The Comparative Investigation and Calculation of Thermo-Neutronic Parameters on Two Gens II and III Nuclear Reactors with Same Powers

Mousavi Shirazi, Seyed Alireza¹*; Rastayesh, Sima²

Abstract—Whereas in the third generation nuclear reactors, dimensions of core and also the kind of coolant and enrichment percent of fuel have significantly changed than the second generation, therefore in this article the aim is based on a comparative investigation between two same power reactors of second and third generations, that the neutronic parameters of both reactors such as: $K_\infty$, $K_{eff}$ and its details and thermal hydraulic parameters such as: power density, specific power, volumetric heat rate, released power per fuel volume unit, volume and mass of clad and fuel (consisting fissile and fertile fuels), be calculated and compared together. By this comparing the efficiency and modification of third generation nuclear reactors than second generation which have same power can be distinguished.

In order to calculate the cited parameters, some information such as: core dimensions, the pitch of lattice, the fuel matter, the percent of enrichment and the kind of coolant are used. For calculating the neutronic parameters, a neutronic program entitled: SIXFAC and also related formulas have been used. Meantime for calculating the thermal hydraulic and other parameters, analytical method and related formulas have been applied.

Keywords—Nuclear reactor, second generation, third generation, thermo-neutronics parameters.

Nomenclature

$\eta$: Thermal fission coefficient
$f$: Thermal absorption coefficient
$p$: Resonance escape probability
$\varepsilon$: Fast fission coefficient
$P_{FNL}$: Fast neutrons non leakage probability
$P_{THNL}$: Thermal neutrons non leakage probability
$PTNL$: Total neutrons non leakage probability
$q_{th}$: Released heat rate in volume of reactor core
$q'$: Linear heat rate
$r$: Enrichment of fuel
$M_f$: Mass number of fission
$M_{nf}$: Mass number of fertile
$\sigma_f$: Fission cross section
$R_{fuel}$: Fuel rod radius
$dz$: Fuel rod height
$\rho_{pellet}$: Fuel pellet density
$n$: Fuel rods numbers

I. INTRODUCTION

In view of the great advancements in the nuclear reactors technology, the phenomenal and significant changes in the evolution of nuclear reactors is observed. Since the first nuclear reactor made in 1948 until modern reactors, great changes are obvious. These major changes are: the kind of reactor design, the percent of fuel enrichment, the kind of coolant and neutron moderator, the dimensions of core and safety are referred [1]-[2].

In this paper, the reactors considered here are Boiling Water Reactor (BWR): Kashiwazaki-Kariwa (Unit-6) Japan, and Forsmark (Unit-3) Sweden, respectively, that have same thermal power: $q_{th} = 3700$MWt = 1235MWe [3]-[4]. Both Kashiwazaki and Forsmark reactors are enumerated as second generation nuclear reactors.

II. METHODOLOGY

The related information to these reactors is shown in Table I:

### Table I

<table>
<thead>
<tr>
<th>Various parameters</th>
<th>Forsmark reactor</th>
<th>Kashirwazaki reactor</th>
<th>Various parameters</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pellet of UO₂</td>
<td>Zr-2</td>
<td>Zr-2</td>
<td>The kind of clad</td>
</tr>
<tr>
<td>Zr-2</td>
<td>Zr-2</td>
<td>The percentage of fuel enrichment</td>
<td></td>
</tr>
<tr>
<td>2.25 %</td>
<td>2.6 %</td>
<td>8.7 mm</td>
<td>The height of fuel pellet</td>
</tr>
<tr>
<td>10.3 mm</td>
<td>10.4 mm</td>
<td>0.63 mm</td>
<td>The thickness of fuel pellet</td>
</tr>
<tr>
<td>8.2 mm</td>
<td>0.86 mm</td>
<td>0.08 mm</td>
<td>The thickness of fuel external gap ($\delta_f$)</td>
</tr>
<tr>
<td>9.62 mm</td>
<td>12.3 mm</td>
<td>4066 mm</td>
<td>The height of fuel rod</td>
</tr>
<tr>
<td>4.6 m</td>
<td>5.16 m</td>
<td>3914 mm</td>
<td>The diameter of fuel rod</td>
</tr>
<tr>
<td>15.45 cm</td>
<td>15.5 cm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3.75 m</td>
<td>3.71 m</td>
<td>100</td>
<td>The fuel rods numbers at each fuel assembly</td>
</tr>
<tr>
<td>4.6 m</td>
<td>5.16 m</td>
<td>872</td>
<td>The numbers of fuel assembly</td>
</tr>
<tr>
<td>1.6 cm</td>
<td>1.6 cm</td>
<td>700</td>
<td></td>
</tr>
<tr>
<td>15.45 cm</td>
<td>15.45 cm</td>
<td>15.45 cm</td>
<td>The pitch of fuel assembly</td>
</tr>
<tr>
<td>700</td>
<td>872</td>
<td>3.75 m</td>
<td>The height of core</td>
</tr>
<tr>
<td>700</td>
<td>872</td>
<td>4.6 m</td>
<td>The diameter of core</td>
</tr>
<tr>
<td>3.71 m</td>
<td>3.71 m</td>
<td>20.8 m</td>
<td>The height of vessel</td>
</tr>
<tr>
<td>2.25 %</td>
<td>2.6 %</td>
<td>20.8 m</td>
<td>The inner diameter of vessel</td>
</tr>
<tr>
<td>8.7 mm</td>
<td>10.3 mm</td>
<td>6.4 m</td>
<td></td>
</tr>
<tr>
<td>10.4 mm</td>
<td>0.86 mm</td>
<td>7.1 m</td>
<td></td>
</tr>
<tr>
<td>0.63 mm</td>
<td>0.08 mm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>15.45 cm</td>
<td>15.5 cm</td>
<td>3.75 m</td>
<td>The height of vessel</td>
</tr>
<tr>
<td>15.45 cm</td>
<td>15.5 cm</td>
<td>4.6 m</td>
<td>The diameter of core</td>
</tr>
<tr>
<td>20.8 m</td>
<td>21.0 m</td>
<td>6.4 m</td>
<td>The inner diameter of vessel</td>
</tr>
<tr>
<td>20.8 m</td>
<td>21.0 m</td>
<td>7.1 m</td>
<td></td>
</tr>
<tr>
<td>6.4 m</td>
<td>7.1 m</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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The Neutronic Parameters

In this paper by using the neutronic software titled: SIXFAC and also by applying the related formulas, the values of infinite multiplying coefficient \( (K_\infty) \), effective multiplying coefficient \( (K_{\text{eff}}) \) and also the fundamental components of both coefficients for the two mentioned reactors in super critical state have been determined. In addition according to fuel enrichment percent in two cited reactors, the amounts of applied fuel in both reactors have been accurately computed.

The effective multiplying coefficient is: the ratio of generated neutrons in every generation to generated neutrons in last generation. So to operate the nuclear reactor in steady state, this parameter should be: 1 means the generated neutrons in every generation are equal with neutrons which have absorbed or leaked in last generation that means: critical state. The minimum value of \( K_{\text{eff}} \) is: 0 and maximum of it is: \( \nu \) namely: 2.43.

The effective multiplying coefficient is [5]:

\[
K_{\text{eff}} = \eta \cdot f \cdot p \cdot e \cdot P_{\text{FNL}} \cdot P_{\text{TINL}} = \eta \cdot f \cdot p \cdot e \cdot P_{\text{TINL}}
\]

As the thermal fission coefficient \((\eta)\) is [5,6]:

\[
\eta = \nu \cdot \frac{\Sigma_f}{\Sigma_{\text{pm}}} = \nu \cdot N^{235} \cdot \sigma_f^{235} \cdot g_f^{235} + N^{238} \cdot \sigma_f^{238} \cdot g_f^{238} + N^0 \cdot \sigma_f^0
\]

\[
N^{235} = \frac{m^{235} \cdot A}{M^{235}}
\]

\[
N^{238} = \frac{m^{238} \cdot A}{M^{238}}
\]

\[
\Rightarrow N^0 = 2 N^U = 2 \frac{m^U \cdot A}{M^U} = 2 \frac{m^U \cdot A}{r M^{235} + (1 - r) M^{238}}
\]

Also the thermal absorption coefficient \((f)\) is [5]:

\[
f = \frac{\Sigma_p}{\Sigma_f + \Sigma_{\text{pm}}} = \frac{N^{235} \cdot \sigma_a^{235} \cdot g_a^{235} + N^{238} \cdot \sigma_a^{238} \cdot g_a^{238} + N^0 \cdot \sigma_a^0}{N^{235} \cdot \sigma_a^{235} \cdot g_a^{235} + N^{238} \cdot \sigma_a^{238} \cdot g_a^{238} + N^0 \cdot \sigma_a^0 + N^0 \cdot \sigma_a^0 + N^0 \cdot \sigma_a^0 + N^0 \cdot \sigma_a^0}
\]

\[
M_{\nu^{235}} = M_{\nu f} = 235.040 \text{ gr/mol}
\]

\[
M_{\nu^{238}} = M_{\nu f} = 238.029 \text{ gr/mol}
\]

\[
M_{\sigma_a} = 31.998 \text{ gr/mol}
\]

\[
\sigma_f^{235} = 570 \text{ barn}, \quad \sigma_a^{235} = 594 \text{ barn}, \quad \sigma_f^{238} = 2.4 \text{ barn}
\]

\[
g_f^{235} = 0.8956, \quad g_f^{238} = 0.9118, \quad g_a^{238} = 1.0198
\]

The resonance escape probability for fast neutrons \((p)\) also is calculated as follows [5]:

\[
\xi = \frac{A}{A + 2} \frac{2}{3}
\]

\[
I_{\text{ef}} = 3.9 \times \left( \frac{\Sigma_f}{N^{238}} \right)^{0.415}
\]

\[
p = e^{\left( \frac{N^{238}}{\xi \Sigma_f} \right)^{I_{\text{ef}}}}
\]

\[
\Sigma_f = N^{235} \cdot \sigma_f^{235} + N^{238} \cdot \sigma_f^{238} + N^0 \cdot \sigma_f^0 + N^H \cdot \sigma_f^H + N^0 \cdot \sigma_f^H + N^0 \cdot \sigma_f^H
\]

If the enrichment of fuel is 100% then the resonance escape probability for fast neutrons \((p)\) will be maximum value.
**The Thermal Parameters and The Rest Parameters**

According to the enrichment of applied fuel and its mass can write [7]:

\[
\rho_{\text{fm}} = \frac{m_{\text{f}}}{M_{\text{fm}}} = \frac{rM_{\text{f}} + (1 - r)M_{\text{mf}} + M_{\text{mf}}}{m_{\text{fm}}} = \frac{rM_{\text{f}} + (1 - r)M_{\text{mf}} + M_{\text{mf}}}{m_{\text{fm}}} R_{\text{f}}
\]

(12)

Where [8]:

\[
m_{\text{UO}_2} = \frac{N_{\text{fm}}}{A} = \rho_{\text{fm}} V_{\text{fm}}
\]

(13)

\[
m_{\text{U}} = m_{\text{f}} = m_{\text{fm}} \cdot \rho_{\text{fm}} V_{\text{fm}} = \rho_{\text{fm}} V_{\text{fm}} \cdot \frac{rM_{\text{f}} + (1 - r)M_{\text{mf}} + M_{\text{mf}}}{m_{\text{fm}}} R_{\text{f}}
\]

(14)

\[
m_{\text{U}} = m_{\text{f}} = \frac{m_{\text{fm}} \cdot f_{\text{fm}}}{m_{\text{fm}}} = \rho_{\text{fm}} V_{\text{fm}} \cdot \frac{rM_{\text{f}} + (1 - r)M_{\text{mf}} + M_{\text{mf}}}{m_{\text{fm}}} R_{\text{f}}
\]

(15)

\[
m_{\text{U}} = m_{\text{f}} = m_{\text{fm}} - m_{\text{f}} = \rho_{\text{fm}} V_{\text{fm}} \cdot \frac{rM_{\text{f}} + (1 - r)M_{\text{mf}} + M_{\text{mf}}}{m_{\text{fm}}} R_{\text{f}}
\]

(16)

\[
\rho_{\text{UO}_2} = \rho_{\text{fm}} = 10960 \text{ kg} / \text{m}^3
\]

For calculation of specific power, power density and the rate of released heat in reaction unit [7]:

\[
S.P = \text{the total power to mass of heavy atoms} = \frac{q_{\text{h}}}{n(\pi R_{\text{fuel}}^2 \cdot dz) \cdot \rho_{\text{pellet}} \cdot f_{\text{fm}}} = \frac{q_{\text{c}}}{\pi R_{\text{f}}^2 \cdot \rho_{\text{smear}} \cdot f_{\text{fm}}}
\]

(17)

Where:

\[
q_{\text{c}} = \frac{q_{\text{h}}}{l}
\]

(18)

\[
\rho_{\text{smear}} = \frac{\pi R_{\text{f}}^2 \cdot \rho_{\text{pellet}}}{(\pi R_{\text{f}}^2 + \delta_{\text{gap}})^2}
\]

(19)

The Fig. 2 shows modeling the fuel rods in each applied fuel assembly in boiling water reactor that has square fuel networks.

Fig. 2 The sample fuel assembly in a boiling water reactor [9]

In a fuel assembly with square modeling, the power density [7]:

\[
P.D = \frac{q}{\text{pitch}^2}
\]

(20)

Also for calculating the power in volume unit of reactor core can write [8]:

\[
q = \frac{q_{\text{h}}}{V}
\]

(21)

In addition to calculate the power in volume unit of fuel \(q_{\text{c}}\) for a cylindrical nuclear reactor [8]:

\[
q_{\text{c}} = q_{\text{h}} \cdot J_0 \left( \frac{2.405 \times 10^{-4} \pi \cdot \rho_{\text{He}}}{R_{\text{f}}} \right)
\]

(22)

\[
q_{\text{c}} = \frac{1}{2 \pi} \int_{r_{\text{f}}}^{r_{\text{f}}} q_{\text{c}} A_{\text{f}} dz = \frac{2}{\pi} q_{\text{c}} A_{\text{f}} H_{\text{f}}
\]

(23)

Thus for calculating \(q_{\text{c}}\) the equation (24) is outcome:

\[
q_{\text{c}} = \frac{q_{\text{h}}}{0.275 n A_{\text{f}} H_{\text{f}}}
\]

(24)

**III. RESULTS AND CONCLUSION**

By using the information of Table 1 and also by application the mentioned formulas and by using the SIXFAC neutronic program, the thermo-neutronic parameters and other parameters of both reactors (Kashiwazaki Kariwa and Forsmark) according to the Figs:3,4 and also Tables:2,3 have been derived.

According to the obtained values for observable neutronic parameters in Table 2 is deduced that the neutronic parameters including \(K_{\text{e}}, K_{\text{ef}}\) and their components in the Kashiwazaki reactor (third generation reactor) for finite numbers of neutrons in neutron cycle are more than the Forsmark reactor (second generation reactor). Thus in the super critical state (without consideration the effect of negative reactivity injection derived of control rod) is concluded that the neutronic parameters amounts in the third generation reactor are more than the second generation reactor in the same power. According to the obtained values of thermo-hydraulic parameters which are shown in Table 2 is perceived that the Kashiwazaki reactor has more volume for core, fuel and fuel mass than the Forsmark reactor. Also the powers in volume unit of core and fuel of Kashiwazaki reactor are less than the Forsmark reactor. Therefore, it can finally be concluded that the performance of the third generation nuclear reactors is safer than the second generation nuclear reactors from the operation aspect.

**REFERENCES**


**TABLE II**

THE CALCULATED AND OBTAINED VALUES FOR NEUTRONIC PARAMETERS

<table>
<thead>
<tr>
<th></th>
<th>via SIXFAC program for Forsmark reactor</th>
<th>via analytical method for Forsmark reactor</th>
<th>via SIXFAC program for Kashiwazaki reactor</th>
<th>via analytical method for Kashiwazaki reactor</th>
<th>The neutronic parameters in super critical state</th>
</tr>
</thead>
<tbody>
<tr>
<td>$K_{eff}$</td>
<td>1.3615</td>
<td>1.3140</td>
<td>1.4230</td>
<td>1.3868</td>
<td>$K_{eff}$</td>
</tr>
<tr>
<td>$K_{\infty}$</td>
<td>1.3888</td>
<td>1.3384</td>
<td>1.4494</td>
<td>1.4039</td>
<td>$K_{\infty}$</td>
</tr>
<tr>
<td>$\eta$</td>
<td>1.7636</td>
<td>1.7500</td>
<td>1.8017</td>
<td>1.7902</td>
<td>$\eta$</td>
</tr>
<tr>
<td>$f$</td>
<td>0.8966</td>
<td>0.8945</td>
<td>0.9143</td>
<td>0.9090</td>
<td>$f$</td>
</tr>
<tr>
<td>$\rho$</td>
<td>0.8254</td>
<td>0.8203</td>
<td>0.8223</td>
<td>0.8205</td>
<td>$\rho$</td>
</tr>
<tr>
<td>$\varepsilon$</td>
<td>1.0636</td>
<td>1.0423</td>
<td>1.0708</td>
<td>1.0515</td>
<td>$\varepsilon$</td>
</tr>
<tr>
<td>$P_{FNL}$</td>
<td>0.9818</td>
<td>0.9826</td>
<td>0.9823</td>
<td>0.9880</td>
<td>$P_{FNL}$</td>
</tr>
<tr>
<td>$P_{THNL}$</td>
<td>0.9989</td>
<td>0.9992</td>
<td>0.9988</td>
<td>0.9998</td>
<td>$P_{THNL}$</td>
</tr>
<tr>
<td>$P_{TNL}$</td>
<td>0.9807</td>
<td>0.9818</td>
<td>0.9811</td>
<td>0.9878</td>
<td>$P_{TNL}$</td>
</tr>
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## TABLE III
THE CALCULATED AND OBTAINED VALUES FOR THERMAL PARAMETERS

<table>
<thead>
<tr>
<th>Forsmark reactor</th>
<th>Kashiwazaki reactor</th>
<th>The thermal parameters and the rest parameters</th>
</tr>
</thead>
<tbody>
<tr>
<td>14.468 m³</td>
<td>18.071 m³</td>
<td>The volume of fuel ( V_{UO_2} )</td>
</tr>
<tr>
<td>5.445 m³</td>
<td>7.206 m³</td>
<td>The volume of clad ( V_{cl} )</td>
</tr>
<tr>
<td>19.914 m³</td>
<td>25.277 m³</td>
<td>The volume of fuel rod</td>
</tr>
<tr>
<td>62.321 m³</td>
<td>77.582 m³</td>
<td>The volume of core</td>
</tr>
<tr>
<td>669.134 m³</td>
<td>831.430 m³</td>
<td>The volume of vessel</td>
</tr>
<tr>
<td>158569.28 kg</td>
<td>198058.16 kg</td>
<td>The total mass of existent UO₂ in reactor core ( m_{f\text{m}} )</td>
</tr>
<tr>
<td>139774.20 kg</td>
<td>174581.72 kg</td>
<td>The total mass of existent U in reactor core ( m_{f} )</td>
</tr>
<tr>
<td>136629.28 kg</td>
<td>170042.60 kg</td>
<td>The total mass of existent U₂³⁸ in reactor core ( m_{f\text{m}} )</td>
</tr>
<tr>
<td>3144.92 kg</td>
<td>4539.12 kg</td>
<td>The total mass of existent U₂³⁵ in reactor core ( m_{f\text{m}} )</td>
</tr>
<tr>
<td>26.469 W/gr</td>
<td>21.194 W/gr</td>
<td>Specific power (S.P)</td>
</tr>
<tr>
<td>83.728 kW/lit</td>
<td>67.940 kW/lit</td>
<td>Power density (P.D)</td>
</tr>
<tr>
<td>59.370 kW/lit</td>
<td>47.691 kW/lit</td>
<td>Power in volume unit of core ( \dot{q}^+ )</td>
</tr>
<tr>
<td>675.631 kW/lit</td>
<td>532.272 kW/lit</td>
<td>Power in volume unit of fuel ( \dot{q}^- )</td>
</tr>
</tbody>
</table>