STUDY ON NEUTRONIC CHARACTERISTICS OF A JAPANESE 900MWe PWR CORE

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ABSTRACT: Neutronic characteristics of a Japanese PWR 900 MWe were investigated by using SRAC code and nuclear data library JENDL-3.3 with 107 public energy groups. The lattice modules, PIJ and CITATION, have been used for modeling the fuel rods, fuel assemblies and full core. The main neutronic characteristics analyzed in this work include infinite multiplication factors (k_{inf}) versus burnup, the distribution of nuclide concentrations in the pin cells; the pin-wise power distribution in the assemblies; the effective multiplication factor (k_{eff}), and the power distribution in the core.

Keywords: *SRAC, PIJ, CITATION, effective multiplication factor, power distribution, burnup.*

I. INTRODUCTION

Developing resources for nuclear reactor technical researching is one of the most important missions in the Vietnamese nuclear electrical project. For the founding and development of Nuclear Power Center, the staffs have studied the physical characteristics and the technical systems of the Nuclear Power Plants (NPPs) which can be constructed in Vietnam. The Pressurized Water Reactor (PWR) is one choice for the second NPP in Vietnam, for this reason, study and calculating neutronic characteristics of the Japanese 900 MWe reactor core are necessary.

This report proposes cells, assemblies and full-core calculations that will help the young staffs in Nuclear Power Center improve their calculating skills, the objectives are: Improving capabilities of using existing computer codes in Nuclear Power Center; training the young staffs of Nuclear Power Center; learning about physical and technical characteristics of the Japanese reactor; learning about the basic nuclear reactor physical knowledge and using them for neutronics calculations.

For the reasons mentioned above, the authors chose latest design of Mitsubishi's 900MWe reactor as an calculating object.

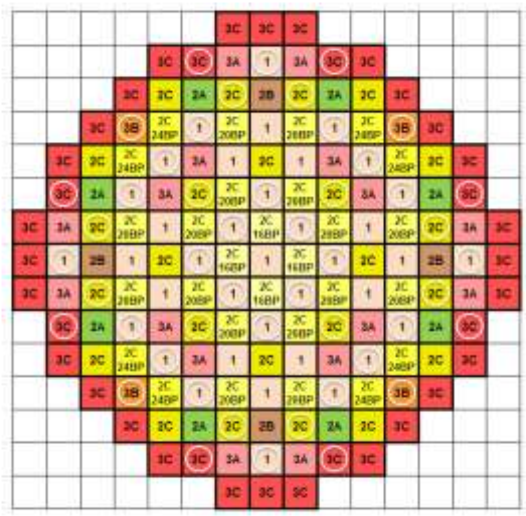

Figure 1: The PWR 900MWe core.

The 900MWe PWR has 3 types of fuel enrichments: 1.6%, 3.5%, 4.4%; 2 types of fuel enrichments 3.5% and 4.4% mixed with 8% Gadolinium (UO_2-Gd_2O_3). There are 157 fuel assemblies in the core, their arrangement is shown in Figure 1. Each assembly has 264 fuel rods, burnable absorber and control rods arranged among fuel rods to equalize power distribution in the core, the number of rods in the reactor is shown in Table 1.

Positions of control rods and UO_2-Gd_2O_3 rods are shown in Figure 2. Burnable absorber (Burnable poison - BP) rods have the same positions as control rods'. In Figure 2b, positions numbered 1 and 2 will be blocked when fuel assembly has 16 burnable absorber rods, positions numbered 1 will be blocked when fuel assembly has 20 burnable absorber rods.

For this calculation, the authors used a Japanese neutronic computer code - SRAC, this code was developed by Japanese Atomic Energy Agency (JAEA). SRAC is a powerful code which can calculate neutronic characteristics of various reactors. SRAC has 5 modules: PIJ, ANISN, TWOTRAN, TUD and CITATION0. In this report, the authors used 2 modules PIJ and CITATION, and nuclear data library JENDL 3.30. The PIJ module is based on 1,2-dimensions collision probabilistic method, can homogenize fuel pins and assemblies, this can be used for cell calculations. The CITATION module is based on multi-dimensions diffusion theory, this can model the reactor core. CITATION divides the core into homogeneous cells, then diffusion equation will be solved in each cell with the data carried out from PIJ calculations.

Table 1: Types of fuel assemblies in PWR 900Mwe.

Types	3A	3B	3C	3C	2C	2C	2C 24BP	2C 20BP	2C 16BP	2B	2A	1	1
Enrichment (%)	4.4				3.5							1.6	
BP rods							24	20	16				
UO_2-Gd_2O_3 rods	24	16								16	24		
With control rods			X			X							X

The main neutronic characteristics are performed in this report: the infinite neutron multiplication factors (k_{inf}) versus burnup, the distribution of nuclear density in fuel rods; the fission rates in each assembly, the effective neutron multiplication factor (k_{eff}) and power distribution in reactor core. Neutronics calculations are necessary because it shows us the characteristics of the fuel rods, assemblies and reactor design, the results can also be used for thermal hydraulics calculations.

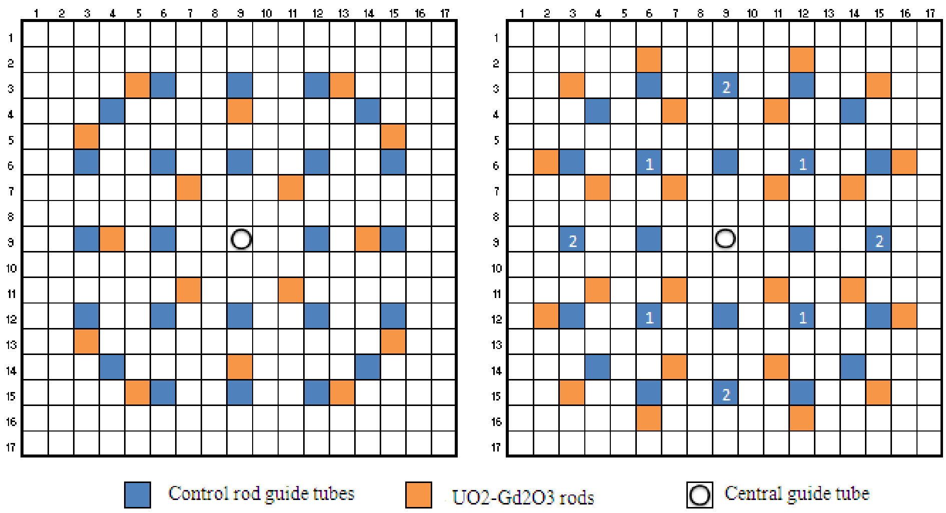

Figure 2: a-Left) Fuel assembly with 16 UO_2-Gd_2O_3 rods, b-Right) Fuel assembly with 24 UO_2-Gd_2O_3 rods.

II. RESULTS AND DISCUSSIONS.

II.1. Results of cell calculations using PIJ

The infinite neutron multiplication factor k_{inf} of UO_2 and UO_2-Gd_2O_3 rods versus burnup are shown in *Figure 3*. The k_{inf} of UO_2 rods decreases dramatically at the first burnup steps because of the contributions of ^{135}Xe and ^{149}Sm. From burnup step 300MWd/t, concentrations of poisons are equibrilium, k_{inf} begins decrease steadily per each burnup step. The k_{inf} of UO_2-Gd_2O_3 rods at the first burnup steps is low because Gadolinium is a strongly thermal neutron absorber. At later steps, the amount of Gadolinium decreases, the k_{inf} increases, from burnup step 30GWd/t, k_{inf} of UO_2-Gd_2O_3 rods decreases as UO_2 fuel rods'.

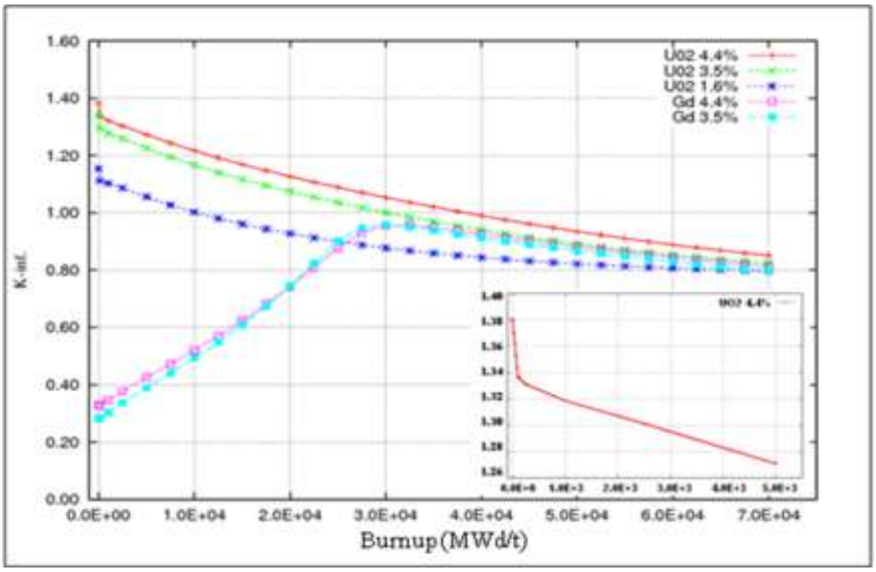

Figure 3: The k_{inf} of fuel rod as distribution of burnup.

The neutron flux of fuel rods in energy range 1.86eV - 10KeV is lower than the neighbouring range because neutron is absorbed in resonant range of ^{235}U. Because of the strongly thermal neutron absorbed capability of Gadolium, neutron spectrum of the UO_2-Gd_2O_3 rods was rougher than UO_2 rods', there is no thermal neutron peak as UO_2 rods. These are shown in Figure 4.

In Figure 5, the densities depended on radius of some isotopes at burnup step 50GWd/t of the 4.4% enrichment fuel rods are shown. The density of ^{235}U decreases from the centre to the cladding; the density of ^{238}U decreases rapidly at the cladding. A large quantity of neutron concentrates at the border of fuel and moderator, because of that, the reaction rate at the border is larger, huge amount of the fission products concentrate at the fuel cladding.

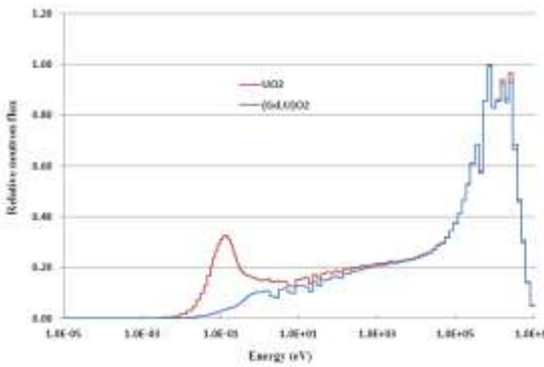

Figure 4: The neutron flux in fuel rod as distribution of energy.

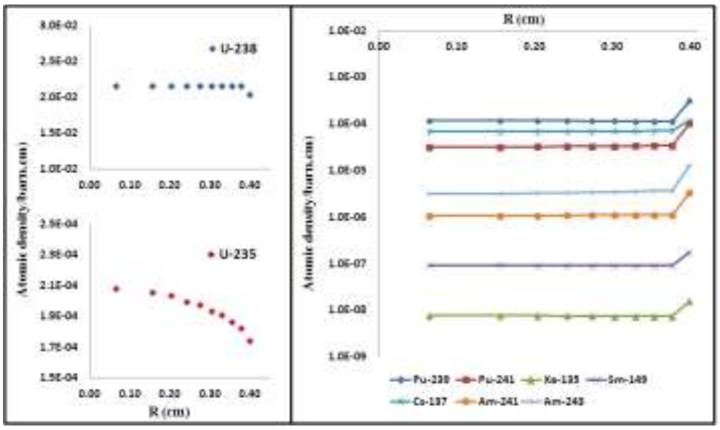

Figure 5: The radial distribution of isotopes.

II.2. Results of fuel assemblies calculations using PIJ

Table 2: The k_{inf} of fuel cells.

Types	3A	3B	3C	3C	2C	2C	2C 24BP	2C 20BP	2C 16BP	2B	2A	1	1
k-inf	1.014	0.815	1.246	0.894	1.184	0.826	0.998	1.025	1.054	1.007	0.936	0.923	0.59

With the assumption that the Boron concentration in moderator is 1700 ppm, the temperatures of fuel rods, cladding and moderator are 900K, 580K and 580K, these results prepare cell constants of fuel assemblies, which can be used for full-core calculation. The assemblies 2C and 3C do not have neutron absorber, because of that, their k_{inf} are higher. The assemblies with control rods have lowest k_{inf}. The k_{inf} of assemblies with UO_2-Gd_2O_3 and burnable poison rods approximates 1. Fission rates of one quarter fuel assembly type 3B (fuel assembly with 4.4% enrichment, 16 UO_2-Gd_2O_3 rods and 24 control rods) are shown in Figure 6. The reaction rates of the neighbouring fuel rods are lower because of control rods. Central guide tube has water, this increases neutron moderation. Reaction rates of UO_2-Gd_2O_3 are low because Gadolinium has large captured cross section for thermal neutron. At the corner of fuel assemblies, fission rates are largest because there are no control roand UO_2-Gd_2O_3 rods.

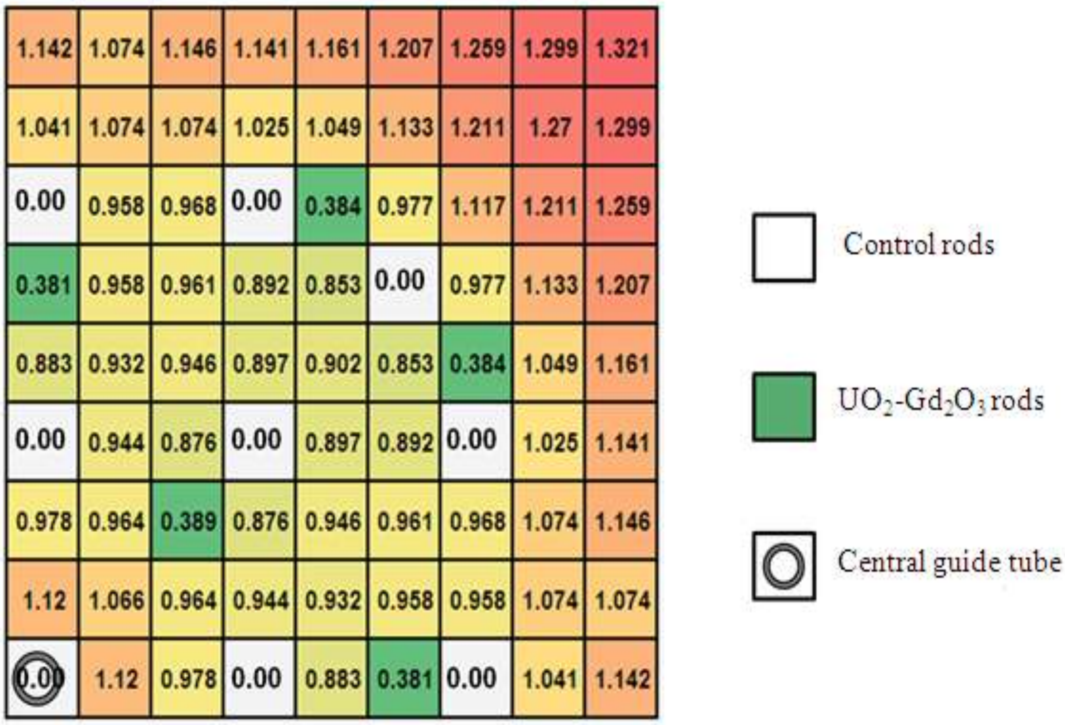

Figure 6: The fission rates of one quarter fuel assembly type 3B.

II.3. Results of the core modeling with module CITATION

In the design of the reactor, boron acid concentration are kept in the range of 2300-3000 ppm when reactor is in cold shutdown. Concentration of boron acid will decrease to 1600 - 1800 ppm when reactor is in hot shutdown - before restart by withdrawing control rods 0. For this reason, the authors chose the calculating condition in which all control rods are withdrawn and the boron concentration is 1700 ppm. The calculated k_{eff} for reactor core is 1.0083, this shown that the core is critical.

In Figure 7, the flux of neutron which has energy under 0.025eV is shown, thermal neutron flux in the core is fairly uniform. The thermal neutron flux in reflector is much higher than in the core, this is caused by two reasons: reflecting in reflector and absorbing of ^{235}U in the core. In Figure 8, neutron flux in the centre of the core is highest. The fast neutron flux is approximate 3 times higher than the thermal neutron flux, in the centre of the core, this value is 3.3 times. Because in this calculation condition, temperature distribution and coolant flow are not taken into account, neutron flux is symmetric with symmetry plane is central horizontal plane of the core. The power distribution in the core is not uniform across the radius, the maximum value of power is 1.48. The power level is lowest in the centre of the core and highest at the boundary where there are high enrichment fuel assemblies. Inside the reactor core, the power peaks are in positions of assemblies type 2C (3.5% ^{235}U fuel).

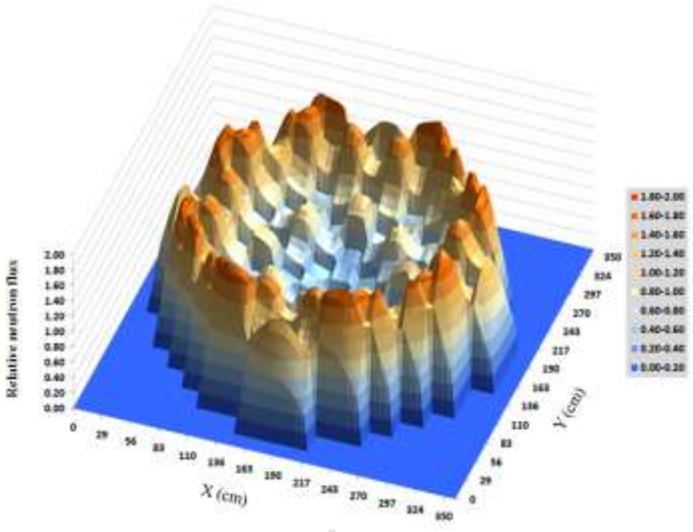

Figure 7: Radial distribution of thermal neutron in the core.

Because low enrichment fuel assemblies arranged in the centre of the core and concentration of boron is 1700 ppm, the power level in the centre of the core is lower than other positions. When boron is burnt and decrease 2-3 ppm/day, power level in centre of the core increases, power level in the core is fairly uniform.

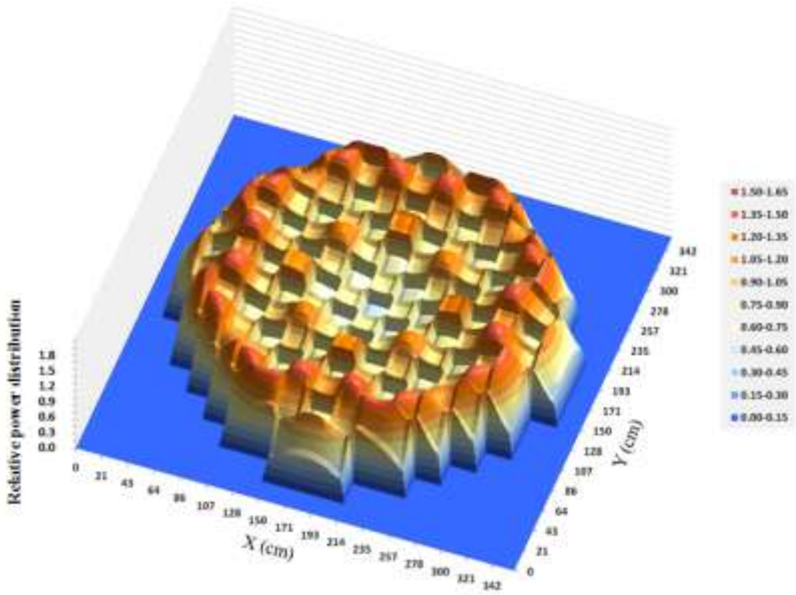

Figure 8: Radial power distribution in the core.

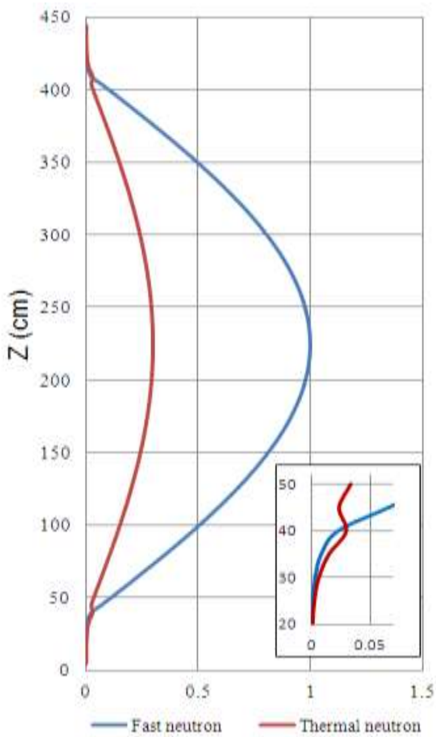

Figure 9: Vertical distribution of neutron flux in the core.

III. CONCLUSIONS

This project performed nuclear characteristics of the Japanese PWR 900MWe calculations. The results are: the infinite neutron multiplication factor(k_{inf}) versus burnup, the distribution of nuclide densities in the pin cells; the pin-wise fission rate distribution in the assembly and the effective multiplication factors (keff), and the power distribution in the core. In the near future, the authors' research orientation is using the Monte-Carlo method to calculate the benchmark problems and estimate the results in this project.

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