Neutronic analysis of PWR core using DRAGON, TRIVAC, and DONJON computer codes

D H Sukarno

1Center of Regulatory System and Technology Assessment for Nuclear Installations and Materials, BAPETEN, Jl. Gajah Mada 8 Jakarta 10120, Indonesia

E-mail: d.hidayanti@bapeten.go.id

Abstract. Operation of NPP has potential of nuclear hazard and radiation hazard. Therefore, all efforts are absolutely needed to be performed to ensure the safety of NPP operation. Neutronics analysis has important role in ensuring that safety aspect. In neutronics analysis, computer codes that could well simulate the neutronics condition in reactor core are needed. In this paper, neutronics analysis of 2652 MWt PWR core has been performed using DRAGON, TRIVAC, and DONJON computer codes. The purpose of simulation is to conduct the calculation of neutronics parameters which are important for safety. The neutronics analyses was conducted in two stages, i.e. cell calculation using DRAGON and full core calculation using TRIVAC and DONJON. For verification, the comparation between the calculation results of DRAGON, TRIVAC, and DONJON, and those of other computer codes results found in literature was performed. The effective multiplication factors calculated by DONJON for conditions where all control rods were fully up and fully down, and for ‘one stuck rod’ condition were 1.23399, 0.69705, and 0.91340, respectively. The core maximum power and average power were found to be 21166.1 kW and 16891.7 kW, respectively, with the power peaking factor of 1.2530. The maximum and average neutron fluxes in the core were 4.019038E+14 n/cm^2 s and 9.035312E+13 n/cm^2 s, respectively. The results of fuel cell calculation using DRAGON and the full core calculation using TRIVAC and DONJON have the differences of 4% and 1.3%, respectively, compared to the calculation results using other computer codes. This research shows that DRAGON, TRIVAC, and DONJON have capabilities to perform the calculation of neutronics parameters for NPP, specifically PWR type.

1. Introduction
Pressurized Water Reactor (PWR) is type of Nuclear Power Plant (NPP) that is mostly operated in the world commercially. European Nuclear Society reported that the number of operating PWR worldwide reached 65% [1]. At the beginning, PWR was designed by Westinghouse Bettis Atomic Power Laboratory for the purpose of power generation in the military ship and then it was developed by Westinghouse Nuclear Power Division for commercial uses [2].

The application of NPP tends to be one of reliable and realistic options in ensuring the sustainability of national electricity supply and enhancing national economic growth. The operation of NPP has hazardous potential, especially radiation and nuclear hazards. Therefore, safety aspect of NPP needs to be assessed and analysed. One of safety aspect that is important to be assessed is neutronics aspect.
performing neutronics analysis, we need computer codes to model neutronics condition in reactor core, such as MCNP, WIMS, SCALE, and CITATION. Ecole Polytechnique de Montreal of Canada has developed several free software for neutronics calculation of nuclear reactor, i.e. DRAGON, TRIVAC, and DONJON. The development of DONJON software is actually dedicated for CANDU reactor calculation. Hence, the applicability of DRAGON, TRIVAC, and DONJON computer codes for other types of nuclear reactor is needed to be investigated.

In this research, the neutronics analysis of typical PWR core was conducted using DRAGON, TRIVAC, and DONJON computer codes. The purpose of simulation is calculating several neutronics parameters which is important for safety. For verification, the results of DRAGON, TRIVAC, and DONJON calculations were compared to other computer codes results in the reference. Through this research, we will know the applicability of DRAGON, TRIVAC, and DONJON computer codes in performing the calculation of important neutronics parameters of NPP, specifically PWR.

2. PWR core design
Typical PWR core analysed in this paper refers to the Huda et.al. research [3]. The core has total power of 2652 MWth with three types of UO$_2$ pellet enrichments, i.e. 2.1%, 2.6%, and 3.1%. UO$_2$ pellets are encased in a zircaloy cladding. He gas is inserted into the cladding to support the heat transfer process and to detect the leakage. All core specifications, including fuel pin, fuel assembly, as well as core configuration refer to the Huda et al. research [3]. The detail specification of fuel pin is given in Table 1.

| Parameter                      | Dimension (mm) |
|--------------------------------|----------------|
| Inner diameter of fuel meat    | 8.19           |
| Cladding thickness             | 0.57           |
| Cladding outside diameter      | 9.5            |
| Pitch                          | 12.6           |
| Fuel pin length                | 3650           |

A fuel assembly consists of 264 grids of fuel rods, 1 grid of water channel, and 24 grids of control rods in 17 x 17 square array configuration (see figure 1). PWR fuel bundles are about 4 meters in length and arranged vertically in the core.

![Figure 1. Horizontal cutaway view of a PWR fuel assembly (17 x 17 array) [3]](image)

The PWR core that was analysed in this research consists of 157 fuel assemblies with the configuration showed at figure 2.
3. Research methodology
The calculation method performed in this research consists of two stages:

- cell calculation using DRAGON. The cell calculation produces the macroscopic cross section data of all core materials.
- core calculation using TRIVAC and DONJON.

All neutronics calculations were done at zero burnup (fresh fuel condition).
3.1. Cell calculation

DRAGON code was used to perform fuel pincell, fuel assembly, moderator region, and reflector assembly calculations. The result of the fuel assembly, moderator region, and reflector assembly calculations would then be used as the input for core calculation done by TRIVAC and DONJON codes. The fuel pin was modelled as mixed 2D cartesian cell and 1D central annular pin. For better result, the fuel meat region was split into two regions. It was assumed that the gap region to be converted into cladding region. So, it needed the re-calculation of material density in the cladding region. Figure 3 shows the basic geometry representation of pincell as treated by DRAGON code.

![Figure 3. Basic geometry representation of pincell as treated by DRAGON code](image)

According to the figure 3, region 1 is fuel meat, region 2 is cladding, and region 3 is coolant (moderator). Fuel assembly was modelled in 2D cartesian as shown at figure 4.

![Figure 4. Basic geometry representation of fuel assembly as treated by DRAGON code](image)

By using diagonal (DIAG) and symmetry (SYME) boundary conditions, only one-eight of the fuel assembly geometry was created. In the fuel assembly simulation, the control rods were assumed not introduced into the core, so twenty-four control rod positions in the fuel bundle were occupied by moderator (light water). It was also assumed that no Boron including in the moderator. Geometrical modelling in DRAGON was conducted by GEO module.

Library used for supplying microscopic cross section data was taken from ENDFB-VI release 8 in DRAGON format (draglib) with SHEM-361 energy meshes. SHEM-361 is an energy mesh defined by Alain Santamarina and Alain Hébert and obtained by refining the group structure of SHEM-281 in the resolved energy domain, above 22.5eV [4]. The number of energy groups used in the calculation was 361 groups (from 1.1E-4 to 1.964E+7 eV).

The spatial domain (geometry) that was previously defined by GEO module was tracked by EXCELT module. The tracking operation performs zone numbering operations, volume and surface area calculations and generates the required integration lines for a geometry [5]. The isotropic tracking condition (TISO) was set with 12 for the number of tracking angles and 20.0 for the density of integration lines. In order to account for the self-shielding effect, the SHI module was activated during cross section generation. The information resulting from this tracking operation was then used for the collision probability calculation done by ASM module. Using EDI module, the complete neutron flux homogenization over all regions took place and the group condensation of the flux was done into two groups with 4.0 eV as the energy limit between those two groups. Finally, the neutron transport equation was solved by FLU module.
3.2. Full-core calculation

Full-core calculation was done using two codes, namely TRIVAC and DONJON. Two energy groups were used for the calculations in TRIVAC and DONJON based on the output data of DRAGON code. TRIVAC is a computer code intended to compute core criticality, while DONJON is intended for core criticality, neutron flux and power distribution calculations. The 3D cartesian geometry was built to simulate the 1/4 full-core of PWR. In TRIVAC, the macroscopic cross sections and diffusion coefficients were read from the input data based on the output data of DRAGON code. DONJON execution depends on DRAGON and TRIVAC codes. The DRAGON modules are used to define the reactor geometry, to provide the macroscopic cross-section libraries and to interpolate the multi-parameter database resulting from the lattice calculations. The TRIVAC solver modules are used to perform a spatial discretization of the reactor geometry and to provide the numerical solution according to the user-selected numerical procedure [6].

The core calculation was done in three conditions, i.e.:

- all the control rods were fully-up;
- all the control rods were fully-down;
- the group of control rods which had the highest reactivity was failed to fall (one stuck rod criterion).

The group of control rods with the highest reactivity was the control rods at the hottest channel position.

4. Results and discussion

The calculation of the infinite multiplication factor ($k_\infty$) for the typical PWR fuel pin and fuel assembly was performed by DRAGON code and the results are shown in Table 3.

| Type of enrichment | Fuel Pin | Fuel Assembly |
|--------------------|----------|---------------|
| 2.1%               | 1.18574  | 1.20302       |
| 2.6%               | 1.24058  | 1.26102       |
| 3.1%               | 1.28102  | 1.30383       |

To assess the accuracy of the calculation results, the $k_\infty$ values produced by DRAGON are compared with those produced by other neutronic codes, such as WIMSD and MCNP [3]. Table 4 depicts the comparison result of the $k_\infty$ values.

| Type of enrichment | Fuel Pin | MCNP | Fuel Assembly |
|--------------------|----------|------|---------------|
| 2.1%               | 1.18574  | 1.23393 | 1.20302 | 1.24853 | 1.25445 |
| 2.6%               | 1.24058  | 1.28635 | 1.28887 | 1.26102 | 1.30372 | 1.30992 |
| 3.1%               | 1.28102  | 1.32481 | 1.32812 | 1.30383 | 1.34424 | 1.35041 |

A good agreement is observed between the calculations performed by the WIMSD and MCNP codes, but the DRAGON result is found to be underpredictive compared to the results of WIMSD and MCNP codes. The differences fall within the range of 4%. The different cross section libraries used is predicted greatly influencing the difference between the DRAGON result and the WIMSD and MCNP...
results. Library used in DRAGON calculation was ENDFB-VI release 8 with 361 energy groups. The WIMSD calculation used libraries integrated from many sources such as ENDF/B-VI, JENDL-3.3, CENDL-2, and JEFF-3.0, with 172 energy groups (from 19 MeV to $10^5$ eV). In the MCNP simulation, all the neutron cross sections were taken from the ENDF/B-VI release 8 and ENDF/B-V. The number of energy groups and its condensation setting are also predicted giving the great influence to the calculation results. To assess the accuracy of the DRAGON result, various parameter, model, and calculation method options available in DRAGON code need to be studied in depth, for example the option of the tracking model with its parameters settings.

The effective multiplication factors ($k_{eff}$) of PWR core calculated by TRIVAC and DONJON codes where all the control rods were fully-up were 1.23416 and 1.23399 consecutively. These results are then compared with other codes calculation results performed by Huda et al. [3] as shown in Table 5.

| TRIVAC  | DONJON  | WIMSD+TWOTRAN | WIMSD+CITATION | MCNP  |
|---------|---------|---------------|----------------|-------|
| 1.23416 | 1.23399 | 1.25679       | 1.25353        | 1.26382 |

The result of TRIVAC calculation agrees well with that of DONJON calculation. But, there is the difference between the TRIVAC or DONJON results and other codes results in the maximum range of 1.3%. The factors contributing to this difference are still the same with the ones predicted on the difference of the cell calculation results. The core effective multiplication factor calculated by DONJON for conditions where all the control rods were fully-down and in 'one stuck rod' condition were found to be 0.69705 and 0.91340 respectively.

To further assess the accuracy of the PWR core model, it is essential to analyse the distribution of neutron fluxes or power within the core. The flux and power distribution within the core can be predicted using DONJON code. The analysis of power and neutron flux distribution was done by assuming that all of the control rods were fully-up. The maximum power was found to be 21166.1 kW which was located on channel G8 (see figure 5), the average power was 16891.7 kW, and the power peaking factor was 1.2530.

![Figure 5. Fuel channel map](image)

The radial power distribution within the core (G1-G15) calculated by DONJON code is shown in figure 6 and the axial power distribution at the hottest channel is illustrated in figure 7.
The maximum neutron flux was found at channel G8 to be $4.019038 \times 10^{14}$ n/cm$^2$s and the average neutron flux over the reactor core was $9.035312 \times 10^{13}$ n/cm$^2$s. The radial and axial neutron flux distributions within the core are shown in figure 8 and 9, respectively.

Figure 6. Radial power distribution within the PWR core

Figure 7. Axial power distribution at the hottest channel

Figure 8. Radial thermal neutron flux distribution within the core
5. Summary
The calculation of several neutronics parameters of a typical 2652 MWt PWR core at zero burnup value has been conducted by using DRAGON, TRIVAC, and DONJON codes. The DRAGON calculated infinite multiplication factors for fuel pin and fuel assembly were found to be 1.18574 and 1.20302 for 2.1% enrichment, 1.24058 and 1.26102 for 2.6% enrichment, and 1.28102 and 1.30383 for 3.1% enrichment, respectively. The core effective multiplication factor calculated by DONJON for fully up control rods, fully down control rods, and one stuck rod conditions were found to be 1.23399, 0.69705, and 0.91340, respectively. The maximum power was 21166.1 kW at G8 fuel channel with the power peaking factor of 1.2530. The average power was 16891.7 kW. The maximum thermal neutron flux was 4.019038E+14 n/cm² s and the average thermal neutron flux over the reactor core was 9.03512E+13 n/cm² s. The infinite multiplication factors calculated by DRAGON and other codes (WIMSD and MCNP) show the maximum difference of about 4%. The maximum difference of core effective multiplication factor between TRIVAC or DONJON results and other codes (WIMSD+TWOTRAN, WIMSD+CITATION, MCNP) results was to be 1.3%. Finally, it has been shown that DRAGON, TRIVAC, and DONJON computer codes have the capability to perform the PWR type neutronics calculation.

Acknowledgments
Author wishes to acknowledge the assistance and support of W.F.G. van Rooijen from Research Institute of Nuclear Engineering, University of Fukui, Japan, under MEXT Programme.

References
[1] European Nuclear Society 2012 Nuclear power plants, world-wide, reactor types, European Nuclear Society (http://www.euronuclear.org/info/encyclopedia/n/npp-reactor-types.htm).
[2] Pongpuak J 2010 Pressurized water reactors Nuclear Energy Materials and Reactors Vol.I
[3] Huda M Q et al. 2011 Design studies of a typical PWR core using advanced computational tools and techniques J. Annals of Nuclear Energy 38 p 1939–1949.
[4] Hebert A et. al. 2008 Draglib Download Page (http://www.polymtl.ca/merlin/libraries.htm/CITEhem361).
[5] Marleau G et al. 2012 A User Guide for DRAGON Version4 (Technical Report IGE–294).
[6] Sekki D et al. 2012 A User Guide for DONJON Version4 (Technical Report IGE–300)