UNCERTAINTIES IN CALCULATIONS OF NUCLEAR DESIGN CODE SYSTEM FOR THE HIGH TEMPERATURE ENGINEERING TEST REACTOR (HTTR)

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Abstract

Japan Atomic Energy Research Institute(JAERI) has extensively done the research and development to construct the High Temperature Engineering Test Reactor(HTTR) of 30MW(t). The main objectives of the HTTR are to establish basic technologies for advanced HTGR in future and to be served as an irradiation test reactor in order to conduct researches in innovative high-temperature technologies. The HTTR is a graphite moderated and helium gas cooled reactor with prismatic fuel blocks. The inlet and outlet coolant temperatures are 395°C and 950°C, respectively. The reactor employs pin-in-block type fuel assemblies which contain low enriched UO_2.

The nuclear design code system for the HTTR consists of the computer codes DELIGHT, SRAC, TWOTRAN-2 and CITATION-1000VP. DELIGHT and SRAC are one dimensional cell burnup codes which have been developed in JAERI. TWOTRAN-2 is a transport code which is used to provide the average group constants of the graphite blocks where the control rods are inserted. CITATION-1000VP is a reactor core analysis code which has been improved from CITATION for the increase of memory and vectorization.

The nuclear characteristics of the HTTR such as shut down margin and power density distribution should satisfy safety related design criteria. In order to assure the sufficiency to the criteria, the uncertainty of the calculation was investigated.

For the uncertainty evaluation, the comparison of the calculations with the experiments was performed with a graphite moderated critical assembly, Very High Temperature Reactor Critical Assembly(VHTRC), whose characteristics are similar to those of the HTTR. Among the various experiments performed at VHTRC, the following integral quantities were used to confirm the uncertainty; (1) effective multiplication factor, (2) neutron flux distribution, (3) control rod reactivity worth and (4) burnable poison rod reactivity worth. It was confirmed that the discrepancies between calculations and experiments were small enough to be allowable in the nuclear design of HTTR.

1 Introduction

Japan Atomic Energy Research Institute (JAERI) has extensively done the research and development to construct the High Temperature Engineering Test Reactor(HTTR) of 30MW(t)[1]. The bird's-eye view of the HTTR is shown in Fig.1.

The main objectives of the HTTR are to establish basic technologies for advanced HTGRs in future and to be served as an irradiation test reactor in order to conduct researches in innovative high-temperature technologies.

The HTTR is a graphite moderated and helium gas cooled reactor with prismatic fuel blocks. The active core,290cm in height and 230cm in effective diameter, consists of 30 fuel columns and 7 control rod guide columns(Fig.2). The inlet and highest outlet coolant temperatures are 395°C and 950°C,respectively. The reactor employs pin-in-block type fuel assemblies which contain low enriched UO_2. Also, burnable poison(BP) rods are distributed in the core.

The nuclear design code system for the HTTR consists of the computer codes DELIGHT[2], SRAC[3], TWOTRAN-2[4] and CITATION-1000VP[5]. The program structure of nuclear design code system for HTTR is shown in Fig.3. DELIGHT is cell burnup code which has been developed in JAERI and is used to provide few-group constants of fuel blocks, graphite blocks and so forth for succeeding core calculation. TWOTRAN-2 is a transport code which is used to provide the average group constants of the graphite

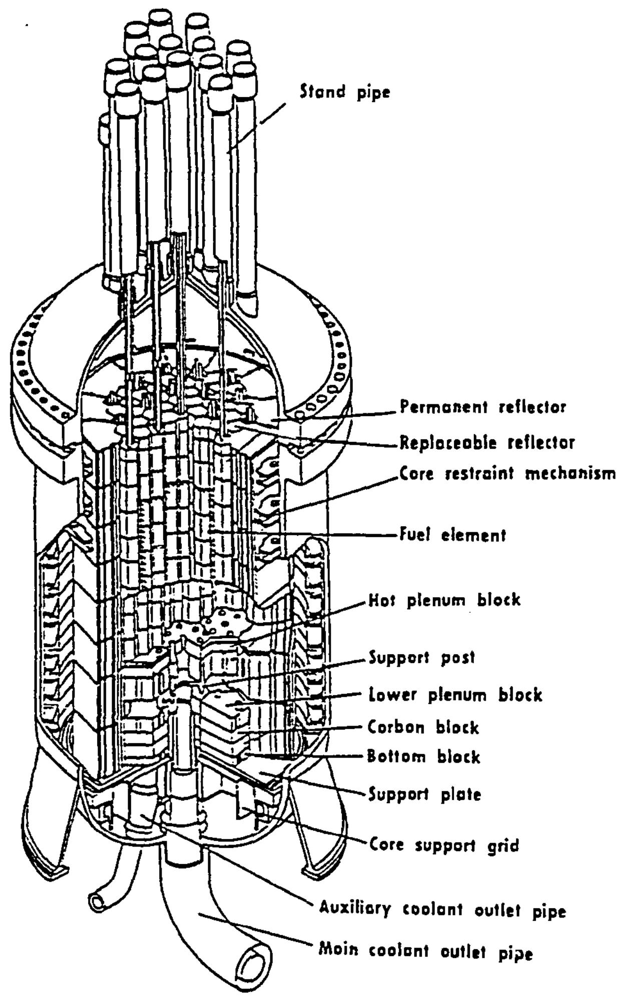

Fig.1 Bird's-eye view of the reactor vessel and core.

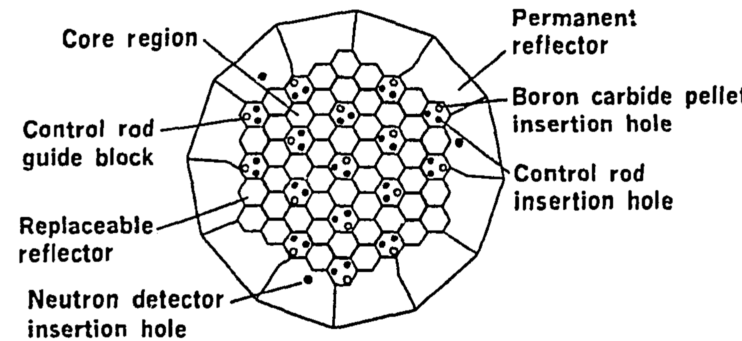

Fig.2 Cross sectional view of irradiation regions.

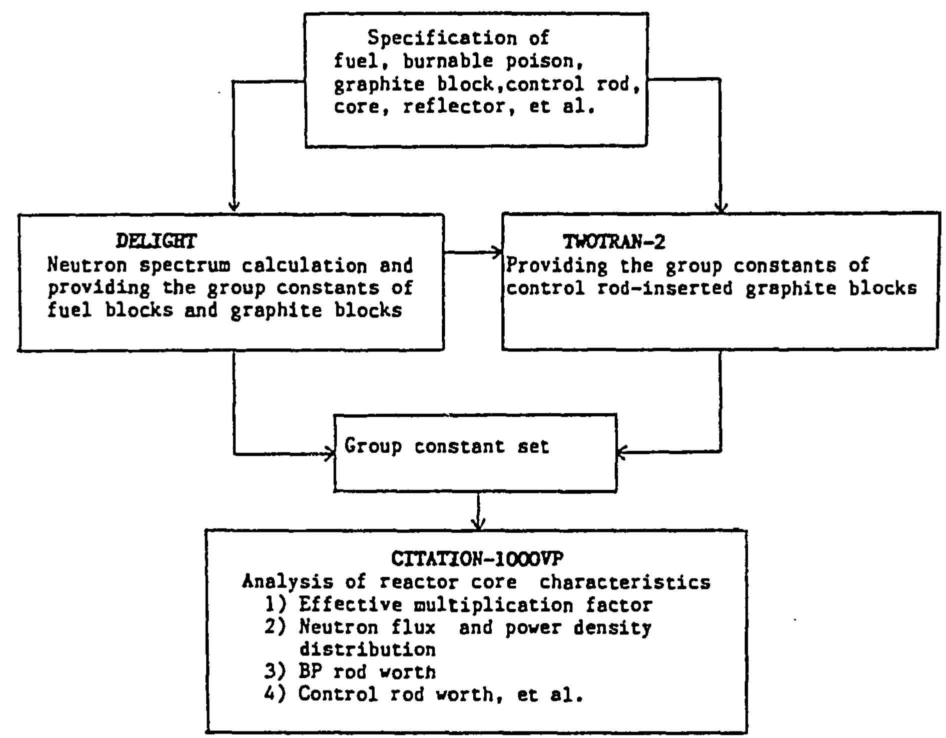

Fig.3 Program structure of nuclear design code system for HTTR.

blocks where the control rods are inserted. CITATION-1000VP is a reactor core analysis code which has been improved from CITATION[6] to be able to analyze a full core of HTTR. The nuclear characteristics of the HTTR such as shutdown margin and power density distribution should satisfy safety related design criteria. In order to assure the sufficiency to the criteria, the uncertainties in calculations of nuclear design code system for HTTR were investigated.

For the uncertainty evaluation, the comparison of the calculations with the experiments was performed on a graphite moderated critical assembly, Very High Temperature Reactor Critical Assembly(VHTRC)[7,8], whose characteristics are similar to those of the HTTR. Among the various experiments performed at VHTRC, the following integral quantities were used to confirm the uncertainty :(1) Effective multiplication factor,(2) Neutron flux distribution,(3) Control rod reactivity worth and (4) Burnable poison (BP) rod reactivity worth.

2 Accuracy requirements in safety design

The requirements for nuclear design in the safety related design are arisen from the confirmation of shutdown margin (effective multiplication factor,reactivity worths of control rod and burnable poison),the reduction of maximum fuel temperature (adjustment of power distribution),the preparation of the calculational conditions (addition rate of control rod reactivity worth and temperature coefficients) for safety analyses and so on.

3 Nuclear design code system

The nuclear design code system for the HTTR consists of DELIGHT,TWOTRAN-2 and CITATION-1000VP codes.

DELIGHT calculates the multi-group neutron spectrum of a fuel cell containing coated fuel particles and provides the group constants for core calculation. The calculations incorporated in this code are calculations of resonance, neutron spectrum, fuel cell, BP cell, criticality and burnup. The nuclear data is based on the ENDF/B-4 mainly. In the resonance range, the code is able to consider the effect of double heterogeneity caused by coated fuel particles and assembled fuel rods. The neutron spectrum is obtained by cylindrical model using the collision probability method. The spectrum calculation is performed with 61 groups in the fast energy range from 2.38eV to 10MeV and with 50 groups in the thermal energy range from 0.0 to 2.38eV. The collision probability method is applied for the fuel and BP cell calculations.

TWOTRAN-2 is a two-dimensional neutron transport code which is employed to evaluate the shielding factors of control rods and to provide average group constants of a graphite block inserted a pair of control rods.

CITATION-1000VP is a reactor core analysis code which has been developed so as to enable the multi-neutron group calculation of the HTTR in the three dimensional full core model through extending number of zones and meshes of CITATION code and enhancing a calculation speed with the vectorization.

4 Very High Temperature Reactor Critical Assembly(VHTRC)

The VHTRC is a split table type critical assembly which consists of two halves-fixed and movable ones. The core of VHTRC is shaped in a hexagonal prism, whose side and axial length are 175cm and 240cm, respectively. Fuel blocks and reflector blocks are piled in a hexagonal iron frame to form the core and the reflector regions respectively. Core configuration could be flexibly changed by loading the fuel or graphite rods. Instead of the fuel and graphite rods, BP rods can be loaded into graphite block to measure a BP rod reactivity worth.

5 Analysis of experimental data of VHTRC

For the uncertainties of the nuclear design code system, the experimental data of the initial core (VHTRC-1) was analyzed. The VHTRC-1 core is made by loading about 280 fuel rods which contain fuel compacts of 4wt% enriched uranium coated particles. The cross section of VHTRC-1 core is shown in Fig.4.

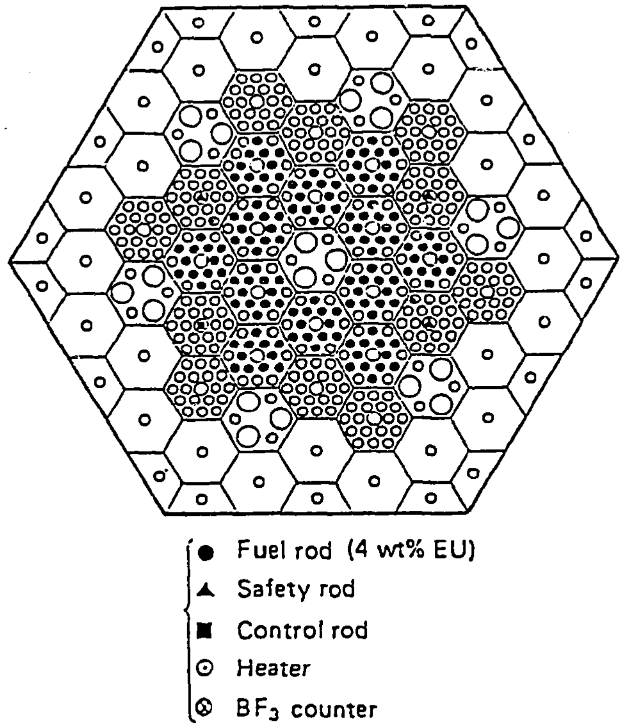

Fig.4 Cross section of VHTRC-1[7].

For the calculation of group constants of fuel with DELIGHT, one-dimensional cylindrical model is used. For the calculation of averaged group constants of a pair of control rods and a graphite block, TWOTRAN-2 with a two-dimensional X-Y model is used.

Core calculation performed by CITATION-1000VP employs the three-dimensional triangular-mesh model with triangular-mesh division in a fuel block.

(1) Effective multiplication factor (k_{eff})

The comparison between calculations and experiments was shown in Fig.5. The maximum discrepancy is about 1% Δk.

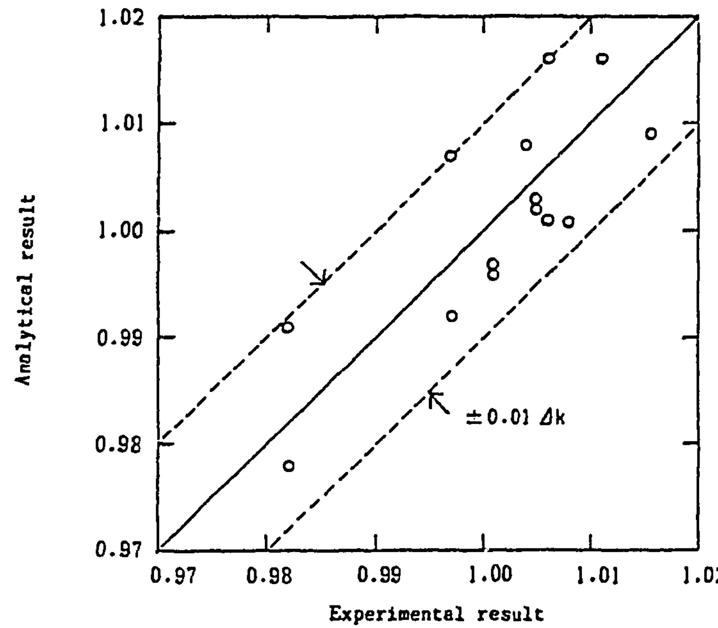

Fig.5 Comparison of the multiplication factors obtained by VHTRC experiment with the analytical results.

(2) Neutron flux distribution

The measurement of the neutron flux distribution is analyzed to grasp the uncertainty for calculation of power distribution in the HTTR. The analysis is performed for the VHTRC-1 with natural Cu foils and pins as irradiated material.

The comparison between the analytical and experimental radial reaction rate distributions is shown in Fig.6. A quite good agreement is obtained in the comparison. The maximum uncertainty between experimental and analytical values is about 2.9%.

(3) Burnable poison(BP) rod reactivity worth

The burnable poison rod reactivity worth is analyzed on the core configuration representing one poison rod inserted along core axis.

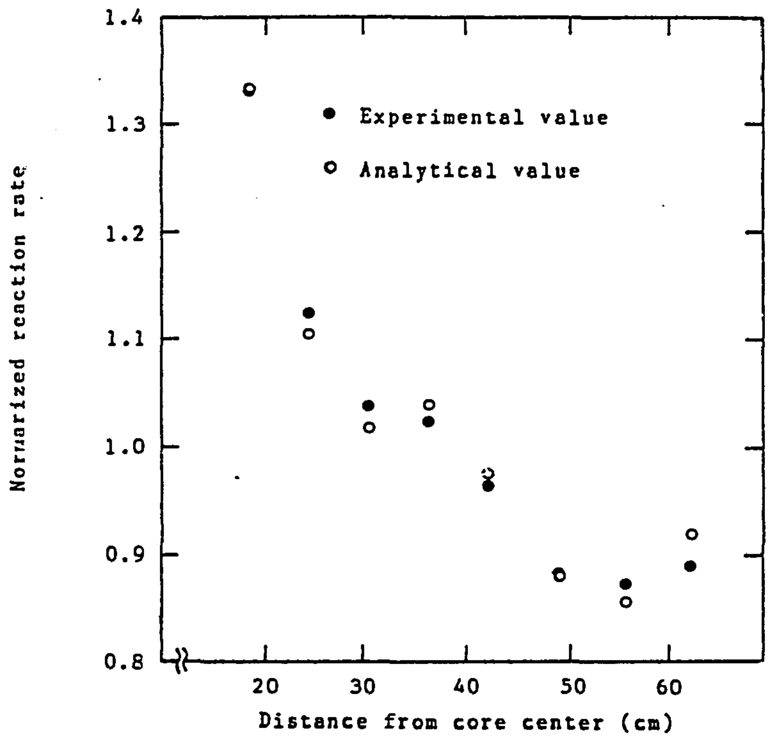

Fig.6 Comparison between the analytical and experimental radial reaction rate distributions.

The result of the analysis is compared with experimental data. A quite good agreement was obtained. The discrepancy is nearly 0%.

(4) Control rod reactivity worth

The measurement of control rod reactivity worth has been performed for the core with a pair of control rods inserted in core central axis.

The analytical result agrees with experimental result within about 2.6%.

6 Conclusion

The analyses have been performed for the experiments of VHTRC-1 core on effective multiplication factor, neutron flux distribution, BP rod reactivity worth and control rod reactivity worth.

The uncertainties between the experiments and calculations are 1% Δk, 2.9%, nearly 0% and 2.6%, respectively.

On the other hand, the targets for the uncertainties of the nuclear design items are as follows;

1) Effective multiplication factor (room temperature)	$\lesssim$ 1% Δk
2) Reactivity worth of control rods	$\lesssim$10%
3) Reactivity worth of burnable poisons	$\lesssim$10%
4) Radial power distribution	$\lesssim$ 3%
5) Temperature coefficient	$\lesssim$20%

It was confirmed that the discrepancies for the nuclear design code system were small enough to be allowable in the nuclear design for HTTR.

ACKNOWLEDGEMENTS

The authors wish to thank Messrs. S. Saito, T. Tanaka, Y. Sudo, H. Yasuda, F. Akino and T. Yamane for useful comments and supports.

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AUTOMATED DIFFERENTIATION OF COMPUTER MODELS FOR SENSITIVITY ANALYSIS

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Abstract

Sensitivity analysis of reactor physics computer models is an established discipline after more than twenty years of active development of generalized perturbations theory based on direct and adjoint methods. Many reactor physics models have been enhanced to solve for sensitivities of model results to model data. The calculated sensitivities are usually normalized first derivatives, although some codes are capable of solving for higher-order sensitivities. The purpose of this paper is to report on the development and application of the GRESS system for automating the implementation of the direct and adjoint techniques into existing FORTRAN computer codes. The GRESS system was developed at ORNL to eliminate the costly man-power intensive effort required to implement the direct and adjoint techniques into already-existing FORTRAN codes. GRESS has been successfully tested for a number of codes over a wide range of applications and presently operates on VAX machines under both VMS and UNIX operating systems.

I. INTRODUCTION

Sensitivity analysis is an important component of any computer code application for modeling physical systems. The role of sensitivity analysis is to provide a quantitative measure of the effect of computer code data and inputs upon key performance indices. Sensitivity analysis also helps limit the scope of the more complicated problem of quantifying uncertainties.

Sensitivity analysis of computer-generated results consists of determining the effect of model data upon the calculated results of interest. Because computer model equations can be differentiated analytically, sensitivities can be precisely defined and calculated in a deterministic fashion using both direct and adjoint methods.[1-8] The deterministic approach is particularly suited to large-scale problems for which direct perturbation of the model data becomes impractical from a cost standpoint. The main drawback to the deterministic approach has been the initial manpower investment to add the computational capability for calculating the necessary derivatives into existing computer models.

This paper presents the theory and application of the Gradient-Enhanced Software System, GRESS,[9] and its role in calculating model derivatives and sensitivities without a prohibitive initial manpower investment. Storage and computational requirements are discussed.