Analysis of Fuel Temperature Increase Accident in RDE core

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Abstract. The 10 MW Indonesia’s Experimental Power Reactor (RDE) is a High Temperature Gas Cooled Reactor (HTGR)-type and planned to be operational in the Puspiptek Serpong area in 2022/2023. The reactor applies helium gas coolants, graphite moderator and 17% $^{235}\text{U}$ enrichment fuels and has 8 control fuel rods. To design a nuclear reactor, there are many safety aspects which should be taken into account and one of them is neutronic safety analysis dealing with an accident of reactivity core change due to increases of fuel and moderator temperatures. To begin with the safety analysis, the RDE core should be modeled utilizing the combination of nuclear data libraries and a computer code. Prior to estimate the reactivity core change in the RDE core, the calculation of neutron effective multiplication factors during the accident was accomplished using the MCNPX computer code. In this paper, the neutronic analysis of RDE core has been focused on how the fuel temperature increase implies the reactivity core change in the RDE reactor. By combining the three major world nuclear data libraries and the MCNPX code, the all calculated results showed that all core reactivity changes in the RDE core due to fuel temperature increases starting from 26.85 $^\circ$C (300 K) to 2,726.85 $^\circ$C (3,000 K) are all negatives except those in the reflector zone which principally accumulates generated neutrons always coming from the central of the core during reactor operation. Therefore, those do not affect at all the safety of the reactor core during the accident. Indeed, the RDE reactor core is totally safe in the event of fuel-temperature-increase during its reactor operation and hence the RDE reactor is in a steady, safe operation.

1. Introduction

In the chance of the Plenary Meeting of the National Energy Chamber (DEN) held on June 22, 2016, the President of the Republic of Indonesia commanded the development of power research reactor and supporting laboratories in which Indonesian nuclear experts are able to express, interact, create and support all nuclear research activities in the country. The all significant results can be preserved to strengthen international collaboration and hence always well be informed concerning the world latest nuclear technology. The national long-term planning 2007-2025, the master plan on national industry 2015-2035 as well as the 2014 government regulation on the construction of nuclear power plant in Indonesia also supports the development of research power reactor previously mentioned by the President. Referring to the Globe trends and R&Ds on HTGRs [1, 2], the National Nuclear Energy Agency (BATAN) is firm to advance a High Temperature Gas-cooled Reactor (HTGR), called the...
Indonesia’s experimental power reactor (RDE), to be available in the country. The HTGR technology has been selected, since the HTGR owns not only a very good safety performance, but also the reactor can be utilized for other purposes, such as, hydrogen production, water desalination, coal liquefaction and others [3]. In the future, the similar reactors can also be applied to support small-medium electric power (10 MW to 200 MWe) in the Eastern part of Indonesia [4].

There are design accident scenarios which can be outlined in proposing a nuclear power plant (NPP) design and a nuclear safety inquiry is one of them. Therefore, a neutronic safety investigation is a significant phenomenon that is entitled to be one of design basis accidents (DBA) to be fulfilled in an HTGR design. This paper designates a safety analysis due to the RDE reactivity core change using Monte Carlo MCNPX. A reactivity accident may deal with a sudden and rapid insertion of positive reactivity, for example, a control rod ejection in pressurized water reactors (PWRs) and a control rod drop accident (CRDA) for boiling water reactors (BWRs). Fuel temperatures may hastily increase prompting fuel pellet thermal expansion and hence the reactivity of the reactor core will rapidly augment as well [5]. The RDE operates fuel with the $^{235}$U enrichment of 17% at inlet and outlet temperatures at 250 °C and 700 °C respectively. Referring to R&Ds of HTGRs and RDE, there is a lot of papers to deal with those matters [6-14], however, there has not yet been a safety analysis due to an accident caused of fuel temperature increase in RDE reactor core and hence the reactivity change of the RDE increases.

This paper aims to analyse the core reactivity changes in RDE core caused of fuel temperature increases during an accident. For this accident scenario, fuel temperature increases from a room temperature 300 K to 3,000 K. If fuel temperature increases, the amount of neutrons in RDE fuels will increase as well. Since the amount of neutrons increases, the power of the RDE reactor will go up as well not only in fuel zones, but also in other zones, such as, fuel kernel, coating layer, matrix, shell, dummy, coolant, and reflector and shield. This is due to RDE fuels called the TRISO pebble fuels dealing with those zones. If the production of neutrons increases from time to time in RDE core, the effective multiplication factors ($k_{eff}$) of neutrons in RDE core will change too. Therefore, the change of neutrons in RDE core implies to the reactivity change of the core. However, the reactivity change of the core can be either positive or negative depending on temperature change and types of zones in RDE core.

For the analysis of the previous mentioned accident, MCNPX [15], a well known, widely used Monte Carlo transport code, was expended to model the RDE reactor and the modelling technique used in appraising the reactivity change accident in the RDE core is considered the most representative detail one. Once the RDE reactor core modelling completed, the MCNPX code was focused to estimate the effective multiplication factors of the generated neutron in the RDE core by smearing the world three major nuclear data libraries of ENDF/B-VII, CENDL 3.1 and JEFF-3.1 [16-18]. Only RDE TRISO pebble fuels consisting of 17% $^{235}$U-enrichment were applied. The TRISO Pebble fuels deals with some important areas/zones, namely, fuel kernel, coating layer, matrix, shell, dummy, coolant, and reflector and shield. The accident analysis deals with those areas as well as the existence of coolant, reflector and shielding. While the coolant principally cools the reactor during its operation, the reflector and shielding accumulate all generated neutrons at the edge of the core and hence the all generated neutrons do not leak from the reactor core at all during the reactor operation.

To obtain the accuracy of $k_{eff}$ computations in the RDE reactor core, the amount of 10,000 neutrons per cycle for 50 non-active cycles and 200 active cycles were allowed for. To scrutinize the fuel temperature increase accident in the RDE core utilizing MCNPX, two options of KCODE and KSRC were deliberated to estimate $k_{eff}$ in the core [15]. The results showed that all reactivity changes in the RDE core due to fuel temperature increases starting from 300 K to 3,000 K are all negatives and hence those are not at all to influence the safety of the reactor core during the previously mentioned accident. Indeed, the RDE reactor core is totally safe in the event of fuel-temperature-increase accident during the RDE reactor operation.

2. Methodology of calculation
The RDE is a high temperature gas cooled reactor (HTGR) which would be commissioned in 2027/2028 and the RDE reactor is an extended design of the High Temperature Reactor of 10 MW in China (HTR-
which achieved first criticality in 2000. The RDE is newly developed to enhance design, construction, operation and maintenance and hence enabling to face HTGR commercialisation in Indonesia in the future and the design specification of RDE can be seen in Table 1.

Table 1. Design specification of RDE [4]

| Design specification of RDE                                                                 |
|-------------------------------------------------------------------------------------------|
| Reactor thermal power                                                                     | 10 MW                                      |
| Outlet coolant temperature                                                                | 700°C                                     |
| Inlet coolant temperature                                                                 | 250°C                                     |
| Fuel material                                                                            | UO<sub>2</sub>                              |
| Primary coolant pressure                                                                  | 3.5 MPa                                   |
| Plant lifetime                                                                           | 40 years                                  |
| Volume of active core                                                                    | 5 m³                                      |
| Reactor core diameter                                                                    | 180 cm                                    |
| Reactor core height                                                                      | 197 cm                                    |
| Total steam flow                                                                        | ~4.0 kg/s                                 |
| # Control assemblies at reflector side                                                   | 10                                        |
| # Absorber balls at reflector side                                                       | 7                                         |

The RDE applies graphite moderator and helium gas coolant to maintain inlet/outlet temperatures at 250 °C/700 °C. The ratio between fuel pebbles in the active core and moderator pebble is 57/43. The active core which contains mixed fuel and moderator pebbles is surrounded graphite reflector. The reflector is also blocked by boronated carbon bricks. Inside of the reflector, near to the active core, there are 10 channels with 130 mm diameter for control element insertion, and 7 channels for small ball absorbers, and 3 channels with 130 mm diameter for irradiation. In outlet of reflector, there are 20 channels with 80 mm diameter for helium inlet of the reactor. The used method of multi-pass for core loading method is initially to substitute either moderator pebble or graphite-based dummy balls into the cone at the bottom of the reactor core. The fuel and moderator pebble mixed is gradually loaded until the core achieves firstly critical. The 53% and 47% percentages are respectively assigned for fuel and moderator pebbles. Once the first criticality achieved, the mixture of those pebbles which has the same percentage is further loaded until the full core. At this time it is expected the power of the reactor achieves full power. It is estimated that the full core has a volume of around 5 m³ consisting of around 27,000 pebble fuels elements spread out randomly and they also have a packing fraction of 0.61.

It is noted that each fuel bed consists of 5 grams made from 8,335 TRISO-layer particles in graphite matrix dispersion. Furthermore, each TRISO particle contains UO<sub>2</sub> kernel (density of 10.41 gr/cm<sup>3</sup> and 0.0250 cm radius). The TRISO coating also consists of 4 layers, such as, a porous carbon buffer layer (C) with 1.14 gr/cm<sup>3</sup> density and 0.0340 cm radius. Inner pyrolytic carbon (IPyC) layer has density 1.89 gr/cm<sup>3</sup> and 0.0380 cm radius, while a silicon carbide (SiC) layer has 3.20 gr/cm<sup>3</sup> density and 0.0415 cm radius. The last layer is outer pyrolytic carbon (OPyC) layer which has 1.87 gr/cm<sup>3</sup> density and 0.0455 cm radius. It is noted that specific coating is dedicated to resist and contain gaseous and metallic fission product and it is then able to maintain the integrity of TRISO particle. In addition, a 6-cm-diameter fuel pebble consists of fueled zone with 5 cm diameter in which the composition of graphite matrix contains TRISO and graphite shell with 0.5 cm thick. The graphite shell as a moderator surrounding fuel is
reserved to protect fueled zone during pebble movement. Finally, Figures 1 and 2 deploy a detail-vertical-and-top view RDE core showing fuel kernel to reflector and shield areas in detail.

![Cross section of RDE reactor core](image)

**Figure 1.** Cross section of RDE reactor core

Each fuel bed contains 5 grams formed from 8,335 TRISO-layer particles in graphite matrix dispersion and each TRISO particle consists of UO$_2$ kernel (density of 10.41 gr/cm$^3$ and 0.0250 cm radius). The TRISO coating consists of 4 layers, namely, a porous carbon buffer layer (C) with 1.14 gr/cm$^3$ density and 0.0340 cm radius. While inner pyrolytic carbon (IPyC) layer has density 1.89 gr/cm$^3$ and 0.0380 cm radius, a silicon carbide (SiC) layer has 3.20 gr/cm$^3$ density and 0.0415 cm radius. The last is outer pyrolytic carbon (OPyC) layer having 1.87 gr/cm$^3$ density and 0.0455 cm radius. Specific coating is devoted to resist and contain gaseous and metallic fission product and hence to maintain the integrity of TRISO particle. Fuel pebble with 6 cm diameter consists of fueled zone with 5 cm diameter in which the composition of graphite matrix contains TRISO and graphite shell with 0.5 cm thick. The graphite shell as a moderator surrounding fuel is reserved to protect fueled zone during pebble movement. Figures 1 and 2 deploy a detail-vertical-and-top view RDE core showing fuel kernel to reflector and shield areas in detail.

To begin with the calculation modelling for core reactivity change due to fuel temperature increase in the RDE core, the MCNPX code was applied. The code calculation was begun by modelling TRISO particles in a fuel pebble. While Figure 3 deploys the TRISO particle in a graphite matrix using a simple cubic lattice (SC), Figure 4 shows a pebble model using body centered cubic (BCC) lattice. The modelling of TRISO particle, fuel pebble and RDE core have been implemented based on the transport code of Monte Carlo MCNPX as mentioned previously.
Figure 2. RDE core top-view

Figure 3. Modelling of a simple cubic lattice (SC)
To estimate effective multiplication factors ($k_{\text{eff}}$) of neutrons generated in the RDE core, the calculation chart of the code is as followed:

\begin{figure}
\centering
\includegraphics[width=\textwidth]{calculation_flowchart.png}
\caption{Flow chart of calculation}
\end{figure}

From Figure 5, the algorithm to estimate effective multiplication factor ($k_{\text{eff}}$) in the RDE core is the followings. First, to begin with the calculation, the generation of MCNPX should be firstly implemented and all reactor inputs, such as, core size, fuel enrichment and temperature etc. All those inputs are automatically processed by the code MCNPX. Second, to generate the results, as arranged for defaults, the model of the core, axially and horizontally, and using the KCODE and KSRC codes, the calculation begins. By the combination of the three world nuclear data libraries (ENDF/B-VII.0, CENDL 3.1 and JEFF-3.1.1), $k_{\text{eff}}$ and other physical important matters can then be obtained. Third, variation of fuel temperature should be decided by starting the temperature of 100 °C. One calculation completed, the other estimates of $k_{\text{eff}}$ using other fuel temperatures, such as, 473.15 K up to 3,000 K are implemented. Fourth, for one fuel temperature decided, the $k_{\text{eff}}$ calculation is iterated till the standard deviation for that
calculation achieved. At the end, the calculation is automatically lapsed by the code, when all outputs defined at the beginning of the deviousness have been accomplished.

3. Results and Discussions

For the whole computation, the three major world nuclear data libraries of cross section (XS), namely, ENDF/B-VII.0, CENDL 3.1 and JEFF-3.1 [16-18], have been smeared and the varieties of fuel temperatures have also been applied starting from 300 K up to 3,000 K with the range of 300 degrees. The reason to take the temperature as low as the former is that since the condition is considered as the room temperature in the RDE core. The latter is respected as the highest temperature to be proficient in the fuel during reactor operation.

Prior to show the calculated results for the accident previously mentioned, it is needed to clarify the three world nuclear data libraries. Firstly, ENDF/B-VII.1 library is the latest revision to the United States’ Evaluated Nuclear Data File (ENDF). The revision expands upon that library, including the addition of new evaluated files from 393 to 423. Continuous energy cross section libraries, very suitable for use with the MCNP Monte Carlo transport code, have been applied to a suite of nearly one thousand critical benchmark assemblies defined in the International Related Standards. Secondly, CENDL-3.1 has been tested and improved extensively using the International Benchmark Calculations combined with MCNP4c code. All separately calculated results based on CENDL-2.1, CENDL-3.1 and ENDF/B-VII.0 were compared with the benchmark values. Lastly, JEFF library made by NEA OECD [18] contains sets of evaluated nuclear data which have been tested extensively using benchmark calculations [17]. The calculations were performed with the latest release of the continuous energy Monte Carlo neutronic code MCNP, i.e. MCNP6.

The previously mentioned three world nuclear data libraries, in combination with MCNP6, showed good and the results for the shielding benchmarks are generally good with very similar results for the three libraries in the majority of cases [17]. This paper accomplishes the RDE core \( k_{\text{eff}} \) computation which allows for the fuel temperature increase using the world three nuclear data libraries. In the case of using ENDF/B-VII.0, the cross section (XS) for the temperature of 300 K was joined and the XS in certain ranges of other temperatures can then be accurately approximated benefiting the available extrapolation model in the library. Finally, for the calculation employing CENDL 3.1 and JEFF-3.1, the XS data base reflecting 8 groups of XS and accounting neutron temperatures from 300 K to 3000 K and thermal energy group from \( 2.585 \times 10^{-8} \) to \( 2.585 \times 10^{-7} \) MeV has been established. Scheming of effective multiplication factors \( (k_{\text{eff}}) \) has been adopted for fuel area and its surroundings, such as, fuel kernel, coating layer, matrix of fuel, shell, dummy, coolant and reflector and shielding. For those calculations, dry air and helium gas have been pertained.

Figure 6. Fuel kernel temperature versus \( k_{\text{eff}} \) using the world nuclear data libraries
As previously mentioned that TRISO Pebble fuels deals with some important areas and fuel kernel is very evident consisting of UO$_2$ with the 17\% $^{235}$U enrichment. In this area, when the accident takes place, the accumulation of neutron produced will be maximum. Coating layer as previously defined contains some layers, namely, inner pyrolytic carbon (IPyC), a silicon carbide (SiC) and outer pyrolytic carbon (OPyC). All of these layers are to maintain the integrity of TRISO coated fuels, but there is also another matrix which bonds the TRISO coated fuels to become fuel zone inside pebble. Shell is also an important part of fuel zone which is excised out of the fuel zone. The existence of carbon dummy should be mostly considered since it is very related to moderate fast neutrons to thermal neutrons in the RDE core and hence in case of the accident, those will effectively reduce the amount of generated neutrons in the core. Finally, the accident analysis deals with the existence of the coolant as well as the reflector and shielding. While the former principally cools the reactor during its operation, the latter resists the all neutrons produced at the edge of the core and hence those neutrons do not leak out of the reactor core at all during the reactor operation.

![Graph](image)

**Figure 7.** $K_{eff}$ in the RDE core calculation using nuclear data ENDF/B-VII.0

It is noted from Figures 6, 7 and Table 2, there are two gases considered, namely, dry air and helium. The use of dry air has been done for HTR-10 China to verify the theoretical results obtained, but for the time being and onwards, HTGRs specifically for RDE utilizes helium gas to cool the RDE reactor core.
Table 2. Fuel temperature vs $k_{\text{eff}}$ using the world major nuclear data libraries in 10-group energy

| Temperature (°K) | ENDF/B-VII.0 | CENDL-3.1 | JEFF-3.1 |
|------------------|--------------|------------|-----------|
|                  | Dry air (Helium) | Dry air (Helium) | Dry air (Helium) |
| 300              | 1.18137      | 1.18099     | 1.18246    |
|                  | 1.17517      | 1.17486     | 1.17489    |
| 600              | 1.17262      | 1.17177     | 1.17274    |
|                  | 1.16512      | 1.16597     | 1.16627    |
| 900              | 1.16652      | 1.16473     | 1.16675    |
|                  | 1.15972      | 1.16021     | 1.15877    |
| 1200             | 1.16168      | 1.16055     | 1.16037    |
|                  | 1.15474      | 1.15412     | 1.15335    |
| 1500             | 1.15680      | 1.15734     | 1.15622    |
|                  | 1.15070      | 1.15113     | 1.14968    |
| 1800             | 1.15329      | 1.15254     | 1.15294    |
|                  | 1.14708      | 1.14694     | 1.14610    |
| 2100             | 1.14955      | 1.15031     | 1.15062    |
|                  | 1.14346      | 1.14311     | 1.14364    |
| 2400             | 1.14780      | 1.14659     | 1.14742    |
|                  | 1.14034      | 1.14099     | 1.14089    |
| 2700             | 1.14540      | 1.14479     | 1.14488    |
|                  | 1.13889      | 1.13702     | 1.13789    |
| 3000             | 1.14263      | 1.14313     | 1.14333    |
|                  | 1.13665      | 1.13607     | 1.13540    |
Table 3. Standard deviation for fuel kernel temperature versus $k_{\text{eff}}$ using the different world nuclear data libraries in 10-group energy

| Temperature (°K) | ENDF/B-VII.0 | CENDL-3.1 | JEFF-3.1 |
|------------------|--------------|------------|----------|
|                  | Dry air      | Dry air    | Dry air  |
|                  | Helium       | Helium     | Helium   |
| 300              | 0.00073      | 0.00067    | 0.00067  |
|                  | 0.00069      | 0.00070    | 0.00073  |
| 600              | **0.00077**  | 0.00072    | 0.00070  |
|                  | 0.00071      | 0.00071    | 0.00068  |
| 900              | 0.00071      | 0.00071    | 0.00068  |
|                  | 0.00069      | 0.00069    | 0.00075  |
| 1200             | 0.00071      | 0.00071    | 0.00069  |
|                  | 0.00074      | 0.00068    | 0.00065  |
| 1500             | 0.00069      | **0.00077**| 0.00072  |
|                  | 0.00073      | 0.00065    | 0.00070  |
| 1800             | 0.00069      | 0.00070    | 0.00070  |
|                  | 0.00067      | 0.00068    | **0.00077**|
| 2100             | 0.00069      | 0.00067    | 0.00075  |
|                  | 0.00072      | 0.00065    | 0.00073  |
| 2400             | 0.00073      | 0.00070    | 0.00067  |
|                  | 0.00074      | 0.00073    | 0.00072  |
| 2700             | **0.00077**  | 0.00069    | 0.00067  |
|                  |              | 0.00075    | 0.00073  |
| 3000             | 0.00068      | 0.00068    | 0.00065  |
|                  | 0.00073      | 0.00080    | 0.00067  |

For the $k_{\text{eff}}$ calculation results using the different three world major nuclear data libraries, those are very good or excellent, and it may be concluded that our calculation is very appropriate, since standard deviations as shown in Table 3 are excellent due to only maximum of 0.00077 or only around 0.07%. Furthermore, a difference of $k_{\text{eff}}$ for fuel kernel is just for the energy group 5 (1.29255E-07 MeV), for example, at the temperature of 1,500 K and written bold, the calculated difference is only around 0.007 (1.15622 minus 1.14968) or 0.61% (99.39% accuracy). It may then be concluded that the use of the MCNPX code for the event of reactivity core change due to the temperature increase in the RDE reactor core is very appropriate.

To estimate the reactivity core change ($\Delta \rho$) in the RDE reactor during the event of fuel temperature increase, the following formula of $\Delta \rho$ was then applied:
\[ \Delta \rho = \frac{(k_{n,\text{eff}}^n - k_{0,\text{eff}}^n)}{(k_{n,\text{eff}} x k_{0,\text{eff}})} \times 100\% \]  

(4),

where:

- \( \Delta \rho \) = reactivity change in the RDE core during the accident,
- \( k_{n,\text{eff}}^n \) = effective multiplication factor in the RDE core at condition \( n \) and
- \( k_{0,\text{eff}}^n \) = effective multiplication factor in the RDE core at room temperature.

**Figure 8.** Core reactivity change vs fuel temperature using the world major nuclear data libraries

Regarding reactivity core change, \( \Delta \rho \), in the RDE core as seen in Figures 8 and 9, the estimated results utilizing the three world major nuclear data libraries at fuel kernel and dealing with the fuel temperatures from 300 K to 3000 K are all negatives, and hence the RDE reactor core is in a steady, safe condition. Furthermore, the same computation has been focused on the reactivity core change (\( \Delta \rho \)) at the fuel zone and its surroundings, such as, coating layer, matrix, shell, dummy, coolant and reflector and shield. The results also showed that unlike the reactivity core changes in the reflector and shield zone, those in the other zones are all negatives. It is then understood why the reactivity core changes in the reflector and shield are positive, because the reflector in a nuclear reactor principally accumulates all generated neutrons which comes from the central of the reactor core in which all nuclear fuels are available and a nuclear chain reaction occurs. But, summation of all reactivity core changes should be negative, since the volume of reflector zone is less than that of other zones. Finally, from the safety point of view, although the RDE reactivity core changes in the reflecting, shielding zone is positive, the reactor core is however safe because the reflector zone is far enough, also out of the reactor core and the total of reactivity core changes negative. Indeed, the RDE reactor core is totally safe in the event of fuel temperature increase even up to 3,000 K and hence the RDE reactor is in a steady, safe operation.
4. Conclusions
Accident analysis during fuel temperature increase in the RDE reactor core has been finalized using Monte Carlo code MCNPX. The three world major nuclear data libraries of ENDF/B-VII, CENDL 3.1 and JEFF-3.1 taking into account the temperatures from 300 K to 3,000 K have been utilized for all calculations. To achieve the accuracy of $k_{\text{eff}}$ computation in the RDE reactor core, the amount of 10,000 neutrons per cycle for 50 non-active cycles and 200 active cycles were adopted. To analyze the fuel-temperature-increase accident in the RDE core utilizing the MCNPX code, two options of KCODE and KSRC were considered to estimate $k_{\text{eff}}$ in the core. The calculated results showed that only in reflector and shielding zone, the reactivity core changes are positives, because the reflector in a nuclear reactor principally accumulates all generated neutrons which come from the central of the reactor core in which all nuclear fuels are available and a nuclear chain reaction occurs. Fortunately, from the safety point of view, the reflector zone is far enough and also out of the reactor core. Indeed, the RDE reactor core is totally safe in the event of fuel temperature increase even up to 3,000 K and hence the RDE reactor is in a steady, safe operation.

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