broad group structures are four for thermal groups and five for fast groups as shown in Table 1. And the thermal neutron cut-off energy is 1.125 eV.

Fig. 3 shows the hexagonal geometry model to analyse the TRIGA-Mark III reactor core for the first cold-clean critical core and the initial operating core.

Calculations for criticality, temperature coefficient, flux and power distributions etc. were carried out by the two dimensional diffusion code CITATION. The axial (vertical) buckling required for the two dimensional calculations was obtained from the following equation

Table 1. Broad group energy structure

Energy Regime	Group Number	Energy Range
Fast Group	1	15.0MeV~3.0MeV
	2	3.0MeV~1.5MeV
	3	1.5MeV~0.6MeV
	4	0.6MeV~9.12KeV
	5	4.12KeV~1.125 eV
Thermal Group	6	1.125 eV~0.42 eV
	7	0.42 eV~0.14 eV
	8	0.14 eV~0.05 eV
	9	0.05 eV~0.0 eV

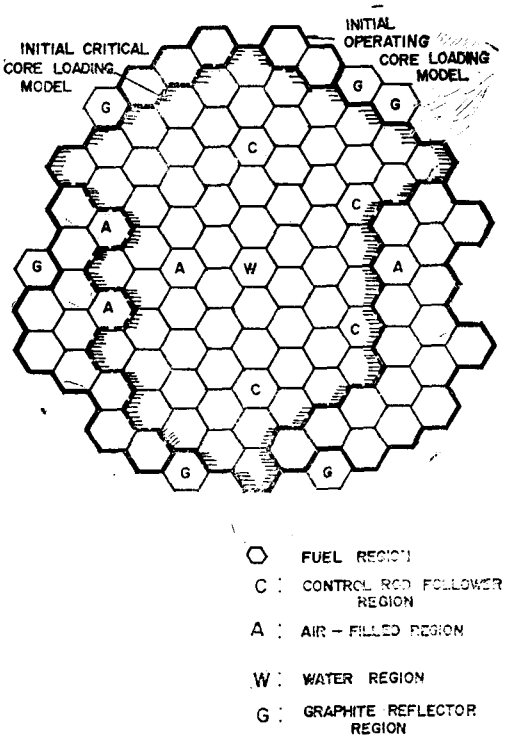

Fig. 3. Hexagonal Models of the Initial Critical and Operating Cores (~30 cm thick water reflector outside cores respectively)

$$B_L^2=\left[\frac{\pi}{H+2\delta}\right]^2 \quad (1)$$

where

B_L^2: axial buckling

H: height of the active fuel rod

δ: reflector saving

From equation (1), the axial buckling is 0.0034022. Fig. 4 briefly shows the procedure of above subsequent calculations.

IV. Results and Discussion

◦ Criticality

In the processes of TRIGA-Mark III start-up experiments, the core wsa reached to the criticality with small excess reactivity (14.9 cent) when 64th fuel moderator element was inserted and all control rods were out. The effective multiplication factor k_{eff} of the first

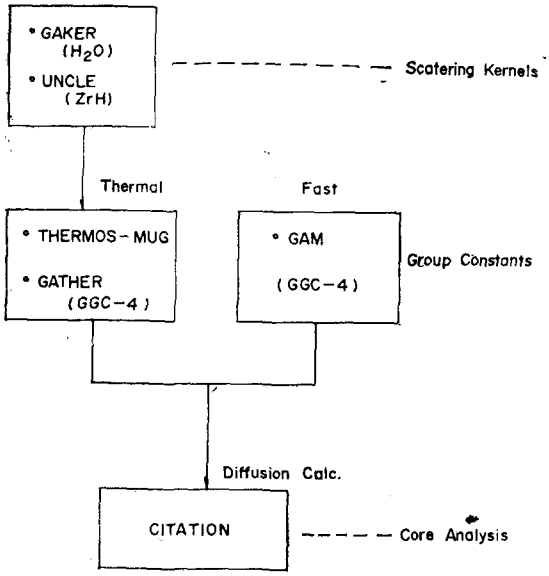

Fig. 4. Calculational Scheme

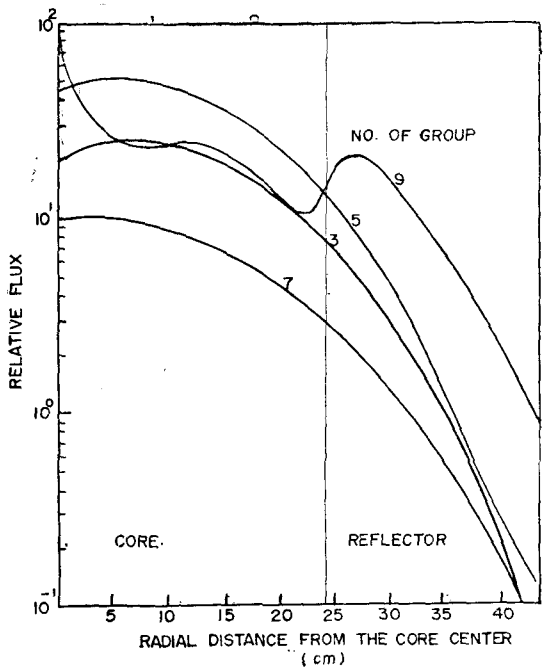

Fig. 5. Radial Flux Distributions of the Initial Operating Core.

cold-clean critical core was calculated with reflector of about 30 cm thick and at 20°C for water temperature. The calculated value was 0.9991.

◦ **Temperature Coefficient**

In order to calculate temperature coefficient, seperate cross section sets were generated for each fuel-moderator element temperature by use of temperature dependent hydride kernels and Doppler broadening of the U-238 resonance integral to reflect the proper temperature. The temperature coefficient was determined numerically by calculating the change in reactivity associated with a uniform heating of the fuel-moderator element while the core water and reflector materials were remained at room temperature. The calculated result is about $1.079\times10^{-5}\frac{\delta k}{k}/°C$ averaged over a fuel moderator temperature range from 23°C to 700°C. According to the Safeguards Analysis Report[3] for the TRIGA-Mark III reactor, the temperature coefficient is $\sim1.0\times10^{-5}\frac{\delta k}{k}/°C$.

◦ **Execss Reactivity**

The effective multiplication factor k_{eff} of the initial operating core was calculated to see the excess reactivity with reflector of about 30cm thick and at 20°C for water temperature. The calculated value was 1.068. Therefore the excess reactivity of the initial operating core is about 0.064 $\Delta k/k$. Comparing with the design value 0.063 $\Delta k/k$, the difference is about 1.6% at most.

◦ **Flux and Power Distributions**

Flux distributions of the initial operating core as a function of radial distance are shown in Fig. 5. And power distribution normalized to the average power is shown in Fig. 6. As shown in Fig. 6, the radial peak to average power generation ratio is 1.61. Comparing with the design value 1.60, the difference is about 0.6% at most.

In the critical experiment, the criticality of the core considered the excess reactivity, 14.9 cent, was 1.001043. Considering above excess

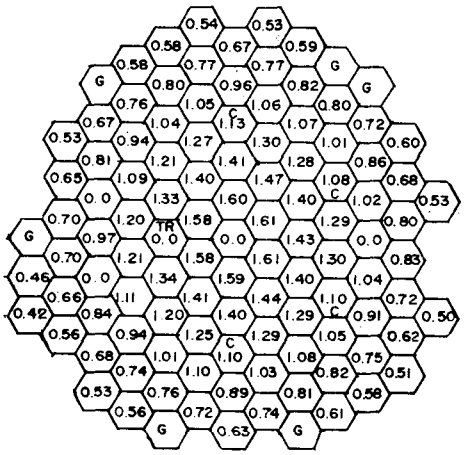

Fig. 6. Power Distribution of the Initial Operating Core

reactivity, the difference between the calculated value and the above is 0.194% at most. The discripancies of the calculated excess reactivity and the radial peak to average power generation ratio from the design values are less than 1.6% and 0.6% respectively. It is thought that the overestimated value could be due to the 8 fuel-moderator elements of 39 g U-235 and the neutron absorption effects of the fission chambers which were not considered in the calculations. Because of the lack of detail experimental data such as fiux or power distributions, it is difficult to comment on the calculated flux or power distributions. The temperature coefficient was calculated from wide temperature range. It is therefore desirable to calculate fine temperature dependent values as a next step.

The differences of TRIGA-Mark III reactor parameters calculated from our computation system are very small comparing with the results of critical experiments and the design parameters in the safeguards analysis report.

It could be confirmed that, based on the considerations discussed above, the prepared computation system consisting of the multigroup constant codes is very useful and gives accurate results for TRIGA-type reactor analysis.

Acknowledgement

The authers would like to express our thanks to Mr. Takeharu Ise of Japan Atomic Energy Research Institute for his helpful assistance in providing the multigroup constant codes.

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