# PERFORMANCE OF EXPERIMENTAL POWER REACTOR COOLING SYSTEM UNDER START-UP CONDITION

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## ABSTRACT

**PERFORMANCE OF EXPERIMENTAL POWER REACTOR COOLING SYSTEM UNDER START-UP CONDITION.** An experimental power reactor has been designed based on high-temperature gas-cooled reactor for experiment purposes. As an experimental reactor, the reactor is able to provide thermal power for various purposes in addition for electricity generation. The reactor is designed to generate 10 MW thermal power from the core cooled by helium gas in the primary cooling system with 700 °C core outlet temperature and cooled by water in the secondary cooling system. The utilization of the thermal energy produced from the reactor is converted to an energy conversion system with a cogeneration configuration. Energy conversion system also serves as a cooling system of the experimental power reactor, which applies indirect cycle. The heat from the primary coolant system is transferred into the secondary coolant system by a heat exchanger acted as a steam generator (SG). The purpose of the paper is to study the performance of the cooling system under start-up condition. Calculation and analysis results related to the thermodynamic parameters and to the cooling system performance are obtained by using CHEMCAD computer code. The calculation shows that if the mass flow rate of the secondary coolant is kept constant at 4.4 kg/second with a constant pressure of 60 bar, the secondary coolant will be entirely in the vapor phase during start-up condition with the the reactor power of 7.5 MW.

**Keywords:** performance, experimental power reactor, cooling system, CHEMCAD

## *ABSTRAK*

***KINERJA SISTEM PENDINGIN PADA REAKTOR DAYA EKSPERIMENTAL SAAT KONDISI START-UP****. Reaktor daya eksperimental adalah reaktor yang didesain berpendingin gas dengan temperatur tinggi untuk keperluan eksperimen. Sebagai reaktor eksperimental, reaktor ini mampu menyediakan energi termal untuk beberapa kegunaan selain untuk pembangkitan listrik. Reaktor ini didesain berdaya termal 10 MW berpendingin gas helium pada sistem pendingin primer dengan temperatur keluaran reaktor sebesar 700 $^{o}$C dan berpendingin air pada sistem pendingin sekunder. Pemanfaatan energi termal yang diproduksi dari reaktor dikonversi melalui sistem konversi energi dengan konfigurasi kogenerasi. Sistem konversi energi juga berfungsi sebagai sistem pendingin terhadap reaktor daya eksperimental yang menerapkan siklus tak langsung. Kalor dari pendingin primer dipindahkan ke pendingin sekunder melalui satu penukar panas yang juga berfungsi sebagai pembangkit uap (steam Generator). Penelitian ini dilakukan untuk mempelajari kinerja sistem pendingin reaktor pada kondisi start-up. Perhitungan dan hasil analisis parameter termodinamika pendingin dan parameter kinerja sistem pendingin diperoleh dengan menggunakan perangkat lunak komputer CHEMCAD. Hasil perhitungan menunjukkan bila laju alir massa pendingin sekunder dipertahankan sebesar 4,4 kg/detik dengan tekanan tetap 60 bar, maka pendingin sekunder akan berada pada fase uap seluruhnya yang terjadi pada kondisi start-up dengan daya reaktor 7,5 MW.*

***Kata kunci****: kinerja, reaktor daya eksperimental, sistem pendingin, CHEMCAD*

## INTRODUCTION

Research and development in energy technology in the world are focused on the development of new and renewable energy, because the availability of conventional energy derived from fossil is uncertain in the long term. One option of new and renewable sources of energy is nuclear energy. The development of nuclear energy enhances its role in supplying the world's energy demands [1,2]. Increasing the role of nuclear technology as an energy supplier will encourage research activities and technological development of nuclear reactor systems. Along with the development of generation IV reactors, nuclear reactor systems development leads to the concept of cogeneration systems [3,4]. Cogeneration system is intended to optimize the use of nuclear energy for generating electricity and supplying thermal energy for industry [5].

The concept of cogeneration is developed within the framework of nuclear energy systems (NES). With the concept of NES, a nuclear reactor is not merely as a power plant but also serves as a supplier of thermal energy such as for hydrogen production for desalination as well as a for other industrial processes [6].

There are various types of generation IV reactors, but the High-Temperature Gas-cooled Reactor (HTGR) is most suitable for the application of the concept of cogeneration because the temperature out of the reactor is very high [7]. Various studies on HTGR cogeneration system have been developed, for example as reported by Hiroyuki Sato [8] that GTHTR300C cogeneration system can be used as power generation, hydrogen production and desalination of sea water. While Gustavo Alonso et al. [9] reported that the cogeneration system at the Pebble Bed Modular Reactor (PBMR) is able to generate electricity and provide thermal energy to produce petrol (gasoline) in the oil refinery.

Related to the above description, Experimental Power Reactor (Reaktor Daya Eksperimental = RDE) is an experimental reactor that is designed based on the high temperature gas cooled reactor (HTGR) with low power. This experimental power reactor is being designed and will be built inside the Puspiptek area in Serpong. As an experimental reactor, the RDE is also designed to be capable of supplying electrical energy to the internal area of BATAN. The energy conversion systems of experimental power reactor are designed with a configuration of indirect cogeneration cycle. The RDE uses helium gas as the primary coolant and water as the secondary coolant. The thermal power of RDE is designed to generate 10 MW with a reactor outlet temperature of 700 $^{o}$C [10].

This study aims to analyze the performance of the experimental power reactor under start-up condition. The analysis was conducted by a simulation using a computer program package CHEMCAD. CHEMCAD is a package of computer programs that have been widely used for the design and analysis of process systems [11,12].

**NUCLEAR STEAM SUPPLY SYSTEM OF RDE**

Conceptually, RENUKO as a consultant organization for BATAN was responsible to propose the design of RDE. Referring to the RENUKO document about the RDE design, the schematic of coolant flow in the nuclear steam supply system is shown in Fig. 1 [13].

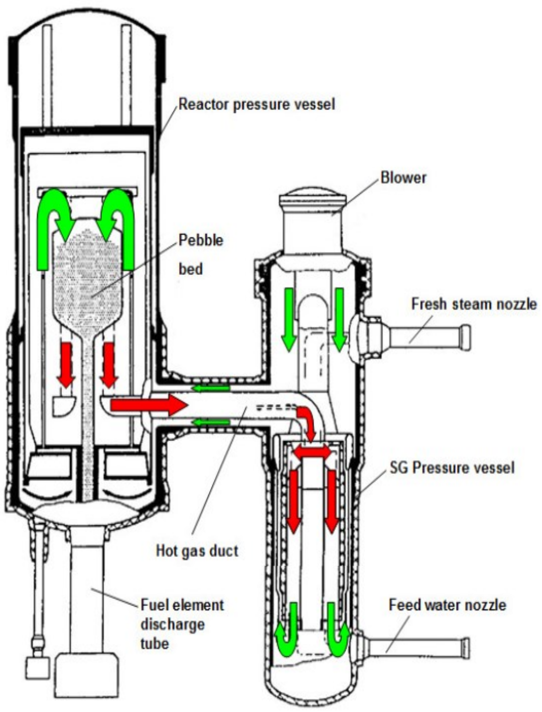


Fig. 1. Schematic Flow Diagram of Nuclear Steam Supply System [13]

The nuclear steam supply system of RDE is referring to the concept of HTR-PM [14]. In this design, the pressure vessel of steam generators is placed lower than the reactor pressure vessel. That is intended to reduce the chances of water intrusion into the reactor. Water intrusion into the reactor or water ingress and air ingress is the condition included in beyond design basis accidents, which is considered in the design of high temperature gas-cooled reactors [15]. The water intrusion into the reactor will affect the structure of the reactor because the biggest part of the structure of the reactor is graphite, while the graphite is highly corrosive if exposed to water [16].

The schematic of flow diagram of the nuclear steam supply system or the secondary system of the RDE as listed in the RENUKO document [17] is shown in Fig. 2. In this design, both the reactor vessel and steam generator vessel (SG) are placed in the reactor building. Hot steam flows out of steam generator to the steam turbine and then flows into the condenser. In the condenser, the steam is condensed into water and is pumped into the feed water tank, which is then pumped back to the Steam Generator.

In the secondary cooling system of RDE, two turbines are installed. There are high-pressure turbine and low-pressure turbine. The low-pressure turbine receives steam from the high-pressure turbine output. In the inlet pipe of the high-pressure turbine, a branch pipe is installed to flow steam into the feed water tank for purposes of reactor start-up and shutdown. The same branch is made to drain the steam into the feed water tank prior entering the low-pressure turbine.

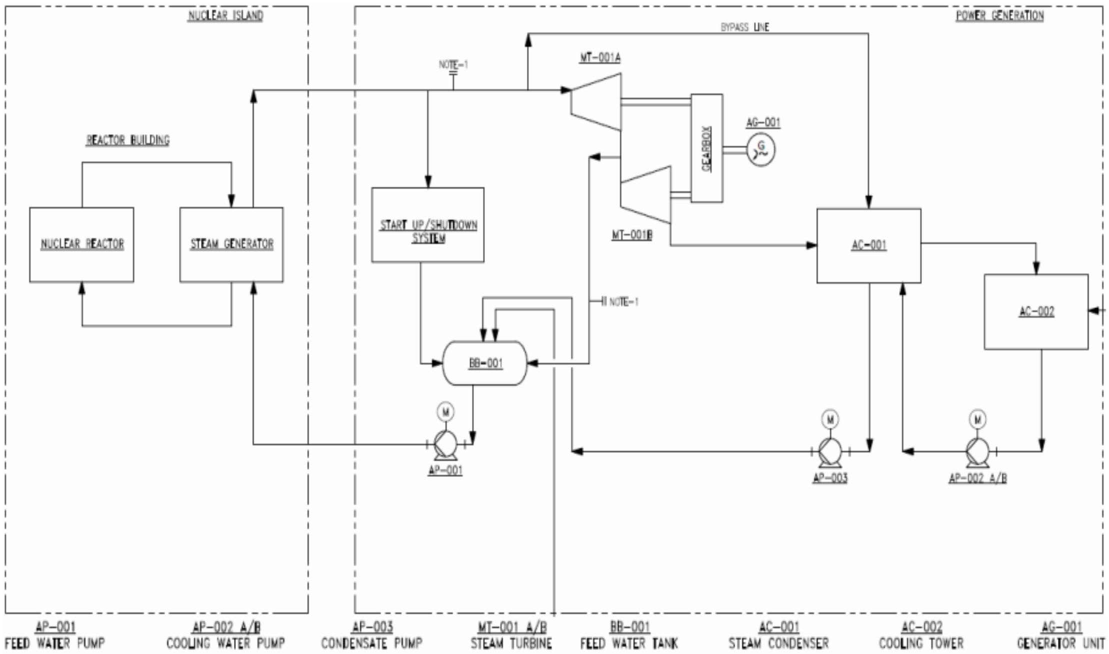


Fig. 2. Schematic of RDE cooling system [18]

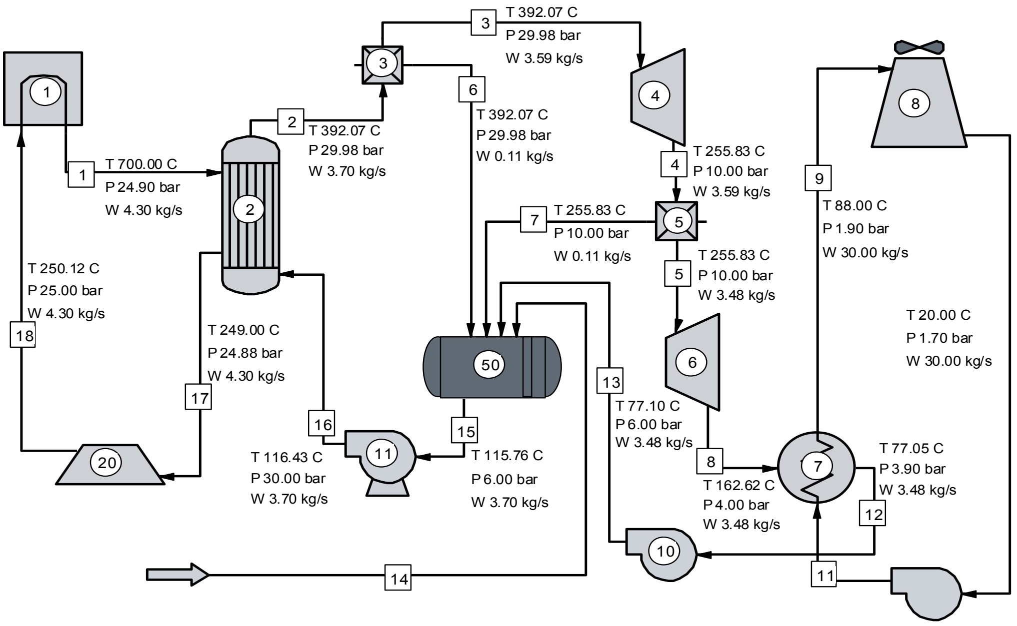


Fig. 3. Model of RDE cooling system using CHEMCAD

**METHODOLOGY**

Performance analysis of the RDE cooling system is carried out by using computer code CHEMCAD. The RDE cooling system consisting of a primary cooling system and secondary cooling system shown schematically in Fig. 2 is modeled using computer code CHEMCAD as illustrated in Fig. 3. Tertiary cooling systems (AC-002) is modeled as a cooling tower to remove heat from the secondary system.

Helium gas as cooling fluid in the primary cooling system is circulated using a compressor mounted on the inlet of the reactor. The steam generator is installed between the primary and secondary cooling system that also serves as an intermediate heat exchanger. The steam generator is modeled as a shell and tube type heat exchanger. Feedwater tank (BO-001) is modeled as a flash tank.

As the input data for simulation is the parameter of each component depicted from the RENUKO design document [10]. The data of reactor is shown in Table 1, the data of pressure vessel unit in Table 2, and the data of steam generator in Table 3. All data parameters were used as input parameters for simulation using CHEMCAD code.

Table 1. The Data of Reactor [17]

| **Parameter** | **value** | **unit** |
|---|---|---|
| Reactor power (thermal) | 10 | MW |
| Mean power density | 2 | $MW/m^3$ |
| Core diameter | 1.8 | m |
| Mean core height | 2.0 | m |
| Primary system pressure | 30 | bar |
| Primary coolant temperature (inlet/outlet) | 250/700 | $^oC$ |

Table 2. The Data of Pressure Vessel Unit [17]

| **Parameter** | **value** | **unit** |
|---|---|---|
| Design pressure | 40 | bar |
| Nominal operating pressure | 30 | bar |
| Reactor pressure vessel height | 10.8 | m |
| Reactor pressure vessel inside Ø | 4.5 | m |
| SG pressure vessel height | 12.35 | m |
| SG pressure vessel inside Ø | 1.5 | m |
| Gas duct pressure vessel inside Ø | 700 | mm |

Performance simulation of the RDE cooling system is carried out under constant pressure and the mass flow rate at the primary and secondary cooling system are fixed. External air temperature in this simulation such as the temperature of water cooling flows out of the cooling tower is 28 °C. The thermal power of reactor is simulated to increase gradually from 0.5 to 10 MW. For each thermal step of 0.5 MW, the reactor was maintained in a stable condition and the pressure is constant. At the stable condition for each step, some thermodynamic parameters of the cooling system are recorded such the reactor inlet and outlet temperature, the primary side of steam generator inlet and outlet temperature, and the secondary side of steam generator inlet and outlet temperature.

Table 3. Data of Steam generator [17]

| **Parameter** | **value** | **unit** |
|---|---|---|
| Thermal power | 10.0 | MW |
| Primary coolant mass flow | 4.4 | kg/s |
| Primary coolant inlet temperature | 700 | $^oC$ |
| Primary coolant outlet temperature | 245 | $^oC$ |
| Primary coolant inlet pressure | 30 | bar |
| Mass flow rate of steam | 4.0 | kg/s |
| Main steam temperature | 530 | $^oC$ |
| Feed water temperature | 160 | $^oC$ |
| Main steam pressure at SG outlet | 60 | bar |
| Number of tubes | 93 | |
| Tube outside diameter (OD) | 23 | mm |
| Heat transfer area | 70 | $m^2$ |
| Vessel diameter | 1.5 | m |
| Vessel height | 12.35 | m |

## RESULT AND DISCUSSION

The results of the simulation using CHEMCAD are shown in Table 4, Fig. 4 and Fig. 5. In this simulation the primary coolant pressure is 30 bar and the mass flow rate of the primary coolant is 4.4 kg/second, whereas the secondary coolant pressure is 60 bar with the secondary mass flow rate of 4.0 kg/second. Table 4 shows the change in temperature at inlet and outlet of the reactor and at inlet and outlet of steam generator. As shown in Fig. 4 and Table 4, each increase in reactor's thermal power will affect the reactor outlet temperature when the pressure and mass flow rate of coolant are kept constant. The previous studies show that the changes of coolant mass flow rate will affect the temperature of the reactor outlet [19, 20]. In this simulation, the outlet temperature of reactor is still less than 700 $^oC$ at the reactor power of 10 MWt.

Table 4. The temperature of inlet and outlet reactor and steam generator for a various thermal power reactor

| No | Power Reactor (MWt) | Reactor temperature ($^oC$) | | Temperature of SG at primary system ($^oC$) | | Temperature of SG at secondary system ($^oC$) | |
|---|---|---|---|---|---|---|---|
| | | Inlet | Outlet | Inlet | Outlet | Inlet | Outlet |
| 1 | 0.5 | 77.44 | 99.32 | 99.32 | 57.35 | 36.80 | 94.17 |
| 2 | 1.0 | 92.00 | 135.76 | 135.76 | 71.07 | 39.76 | 127.71 |
| 3 | 1.5 | 106.13 | 171.75 | 171.75 | 84.40 | 42.78 | 160.63 |
| 4 | 2.0 | 119.64 | 207.12 | 207.12 | 97.14 | 45.86 | 192.66 |
| 5 | 2.5 | 132.38 | 241.73 | 241.73 | 109.14 | 49.06 | 223.55 |
| 6 | 3.0 | 147.30 | 278.53 | 278.53 | 123.21 | 55.62 | 255.95 |
| 7 | 3.5 | 153.60 | 306.53 | 306.53 | 129.14 | 63.22 | 274.57 |
| 8 | 4.0 | 150.69 | 325.56 | 325.56 | 126.41 | 72.06 | 274.57 |
| 9 | 4.5 | 151.74 | 348.60 | 348.60 | 127.39 | 82,20 | 274,57 |
| 10 | 5.0 | 156.26 | 374.92 | 374.92 | 131.65 | 93.54 | 274.57 |
| 11 | 5.5 | 163.64 | 404.17 | 404.17 | 138.62 | 106.11 | 274.57 |
| 12 | 6.0 | 173.38 | 435.90 | 435.90 | 147.80 | 119.85 | 274.57 |
| 13 | 6.5 | 185.27 | 469.68 | 469.68 | 159.00 | 134.67 | 274.57 |
| 14 | 7.0 | 198.99 | 505.17 | 505.17 | 171.94 | 150.47 | 274.57 |
| 15 | 7.5 | 214.39 | 542.54 | 542.54 | 186.46 | 167.37 | 274.57 |
| 16 | 8.0 | 236.88 | 586.82 | 586.82 | 207.66 | 184.43 | 319.03 |
| 17 | 8.5 | 239.48 | 611.34 | 611.34 | 210.11 | 179.86 | 354.38 |
| 18 | 9.0 | 242.11 | 635.85 | 635.85 | 212.59 | 173.81 | 391.26 |
| 19 | 9.5 | 246.69 | 662.33 | 662.33 | 216.91 | 167.76 | 431.06 |
| 20 | 10.0 | 253.67 | 691.17 | 691.17 | 223.49 | 162.06 | 473.55 |

Fig. 4 shows the change of the inlet and outlet temperature of the reactor due to the increase of the reactor thermal power. Along with the heat generation in the reactor core, the reactor outlet temperature increases. The thermal energy of the primary system is used to generate steam in the steam generator.

The characteristics of temperature in the inlet and outlet of the reactor and the steam generator are shown in Fig. 5. At reactor thermal power below 3.5 MWt, the thermal energy is not enough to evaporate the water in the secondary side of the steam generator because the water pressure on the secondary side of steam generator is 60 bar. At that power, the evaporation is just taking place, because the thermal energy from the primary side is used to evaporate water at the latent heat temperature on the secondary side of the steam generator. Therefore, the fluid coming out of the steam generator is still a mixture of liquid and vapor. Because of the latent heat, the outlet temperature of steam generator remains unchanged if the power is increased. Outlet temperature begins to rise after entire water in the steam generator has turned into the steam. All fluids coming out from the steam generator will be in steam phase, when the reactor power reaches 7.5 MWt. Therefore, at the start-up condition, the cooling system configuration of RDE must change or it can also be done for the turbine bypass process.

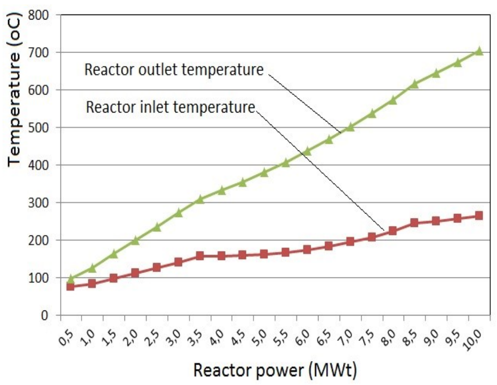


Fig. 4. Temperature of inlet and outlet reactor as function of reactor power

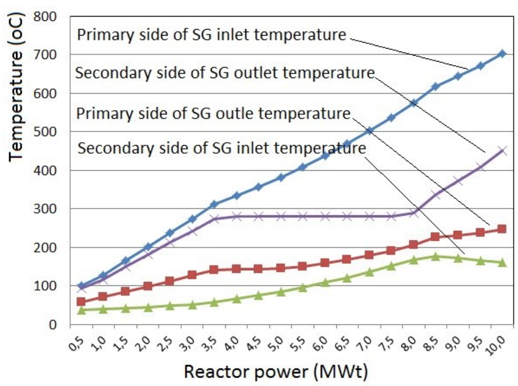


Fig. 5. Steam generator inlet and outlet temperatures on the primary and secondary side as function of reactor power

Table 5. Comparison of reactor and steam generator data from RENUKO and CHEMCAD calculation

| **Reactor** | | Renuko | Model | % Dif |
|---|---|---|---|---|
| 1 | Thermal power reactor, MWt | 10 | 10 | |
| 2 | Primary coolant mass flow rate of He, kg/s | 4.4 | 4.4 | |
| 3 | Primary coolant pressure, bar | 30 | 30 | |
| 4 | Temperatur inlet reaktor, oC | 250 | 253.67 | 1.5 |
| 5 | Temperatur outlet reaktor, oC | 700 | 691.16 | 1.3 |
| **Steam Generator** | | | | |
| 6 | Primary coolant mass flow rate of He, kg/s | 4.4 | 4.4 | |
| 7 | Primary side of inlet steam generator temperature, $^oC$ | 700 | 691.16 | 1.3 |
| 8 | Primary side of outlet steam generator temperature, $^oC$ | 245 | 223.49 | 8.8 |
| 9 | Primary coolant pressure, bar | 30 | 30 | |
| 10 | Feedwater temperature, $^oC$ | 160 | 162.06 | 1.3 |
| 11 | Main steam pressure, $^oC$ | 530 | 473 | 10.8 |
| 12 | Secondary coolant pressure, bar | 60 | 60 | |

Table 5 displays the comparison of reactor and steam generator data between the RDE design from the RENUKO document and the simulation results using CHEMCAD. In this simulation, the mass flow rate and the pressure of coolant are the two parameters that are set in accordance with the values given in the RENUKO document. The biggest differences occur in the outlet temperature of the steam generator due to the differences in the design of steam generator. Therefore, an improvement in the design of steam generator is required.

## CONCLUSION

Simulation to show the performance of experimental power reactor coolant system for start-up conditions has been done. The simulation is carried out by keeping the coolant mass flow rate and the pressure. By increasing the reactor power from 0.5 MW to 10 MW, the behavior of the coolant system has been investigated. It shows that the changes in the thermal power reactor will consequently change the inlet and outlet temperature of the steam generator at the primary side. If the mass flow rate of the secondary coolant is constant at 4.4 kg/sec with a constant pressure of 60 bar, the secondary coolant will be entirely in the vapor phase after the reactor power achieves 7.5 MWt. Therefore, for the start-up condition the cooling system pressure should be reduced according to the increase of reactor thermal power.

## REFERENCES


1. S. GRAPE et al, “New perspectives on nuclear power-Generation IV nuclear energy systems to strengthen nuclear non-proliferation and support nuclear disarmament”, Energy Policy, vol. 73, pp. 815–819, 2014.
2. G. LOCATELLI, M. MANCINI, and N. TODESCHINI, “Generation IV nu-clear reactors: Current status and future prospects,” Energy Policy, vol. 61, pp. 1503–1520, 2013.

3. H. SATO, X. L. YAN, Y. TACHIBANA, and Y. KATO, "Assessment of load-following capability of VHTR cogeneration systems," Annals of Nuclear Energy, vol. 49, pp. 33–40, 2012.
4. P. VIVEK, P. VIJAYA, "Heat Recovery Steam Generator by Using Cogeneration", International Journal of Engine Research, vol. 3, no. 8, pp. 512–516, 2014.
5. X. YAN et al, "A hybrid HTGR system producing electricity, hydrogen and such other products as water demanded in the Middle East", Nuclear Engineering and Design, vol. 271, pp. 20–29, 2014.
6. M. F. RUTH et al, "Nuclear-renewable hybrid energy systems: Opportunities, interconnections, and needs," Energy Conversion and Management, vol. 78, pp. 684–694, 2014.
7. J. R. GESCHWINDT et al, "Performance and optimization of an HTR cogeneration system", Nuclear Engineering and Design, vol. 251, pp. 297–300, 2012.
8. H. SATO et al, "GTHTR300—A nuclear power plant design with 50% generating efficiency", Nuclear Engineering and Design, vol. 275, pp. 190–196, 2014.
9. G. ALONSO, R. RAMIREZ, R. CASTILLO, "Process heat cogeneration using a high temperature reactor", Nuclear Engineering and Design, vol. 280, pp. 137–143, 2014.
10. RENUKO, "The Document Preparation of Preliminary Engineering Design of The Experimental Power Reactor 'Conceptual Design of Water/Steam Cycle' (RDE/ DS-WBS02-207)," 2015.
10. K. SOLANKI, N. M. PATEL, "Process Optimization Using CHEMCAD", International Journal of Futuristic Trends in Engineering and Technology, vol. 1, no. 2, pp. 47–51, 2014.
11. C. CORMOS, "Hydrogen and power co-generation based on coal and biomass/solid wastes co-gasification with carbon capture and storage," International Journal of Hydrogen Energy, vol. 37, no. 7, pp. 5637–5648, 2012.
12. RENUKO, "The Document Preparation of Preliminary Engineering Design of The Experimental Power Reactor 'Conceptual Design of Nuclear Steam Supply System' (RDE/DS-WBS02-201), 2015.
13. Y. ZHOU et al, "Thermal hydraulic simulation of reactor of HTR-PM based on thermal-fluid network and SIMPLE algorithm", Progress in Nuclear Energy, vol. 62, pp. 83–93, 2013.
14. Z. YANHUA, C. FUBING, S. LEI, "Analysis of diffusion process and influence factors in the air ingress accident of the HTR-PM", Nuclear Engineering and Design, vol. 271, pp. 397–403, 2014.
15. J. J. LEE, T. K. GHOSH, S. K. LOYALKA, "Oxidation rate of nuclear-grade graphite IG-110 in the kinetic regime for VHTR air ingress accident scenarios", Journal of Nuclear Materials, vol. 446, no. 1–3, pp. 38–48, 2014.

17. RENUKO, “The Document Preparation of Preliminary Engineering Design of The Experimental Power Reactor ‘Conceptual Design of Steam Turbine and Auxiliaries’ (RDE/DS-WBS02-208), 2015.
18. RENUKO, “The Document Preparation of Preliminary Engineering Design of The Experimental Power Reactor 'Steam and Water Cycle' (RDE-DS-WBS02-207), 2015.
19. I. D. IRIANTO et al, “Thermodynamic Analysis On Rankine Cycle Steam For Cogeneration Systems RGTT200K,” in Prosiding Seminar Nasional Teknologi Energi Nuklir , pp. 865–872, 2016.
20. I. D. IRIANTO, “Design And Analysis Of Helium Brayton Cycle For Energy Conversion System Of RGTT200K”, Jurnal Teknologi Reaktor Nuklir Tri Da-sa Mega, vol. 18, no. 2, pp. 75–86, 2016.