Inherent safety analysis of the UO$_2$ fueled pebble lattice at the RDE using SRAC2006 module of PIJ

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Abstract. Reaktor Daya Eksperimental (RDE) is type of High Temperature Gas Cooled Reactor (HTGR) with nominal power 10 MW$_{th}$ that uses pebble bed type of sperical fuel. In the development of the RDE core design, as the earliest stage is to study characteristic of the fuel pebble lattice cell, mainly related to the inherent safety. In this study, the inherent safety analysis to the transient effect of fuel temperature and water ingress in the pebble fuel lattice on RDE design core has been done by using Standard Reactor Analysis Code (SRAC2006) Probability I J (PIJ) module. In addition, the nuclide content in the spent pebble fuel lattice was also analyzed. PIJ is code based on the collision probability method applicable to analyze the change of k-inf value when transient occurs at certain fuel burn up point. The analysis was performed to the 17.00% and 8.50% enrichment of UO$_2$ fuel pebble at varying levels of uranium heavy metal loading, i.e 3, 5, 7, and 9 gram. The k-inf value of the fuel pebble lattice has been calculated using branching burn up mode module PIJ through double heterogenity model. Calculation results showed that the higher loading level of heavy metal of uranium then the k-inf value will be smaller. The transient with increasing of the temperature at each particular burn up point or the occurrence of water ingress will also decres in the k-inf value of fuel pebble lattice. From the analysis result, it can be concluded that pebble lattice fueled with 17% enrichment of UO$_2$ on RDE has inherent safety properties i.e. the decreasing of k-inf value if transient temperature occurs. And when there is water ingress, the inherent safety characterisic is shown by pebble lattice with maximum loading of 9 g uranium heavy metal.. The percentage of the nuclide content in spent fuel of pebble latice on RDE is similar to the PWR lattice, so it is not possible for a large number of plutonium production. At the transient by increasing of temperature, there are no significant increasing of gaseous fission product production (Xe-135, I-135, Sm-149, Pm-149).

Keywords: Fuel Lattice, Pebble Bed, TRISO, UO$_2$, RDE, PIJ

1. Introduction
HTGR is a nuclear reactor that uses helium gas and graphite respectively as coolant and moderator. Based on the form of fuel used, the HTGR is divided into 2 types namely Prismatic and Pebble type[1]. Both types of fuel have their respective advantages. In Indonesia, as a means of education, and research on the use of nuclear energy as a source of electricity it is expected to be built an experimental reactor named with Reaktor Daya Eksperimental (RDE). RDE is an HTGR type of
reactor that uses pebble bed type fuel with a nominal thermal power of 10 MW. Selection of HTGR type is the result of research conducted by previous researchers by making the value of weight based on aspects of national strategy and techno-economy. The results show that HTGR pebble bed is superior to the prismatic type[2]. Therefore in Indonesia, the study related to RDE has been started since several years ago. Because the core specification of RDE is almost similar to HTR-10 core, then some researcher using HTR-10 data as a research object. In additional, in Indonesia conceptual design of a new HTGR reactor named Reactor Gas Temperatur Tinggi 200 MWth (RGTT200k) also has been made[3,4].

As an initial stage, some researchers conduct research related to RDE at the normal operation condition, such as the following. The optimum fuel composition for Indonesian Experimental Power Reactor was determined around HM 8g/pebble, and the uranium enrichment is approximately 13%[5]. Neutron sources and dose rate at the first core of RDE in the radial direction in the biological shield for the radiation workers is about 6.69915 μSv/hr (International Commission on Radiological Protection, ICRP), so that they are shielded safe from neutron radiation sources[6]. Estimation of radionuclides discharge from the RDE at the normal operation[7]. Analysis inventory on the experimental high temperature gas reactors at the 10 MW, 20 MW and 30 MW[8]. Inventory analysis of fission products of 10 MW HTGR fuel using ORIGEN2.1 with various literature data[9]. HTGR type reactor fuel performance as a fission product barrier research showed that the fuel integrity is influenced by the operating conditions of the reactor, such as temperature, the level of burn up, fission product activation, energy and fuel quality[10].

Research on the abnormal / transient conditions for HTGR core also has been performed, such as following. Analysis for the kinetic parameters for HTR-10 reactor at the transient condition[11]. Simulation for the Design Basic Accident (DBA) on the HTR-10GT[12]. The change of temperature coefficient on RGTT200k core under water ingress accident condition[13]. Experiment related to fission product release from fuel pebbles in the case of Depressurization and Loss of Forced Circulation (DLOFC) condition[14].

Research on the fuel cell lattice also has been done using MCNP to the thorium-plutonium fuelled pebbles[15]. In the development of the nuclear design of RDE, as the earliest stage, study to characterize of the fuel pebble lattice cell, mainly related to the inherent safety is important to perform. The pebble fuel of RDE core is composed by thousand of the UO$_2$ kernel in the TRISO layer with enrichment of 17 wt% and loading content of 5 g uranium heavy metal. Average of the fuel pebble burn up is 80 GWD/Ton[16].

The purpose of this study is to determine the inherent safety characteristic of the pebble fuel lattice on RDE due to increasing temperature or water ingress at the varying in the uranium heavy metal loading and enrichment of UO$_2$ by using PIJ module. In additional, the nuclide content in the spent fuel pebble also analyzed after reaching fuel burn up of 80 GWD/Ton. Data on nuclide content that has been prepared is the result of calculation with ORIGEN-ARP on fuel at the time of burn up reached 28.37 GWD/Ton[17]. It is also to provide data on the content of nuclides in spent fuel pebble with 80 GWD/Ton burn up which can be used as input in the analysis related to fuel integrity.

The PIJ module is one of the modules in the SRAC2006 based on the neutron transport calculation module that uses the Boltzmann equations neutron collision probability theory in solving[18]. PIJ code is applicable to analyze the the change of k-inf value when transient occurs. In the PIJ module, 107 groups of energy neutrons are used. In this study, calculation of the k-inf as burn up function of fuel pebble lattice cell was done for the normal and transient condition. In the normal conditions temperature was assumed 1200K. Transient condition performed by analysis change of k-inf value due to increasing temperature and water ingress. The analysis was performed to the 17.00% and 8.5% enrichment of UO$_2$ fuel pebble at varying levels of uranium heavy metal loading. The k-inf value of the fuel pebble lattice has been calculated using branching burn up mode modul of PIJ through double heterogenity model.
2. Methodology
In this research, the $k_{inf}$ value and nuclide content of UO$_2$ fueled pebble lattice in the RDE design core has been calculated. The calculation was done using the computer program SRAC2006 module of PIJ. The scenario of the occurrence of the disturbance at a particular burn up point is done by the calculation mode of burn up branching in the PIJ module. The concept of $k_{inf}$ calculation as function of burn up due to temperature or moderator changes with branching mode can be explained as bellow. Illustration of the calculation is shown in Figure 1, at a certain point of fuel burn up, transient condition occurs with temperature changes from $T_{f0}$ to $T_{f1}$ or a moderator density increase from $\rho_0$ to $\rho_1$ occurs. $T_{m0}$ and $T_{m1}$ are moderator temperature at the normal and transient condition, respectively. Value of $k_{inf}$ at the normal condition as function of burn up indicated by a straight line. While the value of $k_{inf}$ at the transient condition of a certain burn up level showed with the a bold point.

![Figure 1. Calculation of $k_{inf}$ value due to fuel burn up using mode branching module of PIJ [15]](image)

To make the input of PIJ module, information data about dimensions and material composition of the TRISO, fuel pebble are needed. The following will be explained how modeling has been done in creating module PIJ input.

2.1. The modeling of TRISO fuel and pebble lattice
The 1-dimensional spherical pebble model with an outer diameter of 6 cm is shown in Figure 2. The pebble lattice is divided into 3 zones with radius sizes such as Table 1. Zone 1 with a diameter of 5 cm where there is a dispersed thousand TRISO fuels on a carbon matrix. Zone 2 with a thickness of 0.5 cm is an area which is without fuel and made of carbon material with a density of 1.73 g/cm$^3$ (impurity 0.125 ppm natural Boron). Zone 3 consist of He gas. In case of water ingress, zone 3 will contain the addition of water with the certain percentage.

![Figure 2. Modeling of the UO$_2$ fueled pebble lattice](image)

| No. | Parameters            | Value of radius (cm) |
|-----|-----------------------|----------------------|
| 1   | Fuel zone             | 2.50                 |
| 2   | Uranium free zone     | 3.00                 |
| 3   | Gas He                | 3.54                 |
As shown in Figure 3 and Table 2, the 1-dimensional model of spherical TRISO composed from UO$_2$ kernel (impurity 4 ppm of natural Boron) with density of 10.4 g/cm$^3$ and 17% of enrichment (diameter 0.05 cm), coating material made from PyC buffer layer with density 1.1 g/cm$^3$ (thickness 9.00E-3 cm), inner PyC layer with 1.9 g/cm$^3$ density (thickness 4.00E-3 cm), SiC layer with 3.18 g/cm$^3$ density (thickness 3.50E-3 cm), PyC layer outer with 1.9 g/cm$^3$ density (thickness 4.00E-3 cm), and outer part is a matrix carbon (C) with standard thickness of 7.78E-2 cm depend on amount of uranium heavy metal loading in the pebble.

![Figure 3. Double heterogeneity model of the TRISO fuel](image)

**Table 2. Data of TRISO fuel [15].**

| No. | Parameters        | Value of thickness/radius (cm) |
|-----|-------------------|-------------------------------|
| 1   | UO$_2$ kernel     | 2.50E-2                       |
| 2   | Buffer Layer      | 9.00E-3 / 3.40E-2             |
| 3   | Inner PyC         | 4.00E-3 / 3.80E-2             |
| 4   | SiC               | 3.50E-3 / 4.15E-2             |
| 5   | Outer PyC         | 4.00E-2 / 4.55E-2             |
| 6   | Matrix C          | 7.78E-2 / 1.23E-1             |

2.2. Calculation of k-inf fuel pebble lattice using module of PIJ

The k-inf calculation is performed on the fuel cell lattice of the pebble at the normal and transient conditions. The transients divided into 2 conditions of disturbance change, increasing temperature, and water ingress.

**Table 3. The cut-off energy neutron to the 16 groups**

| No | Upper  | Lower  | No | Upper  | Lower  |
|----|--------|--------|----|--------|--------|
| 1  | 1.00E+07 | 1.35E+06 | 9  | 3.93E+00 | 1.86E+00 |
| 2  | 1.35E+06 | 8.65E+04 | 10 | 1.86E+00 | 1.28E+00 |
| 3  | 8.65E+04 | 9.12E+03 | 11 | 1.28E+00 | 8.76E-01 |
| 4  | 9.12E+03 | 9.61E+02 | 12 | 8.76E-01 | 4.14E-01 |
| 5  | 9.61E+02 | 1.02E+02 | 13 | 4.14E-01 | 3.42E-01 |
| 6  | 1.02E+02 | 1.07E+01 | 14 | 3.42E-01 | 1.67E-01 |
| 7  | 1.07E+01 | 3.93E+00 | 15 | 1.67E-01 | 5.45E-02 |
| 8  | 3.93E+00 | 1.86E+00 | 16 | 5.45E-02 | 5.98E-03 |

- Normal conditions ie fuel temperature of 300 K and 1200 K. UO$_2$ fuel enrichment used is 17% and 8.5%. While the number of kernel content on the ball pebble varied by 3 g, 5 g, 7 g, and 9 g. Calculation k-inf as function of fuel burn up were performed until 120 GWD/Ton.
Transient caused by increasing temperature has been performed at each level point of fuel burn up with an increase in temperature changes to 2100 K.

Perturbation caused by water ingress, that is changes of k-inf due to water density at the outer part of pebble lattice. At a certain point of burn up suddenly there is filled water with densities of 20%, 40%, 60%, and 80%.

In addition, calculations also performed on the content of nuclides in spent fuel of pebble lattice with 5 g uranium heavy loading, and enrichment UO\(_2\) of 17%. The calculation is performed under normal conditions assuming a fuel temperature of 1200 K, burn up reaches 80 GWD/Ton. The above pebble lattice calculations are performed with JENDL-33 cross-section data. Energy group neutrons conducted condensation from 107 energy groups to 16 groups. The cut-off energy neutron is shown in Table 3.

3. Results and discussions

3.1. Multiplication Faktor (k-inf)

Figure 4 and 5 show the k-inf changes due to fuel burn up at varying levels of uranium heavy metal loading of the pebble lattice fueled by 17% and 8.5% enrichment of UO\(_2\), respectively. From the Figure 4 and 5, it can be seen that the increasing uranium heavy metal loading content, it will produce smaller k-inf value. In the same UO\(_2\)-enriched pebble fuel (17%), as shown in Figure 4, the change in the smaller k-inf values with the greater uranium heavy metal content can be explained as follows. With the same pebble bed size (radius 2.5 cm), increasing the number of kernels (uranium heavy metal) will decrease the volume of the graphite moderator. So that, it affects against after a fission reaction occurs between U-235 and a thermal neutron that produces fast neutrons. Because the number of moderator graphite atoms is reduced, then the number of fast neutrons moderated into thermal neutrons also will be decreased. The other thing, with the same UO\(_2\) enrichment in the pebble lattice, increasing the number of kernels (uranium heavy metal) will increase the number of U-238 atoms. So that the ratio of the number of fast neutrons successfully moderated to thermal neutrons and absorbed by U-238 becomes larger. So the ratio of the number of U-235 absorbed thermal neutrons to the number of fast neutrons produced becomes smaller.

Likewise, for the difference in fuel enrichment levels, the 17% enrichment UO\(_2\) will result in a smaller k-inf value than the 8.5% enriched UO\(_2\) fueled pebble lattice with the same heavy uranium
metal loading level. This is because the fuel with the lower enrichment have the smaller number of U-235 atoms, therefore the ratio of the number of fission reactions that occur will also be less.

From the Figure 4 and 5, it can also be seen the change of k-inf value during the fuel burning process. In the case for fuel pebble with 17% enrichment of UO$_2$, from the beginning (burn up 0 GWD/Ton) to the end (120 GWD/Ton) of fuel burn up shows consistent of k-inf change. The bigger of the uranium heavy metal loading will produce the smaller of the k-inf value as the function of fuel burn up. This is because the greater value of burn up fuel, then the U-235 content is smaller caused by fission reaction with neutrons thermal.

As well as for the enrichment UO$_2$ of 8.5%, unless anomaly occurs on fuel pebble with uranium heavy metal loading of 3 gram. That is a drastic reduction in the value of k-inf at 60 GWD/Ton. This is due to the U-235 content on fuel with 3 g uranium heavy metal loading level that is too little, then it will quickly run out at the burn up level. For the fuels with 7 and 9 g uranium heavy metal loading levels, k-inf value was slightly larger than fuel with a loading of 5 g heavy metal of uranium at burn up level of 100 to 120 GWD/Ton. This is caused by the effect of produced more number of Pu-239 and Pu-241 nuclides.

3.2. Temperature Coefficient
The temperature coefficient changes due to the fuel burn up of fuel pebble lattice with 17% and 8.5% enrichment of UO$_2$ at the varying levels of uranium heavy metal loading showed at Figure 5, and 6, respectively. From both figures, it can be seen that the temperature coefficient is negative for all fuel pebble lattice conditions. It shows that fuel pebble lattice has an inherent safety. If at the point of reaching a certain fuel burn up there is an increase in temperature from 1200 K to 2100 K, then naturally the k-inf will decrease (negative temperature coefficients). From both images can be seen that the greater the uranium content of heavy metal, then the absolute value of coefficient temperature is also greater. Or in other words, it can be said that the fuel with a greater uranium heavy metal content will have an inherent safety properties are getting better. This is due to the absorption reaction at the U-238 resonance area at higher temperatures.

![Figure 6. The temperature coefficient changes to the fuel burn up function of UO$_2$ 17% fueled pebble lattice at varying levels of uranium heavy metal (U HM) loading](image)

![Figure 7. The temperature coefficient changes to the fuel burn up function of UO$_2$ 8.5% fueled pebble lattice at varying levels of uranium heavy metal (U HM) loading](image)

3.3. Effect of The Water Ingress
The k-inf changes due to the fuel burn up of fueled pebble lattice with 5 g, 7 g and 9 g uranium heavy metal at the varying levels of water density showed at the Figure 8a, 8b, and 8c, respectively. From the figure, it can be seen that fuel pebble lattice with 5g, and 7 g of uranium heavy metal, the change in k-
in indicates a smaller value due to the water ingress with $\text{H}_2\text{O}$ density from 0% to 20%, 40%, 60%, and 80%. This is because a number of thermal neutrons will be absorbed by $\text{H}_2\text{O}$. So the greater water content in the outermost layer of pebble lattice, it will result in a smaller $k_{\text{inf}}$ value. At the same burn up level, the greater the heavy metal loading of uranium in the fuel pebble lattice, the smaller $k_{\text{inf}}$ value change will occur.

So that for the pebble lattice with 9 g uranium heavy metal loading, if water ingress with 20% of density shows $k_{\text{inf}}$ value very slightly larger than normal condition (without $\text{H}_2\text{O}$). It shows that the change of $k_{\text{inf}}$ value due to water ingress has reached the maximum value. Therefore, the pebble lattice with uranium heavy metal content greater than 9 g does not meet inherent safety requirements. Because $k_{\text{inf}}$ value does not decrease due to water ingress.

### 3.4 Nuclide content and fission yield in the pebble fuel lattice

Depletion of U-235 and production of Pu to burn up on fuel pebble lattice with 5 g uranium heavy metal, enrichment of 17% showed in Figure 10. From the figure, it can be seen that U-235 is constantly reduced with forming a straight line along with increasing the value of fuel burn up. It shows the condition of the composition of fission, absorption and moderation materials in the fuel pebble lattice are proportional. While for the Pu-239 nuclide, at the beginning of fuel burn up to around 60 GWD/Ton will rises drastically, and after that almost close to constant value. It is because...
at the burn up to 60 GWD/Ton, absorption reaction of U-238 with neutron that results producing in Pu-239 nuclides occurred. While the number of U-235 nuclides still much available, then the fission reaction dominated by U-235. At the fuel burn up more than 60 GWD/Ton, U-235 fuel has been reduced by almost 50%. So the fission reactions also began to occur between thermal neutrons and Pu-239. So that the number of Pu-239 nuclides will slightly decline after burn up reaches 60 GWD/Ton.

**Figure 10.** Depletion of U-235 and production of Pu to burn up on fuel pebble lattice with 5 g uranium heavy metal loading, enrichment of 17%

| Nuclide         | PWR lattice (UO₂ 3.4%) | 3 g  | 5 g  | 7 g  | 9 g  |
|-----------------|------------------------|------|------|------|------|
| Uranium         | 96.81                  | 92.745 | 92.174 | 91.520 | 90.789 |
| Plutonium       | 1.29                   | 0.839 | 1.357 | 1.958 | 2.637 |
| Minor Actinide  | 0.22                   | 0.004 | 0.075 | 0.024 | 0.040 |
| Fission Product | 1.68                   | 6.367 | 6.394 | 6.415 | 6.432 |

Comparison of percentage of nuclide content in the fuel pebble lattice on RDE with various weight of uranium heavy metal loading and PWR lattice showed at Table 4. From that table, the percent weight of nuclide content produced by RDE lattice and PWR lattice showed no significant difference. This is because the RDE and PWR are reactors that utilize the neutron thermal as fission reaction with fissile materials such as U-235, Pu-239, and Pu-241. The difference between the two lattices is in the moderator of material as a neutron moderation from fast energy to thermal that is graphite on RDE and H₂O on PWR. So that spent fuel of pebble lattice on HTGR 10 MW based on RDE design is also not possible to produce Pu with a large number.

Gaseous fission product yield of several nuclides (Xe-135, I-135, Sm-149, and Pm-149) increase due to the transient of temperature indicated at the Table 5a and 5b. From the table, it can be seen that the greater of uranium heavy metal loading, it will show a slight increase in gaseous fission product yield. From the table, it can also be well known that in the transient conditions with increasing of the temperatures from 1200 K to 2100 K, there will be the greatest increase of Sm-149 nuclide fission product yield (12.727% for fuel lattice with 5 g uranium heavy metal). While the gaseous fission product yield of I-135 nuclide did not change (fixed), almost not affected by the transient. Therefore it can be said that at the transient by increasing of temperature, there are no significant increasing of gaseous fission product yield (Xe-135, I-135, Sm-149, Pm-149).
Table 5a. Gaseous fission product yield at the normal and transient condition

| Nuclide | 3 g uranium heavy metal | 5 g uranium heavy metal | Diff. (%) | 1200K | 2100K | Diff. (%) |
|---------|------------------------|------------------------|-----------|--------|--------|----------|
| Xe-135  | 2.99E-03               | 3.11E-03               | 4.013     | 3.20E-03 | 3.34E-03 | 4.375    |
| I-135   | 6.31E-02               | 6.31E-02               | 0.000     | 6.32E-02 | 6.33E-02 | 0.158    |
| Sm-149  | 2.71E-10               | 3.05E-10               | 12.546    | 3.30E-10 | 3.72E-10 | 12.727   |
| Pm-149  | 1.09E-02               | 1.09E-02               | 0.000     | 1.09E-02 | 1.10E-02 | 0.917    |

Table 5b. Gaseous fission product yield increase in the pebble lattice the normal and transient condition

| Nuclide | 7 g uranium heavy metal | 9 g uranium heavy metal | Diff. (%) | 1200K | 2100K | Diff. (%) |
|---------|------------------------|------------------------|-----------|--------|--------|----------|
| Xe-135  | 3.45E-03               | 3.62E-03               | 4.928     | 3.73E-03 | 3.91E-03 | 4.826    |
| I-135   | 6.34E-02               | 6.34E-02               | 0.000     | 6.35E-02 | 6.35E-02 | 0.000    |
| Sm-149  | 4.02E-10               | 4.49E-10               | 11.692    | 4.80E-10 | 5.31E-10 | 10.625   |
| Pm-149  | 1.10E-02               | 1.11E-02               | 0.909     | 1.11E-02 | 1.12E-02 | 0.901    |

4. Conclusion
An analysis of the performance of the fuel pebble lattice on HTGR 10 MW based on RDE design under temperature transient and water ingress conditions has been done by using branch-of calculation burn up mode in the module of PJ model through double heterogeneity model. From the analysis to the calculation result of the pebble lattice fueled with 17% enrichment of UO$_{2}$ on RDE, it can be made some conclusions as follow. The pebble lattice has inherent safety properties i.e. the decreasing of k-inf value due to the transient of temperature. In case of the water ingress accident, the inherent safety characteristic is shown by pebble lattice with maximum loading of 9 g uranium heavy metal. The percentage of the nuclide content in spent fuel of pebble bed lattice on RDE is similar to PWR lattice, so it is not possible for large number of plutonium production. At the transient of temperature, there are no significant increasing of gaseous fission product yield (Xe-135, I-135, Sm-149, Pm-149).

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