Evaluation of the AP1000 delayed neutron parameters using MCNP6

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Abstract. The MCNP6 code contains numerous features, one of those is to determine the delayed neutron parameters. The accuracy of calculated delayed neutron parameters affect the accuracy of transient or dynamic condition. The objective of this paper is to determine the delayed neutron parameters of the advance PWR reactor, AP1000, using MCNP6 code with the recent ENDF/B evaluated nuclear data file ENDF/B-VII.1. The MCNP6 calculation results shows that the maximum difference occurred in the $\beta_i$ and $\lambda_i$ parameters are 38.30% and 45.63%, respectively. The superiority of MCNP6 can be seen in the change of prompt neutron life time ($\ell$) parameters that cannot be obtained by the deterministic code, so it can be used in the sensitivity analysis of the delayed neutron parameters. Based on this research work, the accident analysis of the AP1000 reactor use the effective delayed neutron fraction ($\beta_{\text{eff}}$) of 0.0051 and the prompt neutron life time ($\ell$) of 19.5 $\mu$s for the first cycle.

Key-words: PWR, AP1000, delayed neutron parameters, MCNP6, ENDF/B-VII.1

1. Introduction
Recent advances in PC based high performance computation technologies enable us to apply the Monte Carlo codes to large scale problems, such as detailed three-dimensional whole-core calculations for commercial power reactors within acceptable computation time at low cost. One of Monte Carlo codes which contains numerous new features not found in either code is MCNP6 version 1 [1]. The code has capability to determine the delayed neutron parameters, either effective delayed neutron fraction ($\beta_{\text{eff}}$), effective neutron generation time ($\Lambda_{\text{eff}}$), delayed neutron precursor fraction ($\beta_i$) and effective delayed neutron precursor decay constant ($\lambda_i$).

The delayed neutron parameters are fundamental to describe the kinetic and dynamic response of both critical and subcritical nuclear systems to internal or external perturbation. The accuracy of calculated delayed neutron parameters affect the accuracy of transient or dynamic condition. The MCNP code is widely used for the PWR reactor analysis [2-7]. The MCNP6 code as well as the ENDF/B-VII.1 have been validated for many types of nuclear reactor [8-11] that show some improvements.

The delayed neutron parameters of AP1000 reactor [12] has been determined by using deterministic codes in the previous research works [13-14]. The research work in Ref. [13] evaluated the delayed neutron parameters for BOC (beginning of cycle) at the first core cycle. However, Ref [14] evaluated the delayed neutron parameters as a function of burn-up from BOC to EOC (end of cycle) at the first core cycle. Those research works used the old nuclear data and delayed neutron data, such as ENDF/B-VII.1.
VII.0 and JENDL-3.3. Therefore, the research work based on the recent ENDF/B evaluated nuclear data and new version of MCNP code is needed to evaluate the delayed neutron data for complex core configuration such as the AP1000 reactor. The objective of this paper is to evaluate the delayed neutron parameters of the AP1000, such as $\beta_{\text{eff}}$, prompt neutron life time ($\ell$), $\beta_i$ and $\lambda_i$, for BOC and EOC at the first core cycle using MCNP6 code. The results of this work can be used in the accident analysis for the first cycle of AP1000 reactor.

In this research work, the atomic densities of fission product nuclides for 16 types of fuel assemblies are determined by using PIJ module in the SRAC2006 code [15]. The detail geometry model of AP1000 reactor for the MCNP6 code input is adopted from the previous research work that has been validated with the design value [16]. The core calculations are carried out for the BOL (fresh core) and EOL for the given boron concentrations. The average burn-up of AP100 fuel assemblies at the EOL condition are obtained from Ref [17]. The evaluation is carried out by comparing with the design and previous research works [13-14,18].

2. Brief description of AP1000 reactor
The Westinghouse Advanced Passive PWR AP1000 is an 1100 MWe class PWR, which is conservatively based on proven PWR technology, but with an emphasis on safety features that rely on natural forces [12]. The thermal power of 3400 MW is generated by 157 fuel assemblies with three radial regions with the enrichments of 2.35 w/o, 3.40 w/o and 4.45 w/o, as shown in Fig. 1. There are 16 types of fuel assemblies based on the enrichment and the number IFBA and Pyrex [18]. The main data and specification of core and fuel configuration of AP1000 reactor are described in the Ref [18], completely.

![Figure 1. AP1000 core configuration [9][13]](image-url)
Based on the design document [19], the delayed neutron fraction ($\beta_{\text{eff}}$) typically yield values no less than 0.0075 at beginning of cycle (fresh core) and 0.0050 at end of cycle for the first cycle. For the next cycles, the $\beta_{\text{eff}}$ of 0.0049 (BOC) and 0.0044 (EOC) are used in the accident analysis. It is noted that the prompt neutron life time ($\ell$) of AP1000 reactor is 19.8 $\mu$s as described in Ref [18].

3. Methodology
The delayed neutron parameters that can be calculated by MCNP6 are the effective delayed neutron fraction ($\beta_{\text{eff}}$), effective neutron generation time ($\Lambda_{\text{eff}}$), delayed neutron fraction of precursors in group $i$ ($\beta_i$) as well as effective delayed neutron precursor decay constant in group $i$ ($\lambda_i$). Those parameters are calculated by using the data of the delayed neutron data ($\nu_d(E)$) and the delayed neutron spectra ($\chi_d(E)$) from the ENDF/B-VII.1. In this present work, the calculation objects are the BOC and EOC cores at the first cycle, means all fuel assemblies at BOC are fresh. The delayed neutron parameters are calculated in the given boron concentration for BOC and EOC[18].

The following calculation steps are used in this research work:
1. The detail geometry model of AP1000 reactor are adopted from Ref [16] as shown in the Fig 2.

![Figure 2. Detail 3-dimension model of the AP1000 core in X-Y plane (radially) [16]](image)

2. The atomic densities of light water moderator are determined based on the temperature, pressure and the boron concentration. For the BOC case [18], the moderator temperatures are 293K (cold) and 600K (hot) as well as the fuel temperatures are 293 K (cold) and 600K (hot). The pressure is 158.19 bar for hot condition. The boron concentrations for cold and hot (equilibrium xenon) at BOC and EOC are 1574 ppm and 827 ppm, respectively.
3. The average burn-up of EOC core at first cycle is assumed by 20 GWD/MTHM[17]. The atomic densities of fission product nuclides at 20 GWD/MTHM is determined by the PIJ module of SRAC2006 code.

4. The atomic densities changes of burnable absorbers, such as IFBA and PYREX, were calculated by the PIJ module in SRAC2006 code.

5. The evaluation was carried out by comparison with the design values [18] and the deterministic codes of BATAN-3DIFF and CITATION-SRAC2006 results [13-14].

The MCNP6 calculations used 600 cycles with 100,000 neutrons per cycle in the KCODE card. The first 100 cycles are skipped in order to avoid source convergence problem. The delayed neutron parameters are calculated by the KOPTS card. The $S(\alpha,\beta)$ thermal neutron scattering for light water is chosen as same as the temperature condition.

4. Results and discussions

Table 1 shows the calculated effective delayed neutron fraction ($\beta_{eff}$) and prompt neutron life time ($\ell$) parameters for the AP1000 reactor at BOC and EOC with boron concentration of 1574 ppm and 827 ppm, respectively. For the BOC, the MCNP6 gives a $\beta_{eff}$ value that is relative difference by -7.33% from the $\beta_{eff}$ value obtained by design value [18]. The deterministic code calculations by using BATAN-3DIFF and CITATION-SRAC have the relative difference by -2.67% and -5.33%, respectively. In addition, for EOC, the calculation results of MCNP6 and CITATION-SRAC have relative difference with design value by 3.80% and 2.00%, respectively.

Using 3σ uncertainty (99% confidence level), at BOC condition, the results of MCNP6 is within margin with those of BATAN-3DIFF and CITATION-SRAC, but out of range from the design reference. On the other hand, at EOC condition, the calculation result of MCNP6 is within margin with those of CITATION-SRAC and design reference. It seems the different delayed neutron data and the number of neutron energy group being used in those codes do not contributes significantly to the calculated $\beta_{eff}$. It is noted that the BATAN-3DIFF, CITATION-SRAC and MCNP6 use the evaluated nuclear data based on Brady-England, JENDL-3.3 and ENDF/B-VII.1 [13-14], respectively. Beside that the number of neutron energy group for macroscopic cross-section in the BATAN-3DIFF, CITATION-SRAC and MCNP6 codes are 2,10 and continuous energy, respectively.

The value of $\beta_{eff}$ at EOC condition are dominated by the fission products, such as $^{239}$Pu, $^{240}$Pu, $^{241}$Pu and $^{242}$Pu (as shown in Fig 3), so, compared with the BOC condition, the value of $\beta_{eff}$ decrease by 30%, 28.2% and 25.3% in the design value, the CITATION-SRAC and the MCN6 results, respectively. The neutron spectrum at EOC condition become harder than at BOC condition. Therefore, the calculated $\beta_{eff}$ is sensitive in the softer neutron spectrum of BOC condition. The possibility that the boron concentration give effect to the calculated $\beta_{eff}$ needs to be analyzed in the future work.

| Parameters | Design [18] | BATAN-3DIFF [13] | CITATION-SRAC [14] | MCNP6 |
|------------|-------------|------------------|---------------------|-------|
| BOC (cold, zero power, boron concentration of 1574 ppm) | | | | |
| $\beta_{eff}$ | 0.0075 | 0.0073 | 0.0071 | 0.00695±0.00013 |
| | (-2.67%)$^a$ | (-5.33%)$^a$ | (-7.33%)$^a$ | |
| $\ell$, μs | 19.8 | 14.1 | 20.8 | 19.53±0.04 |
| | (-28.79%)$^a$ | (5.05%)$^a$ | (-1.36%)$^a$ | |
| EOC (hot, zero power, boron concentration of 827 ppm) | | | | |
| $\beta_{eff}$ | 0.0050 | - | 0.0051 | 0.00519±0.00012 |
| | - | - | (2.00%)$^a$ | (3.80%)$^a$ |
| $\ell$, μs | - | - | 20.8 | 20.44±0.04 |

Note: a: relative difference with design value
Table 1 shows that the prompt neutron life time ($\ell$) parameters obtained by the MCNP code give a very good agreement with the design value compared with the deterministic codes since the relative difference is -1.36%. The $\ell$ parameter at EOC is higher than BOC by 4.66%. It means the $\ell$ is not influenced by the burn-up condition or neutron spectrum change. The MCNP6 codes can evaluate the small change of $\ell$ parameter while the CITATION-SRAC code give same value for BOC and EOC. Based on the calculation results of BATAN-3DIFF code, the $\ell$ is sensitive to the number of neutron energy group.

![Figure 3. The consumption fuel isotopes and production of Pu isotopes of AP1000](image)

Table 2 shows the calculated $\beta_i$ and $\lambda_i$ values obtained by using BATAN-3DIFF, CITATION-SRAC and MCNP codes. If we used the MCNP result as the reference, it is clear that the two deterministic codes give a large different since the difference is in the range of -27.87% - 38.30% and -31.79% - 19.57% at BOC and EOC, respectively. The maximum difference occurred in the 6th group. It seems that the calculation results is strongly depend on the delayed neutron fraction of precursors in group $i$ and absolute delayed neutron yield for each fission product nuclides.

The calculated effective delayed neutron precursor decay constant in group $i$ ($\lambda_i$) in Table 2 shows same situation as the $\beta_i$. For the BOC, the difference is in the range of -0.31% - 14.73% and -6.36% - 40.19% by using the BATAN-3DIFF and CITATION-SRAC codes, respectively. The difference between BOC and EOC is relatively same since the range is in -5.53% - 45.63% at EOC by using CITATION-SRAC code. As discussed above, the $\lambda_i$ is depend on the delayed neutron decay constant of delay group $i$ for each fission product nuclide, too.
Table 2. The calculated $\beta_i$ and $\lambda_i$ values for AP1000 reactor at BOC and EOC

| Group Precursor, $i$ | BATA-3DIFF [13] | CITATION-SRAC [14] | MCNP |
|---------------------|----------------|-----------------|------|
|                     | $\beta_i$     | $\lambda_i$    | $\beta_i$ | $\lambda_i$ |
| BOC                 |               |                |       |
| 1                   | $2.77 \times 10^{-4}$ | $1.330 \times 10^{-2}$ | $2.11 \times 10^{-4}$ | $1.251 \times 10^{-2}$ | $2.40 \times 10^{-4} \pm 2 \times 10^{-5}$ | $1.336 \times 10^{-2} \pm 0.0$ |
| 2                   | $1.40 \times 10^{-3}$ | $3.250 \times 10^{-2}$ | $1.45 \times 10^{-3}$ | $3.071 \times 10^{-2}$ | $1.21 \times 10^{-3} \pm 6 \times 10^{-5}$ | $3.260 \times 10^{-2} \pm 0.0$ |
| 3                   | $1.20 \times 10^{-3}$ | $1.219 \times 10^{-1}$ | $1.36 \times 10^{-3}$ | $1.150 \times 10^{-1}$ | $1.22 \times 10^{-3} \pm 6 \times 10^{-5}$ | $1.211 \times 10^{-1} \pm 0.0$ |
| 4                   | $2.50 \times 10^{-3}$ | $3.169 \times 10^{-1}$ | $2.80 \times 10^{-3}$ | $3.100 \times 10^{-1}$ | $2.68 \times 10^{-3} \pm 8 \times 10^{-4}$ | $3.059 \times 10^{-1} \pm 2 \times 10^{-5}$ |
| 5                   | $1.27 \times 10^{-3}$ | $9.886 \times 10^{-1}$ | $9.64 \times 10^{-4}$ | $1.208 \times 10^{-1}$ | $1.27 \times 10^{-3} \pm 5 \times 10^{-5}$ | $8.617 \times 10^{-1} \pm 9 \times 10^{-5}$ |
| 6                   | $6.50 \times 10^{-4}$ | $2.954 \times 10^{-3}$ | $3.39 \times 10^{-3}$ | $3.238 \times 10^{-3}$ | $4.70 \times 10^{-4} \pm 5 \times 10^{-5}$ | $2.894 \times 10^{-3} \pm 5.0 \times 10^{-4}$ |
| EOC                 |               |                |       |
| 1                   | -             | -               | $1.42 \times 10^{-4}$ | $1.263 \times 10^{-2}$ | $1.50 \times 10^{-4} \pm 2 \times 10^{-5}$ | $1.337 \times 10^{-2} \pm 0.0$ |
| 2                   | -             | -               | $1.10 \times 10^{-3}$ | $3.060 \times 10^{-2}$ | $9.20 \times 10^{-4} \pm 5 \times 10^{-5}$ | $3.185 \times 10^{-2} \pm 0.0$ |
| 3                   | -             | -               | $9.64 \times 10^{-4}$ | $1.196 \times 10^{-1}$ | $8.30 \times 10^{-4} \pm 5 \times 10^{-5}$ | $1.197 \times 10^{-1} \pm 2 \times 10^{-5}$ |
| 4                   | -             | -               | $1.92 \times 10^{-3}$ | $3.209 \times 10^{-1}$ | $1.95 \times 10^{-3} \pm 7 \times 10^{-5}$ | $3.068 \times 10^{-3} \pm 3 \times 10^{-5}$ |
| 5                   | -             | -               | $7.35 \times 10^{-4}$ | $1.268 \times 10^{-2}$ | $9.40 \times 10^{-4} \pm 5 \times 10^{-5}$ | $8.707 \times 10^{-4} \pm 1.1 \times 10^{-4}$ |
| 6                   | -             | -               | $2.66 \times 10^{-4}$ | $3.274 \times 10^{-3}$ | $3.90 \times 10^{-4} \pm 3 \times 10^{-5}$ | $2.914 \times 10^{-4} \pm 7.9 \times 10^{-4}$ |

5. Conclusions
The evaluation of the AP1000 delayed neutron parameters performed by the recent ENDF/B evaluated nuclear data file ENDF/B-VII.1 with the MCNP6 code leads to very satisfactory results. The superiority of the MCNP6 code can be seen in the change of prompt neutron life time ($\ell$) parameters that cannot be obtained by the deterministic code, so can be used in the sensitivity analysis of the delayed neutron parameter. The maximum difference occurred in the $\beta_i$ and $\lambda_i$ parameters by the range of 38.30% and 45.63%, respectively. The accident analysis of the AP1000 reactor for the first cycle can be carried out using this research work results because the results have a very good agreement with the designn value.

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