A PC-based high temperature gas reactor simulator for Indonesian conceptual HTR reactor basic training

Syarip\(^1\)\(^*\) and L C C Po\(^2\)

\(^1\)Centre for Accelerator Science & Technology, Yogyakarta, Indonesia.
\(^2\)Micro-Simulation Technology 10 Navajo Court, Montville, New Jersey 07045, USA

\(^*\)Corresponding author’s email: syarip@batan.go.id

Abstract. In planning for nuclear power plant construction in Indonesia, helium cooled high temperature reactor (HTR) is favorable for not relying upon water supply that might be interrupted by earthquake. In order to train its personnel, BATAN has cooperated with Micro-Simulation Technology of USA to develop a 200 MWt PC-based simulation model PCTRAN/HTR. It operates in Win10 environment with graphic user interface (GUI). Normal operation of startup, power maneuvering, shutdown and accidents including pipe breaks and complete loss of AC power have been conducted. A sample case of safety analysis simulation to demonstrate the inherent safety features of HTR was done for helium pipe break malfunction scenario. The analysis was done for the variation of primary coolant pipe break i.e. from 0,1% - 0,5 % and 1% - 10 % helium gas leakages, while the reactor was operated at the maximum constant power of 10 MWt. The result shows that the highest temperature of HTR fuel centerline and coolant were 1150 °C and 1296 °C respectively. With 10 kg/s of helium flow in the reactor core, the thermal power will back to the startup position after 1287 s of helium pipe break malfunction.

1. Introduction
The National Nuclear Energy Agency (BATAN) is initiating for development of Experimental Power Reactor or RDE which is an HTR type, with the objective, among others: to demonstrate a small NPP which operates safely, to conduct a program of integrated research and development of new energy and renewable energy, to increase mastery of NPP technology in the field of design, construction and operation and maintenance [1]. By demonstrating the RDE therefore it is expected that the trust of the public towards the security in operation of the power reactor and the capability of the Indonesia nation in operating the nuclear reactor would increase. The RDE could be used for electricity generation, heat generation and for production of hydrogen. A functional simulator of this type of reactor should be developed as a tool for understanding the RDE operation.

Micro-Simulation Technology had developed a PC-DOS functional simulator of the 30-MW High Temperature Test Reactor (HTTR) for Japan Atomic Energy Research Institute (JAERI or the current Japan Atomic Energy Agency JAEA) at Oarai, Japan in 1998 [2,3]. In 2015, BATAN issued specifications of a simulator of the maximum power of 200 MWt (10 MWe \(\leq P \leq \sim 60\) MWe) with: neutron flux and reactor power variations, plant efficiency, electric power, load factor, core power density and heat flux. Reactor coolant gas conditions: flow rate, pressure, temperatures at core inlet-
outlet, inlet-outlet steam temperatures, steam pressure & temperature at turbine inlet, feedwater temperature, condenser pressure, steam mass flow, all for various reactor power levels. Gas coolant radioactivity and fuel temperature for various reactor power levels. The parameters are displayed graphically and numerically can be saved in output files.

The PCTRAN/HTR functional simulator as a tool for understanding operation and mechanism work flow of RDE. The simulator is in-line with the need of Nuclear Training Centre (NTC) which is proposed to be located at BATAN Yogyakarta [4,5]. The NTC is a training program in the field of nuclear and applied physics particular in reactors utilization, for educational institutions and research personnel in the research activities in the field of ionizing radiation and reactor technology [6,7]. In this paper normal power maneuver and the most severe loss of helium circulation without scram event are presented to show RDE or HTR’s upmost safety among all nuclear power reactors.

Beside functional simulator, it is also important to have an engineering simulator. The engineering simulator was applied to simulate the behavior of HTR type under steady-state operation, startup and shutdown processes, and accident conditions [8,9]. The coupling effect during the condition conversion process and the thermal characteristics under accident conditions of HTR-PM were analyzed by this simulator. In addition, the areas for the design, development, and licensing of HTRs are the verification and validation of analysis tools from the core physics, thermal hydraulics, and heat transfer [10,11]. The PCTRAN/HTR can be used as a preliminary step in those verifications and validation analyses.

The theory models include reactor kinetics, thermal hydraulics for the reactor coolant and containment and plant control systems.

A point reactor kinetics model with one delayed neutron group and reactivity control from external sources (e.g. control rods and boron injection) and feedback from moderator temperature, Doppler effects, and void was formulated. The point-kinetics equation is expressed by a point kinetics model with six delayed neutron groups and reactivity control from external sources and feedback was formulated. They are expressed by equation (1) and (2):

\[
\frac{dC_i}{dt} = \beta_i n - \lambda_i C_i
\]  

\[
\frac{dn}{dt} = \frac{\rho - \rho^0}{\lambda} n + \sum_{i=1}^{6} \lambda_i C_i + S
\]

where: \( n \) = neutron density, \( \rho = \text{reactivity} = (k -1) / k \), \( k \) = effective multiplication factor, \( \beta_i \) = delayed neutron fraction for the \( i^{th} \) group, \( \lambda \) = neutron life time, \( \lambda_i \) = decay constant for the \( i^{th} \) group, \( C_i \) = precursor concentration, \( S \) = neutron source.

The six-group point kinetics equations are solved by finite difference method in the program. Reactivity is controlled by rod movement and boron concentration adjustment. Its feedback in moderator’s density and fuel temperature (Doppler) effects are considered, and the reactivity corrections in the model are calculated as (3):

\[
K_c = K_0^0 \left( \frac{1 - \Delta K_B}{K_0^0} \right) \left( \frac{1 - \Delta K_{DOP}}{K_0^0} \right) \left( \frac{1 - \Delta K_{Xe}}{K_0^0} \right) \left( \frac{1 - \Delta K_{Sm}}{K_0^0} \right)
\]  

where: \( K_c \) = the corrected infinite neutron multiplication factor of the core, \( K_0^0 \) = the uncorrected \( K_0 \) of fuels (combination results of rod controlled and uncontrolled \( K_0 \)) in the core. \( \Delta K_B \) = the reactivity correction of core due to boron, \( \Delta K_{DOP} \) = the reactivity correction of fuel temperature in the core, \( \Delta K_{Xe} \) = the reactivity correction of core due to Xenon-135, and \( \Delta K_{Sm} \) = the reactivity correction of core due to Samarium-149.

Neutron density is directly proportional to the fission power, the core power is provided by fission product decay heat (Q_{DH}) and it is given by an eleven group equation (4):

\[
Q_{DH} = \sum_{i=1}^{11} E_i \exp(\lambda_i t)
\]
where: $E_j$ = amplitude of $j$-th term, $\lambda_j$ = decay constant of $j$-th term, and $t$ = elapsed time since shutdown.

Concentration of each decay group $\gamma_j$ is represented by (5)

$$\frac{d\gamma_j(t)}{dt} + \lambda_j \gamma_j(t) = E_j n(t) \quad (5)$$

The total power in the core is given by (6)

$$P(t) = P_0 \left( n(t) E_f + \sum_{i=1}^{11} \lambda_i \gamma_i \right) \quad (6)$$

where $E_f$ is about 0.93, i.e. fission heat is about 93% of total core power.

The basic thermal hydraulics of the mathematical model is first-principle in mass and energy balance. This ensures credible and realistic simulations. For a gas-cooled reactor, the primary and secondary helium systems are modelled by two control volumes described by the ideal gas law:

$$PV = N R \quad (7)$$

where : $P$ = pressure, $V$ = volume, a constant for either the primary or secondary gas system, $N$ = mole number, $R$ = gas constant, and $T$ = absolute temperature. The pressure and temperature transient during a transient are then governed by the simple derivative of the above equation and mass and energy input to the system as follows (8):

$$\frac{dP}{dt} = \left( \frac{dN}{dt} \right) RT + NR \frac{dT}{dt}$$

$$\frac{dN}{dt} = \frac{1}{M_a} (W_{in} - W_{out}) \quad (8)$$

$$MC_p \frac{dT}{dt} = Q$$

where, $M_a =$ molecular weight of the gas, $W_{in} =$ gas flow influx, $W_{out} =$ gas flow out-flux, $C_p$ specific heat of the gas at constant pressure, and $M =$ total mass of the gas in the volume.

PCTRAN-HTTR uses a semi-empirical method for reactor coolant flow that encompasses both forced and natural circulation conditions. For forced circulation when the reactor coolant pumps are on, full (volumetric) rated flow is assumed. In the course of pump trip, the flow coasts down exponentially until stable natural circulation is established (9).

$$W(t) = W_1 + (W_0 - W_1) \quad (9)$$

where, $W_0$ is the nominal rated flow and $W_1$ is the eventual natural circulation flow, which is usually a few percent of the nominal. $\tau$ is the characteristic time. This equation is also used for pump run-out to exceed 100% of its nominal value, i.e., $W_1 > W_0$. Whiles, the reactor hot and cold leg temperatures for a HTR are calculated by the heat balance from the reactor core and removal rate by the steam generators at a given loop flow rate as follows:

$$\Delta T = \frac{Q_{sg}}{W_{Re} C_p}, T_H = T_{avg} + \frac{\Delta T}{2}, T_c = T_{avg} - \frac{\Delta T}{2} \quad (10)$$

where $T_{avg}$ is the reactor coolant (RC) average temperature.

1.1 Fuel temperature of the core

A simplified model accounting for the temperatures of fuel and cladding has been constructed in PCTRAN-HTTR. This model can simulate: thermal power transmitted into the coolant in contrast to
nuclear power generated by the core during normal operation; fuel and cladding heatup during accident conditions. Core thermal power $Q_{MWT}$ is represented by

$$Q_{MWT} = UF * (T_F - T_{avg})$$ (11)

Where $T_F$ is the known average fuel temperature at 100% rated power. The heat transfer coefficient $UF$ is then calculated such that the core thermal power is equal to the neutron power from the kinetics equation at steady state.

Transient fuel temperature will be calculated by the imbalance between nuclear power and thermal power. The heat transfer coefficient $UF$ varies with the core flow to the 0.8th power according to Dittus-Boetlller correlation during forced circulation and flow coast down. It will settle at a small value for natural circulation. The graphite sleeve and reflector’s temperature is calculated similarly.

2. Methods

2.1 Description of PCTRAN/HTR Functional Simulator

The display Graphic User Interface (GUI) mimic of PCTRAN/HTR200 is shown in Figure 1.

![Figure 1. PCTRAN/HTR200 Graphic User Interface mimic](image)

In the above GUI mimic, the core is modelled by simple point-kinetics of 6 delayed neutron and 11 decay Gamma groups. Feedback of moderator temperature and fuel Doppler are considered. The fuel uranium and graphite heat capacitance and heat transfer characteristic are considered in the solution algorithm. The ideal gas law is used for helium coolant calculation. The reactor outlet temperature is in the order of 800ºC. The helium pressure is about 7 MPa. The core helium flow is driven by a circulator. An inverted U-bend steam generator containing saturated water and steam converts core heat to steam for power conversion. Simulation speed can be controlled either real-time or accelerated to 2, 4, 8 up to 64 times faster.

The reactor power is controlled by the control rods. During start-up the neutron flux advances many decades till criticality. They are categorized as start-up, intermediate to power range as shown in a vertical semi-log bar in the panel. System control is achieved by combination of rod move, helium circulator speed, hot leg/cold leg bypass valve, helium injection/exhaust and SG feed-water/pressure controls. The reactor protection system that shuts down the chain reaction at extreme conditions.

There is a reactor cavity heat removal system to maintain the containment condition. After shutdown the shutdown cooling system removes the core decay heat.

2.2 Case study using PCTRAN/HTR

The normal startup, power maneuvering, shutdown and accidental pipe breaks, inadvertent rod insertion/withdrawal cases successfully will be performed. Normal power maneuver and the most severe loss of helium circulation without scram event will be demonstrated to show HTR’s performance.
3. Results and Discussion

3.1. Sample run of normal operation - power reduction to 90% and back to 100%
Starting from 100% steady state condition and set the power demand to 90%, the neutron flux drops rapidly whereas the core thermal power follows much slower. This is owing to helium gas small heat capacitance and large fuel heat capacitance. Then we raise the power demand back to 100%. The following figures: Figure 2, Figure 3, Figure 4 and Figure 5, show the HTR operates smoothly. The figures show the curve of power core thermal and neutron flux, peak clad and gas temperature, helium gas and SG pressures, curve of core helium flow, SG feed water and steam flows.

![Figure 2](image2.png)  
**Figure 2.** Power reduction to 90% & back to 100%, curve of power core & neutron flux

![Figure 3](image3.png)  
**Figure 3.** Power reduction to 90% & back to 100%, showing curve of peak clad and gas temperatures.

![Figure 4](image4.png)  
**Figure 4.** Power reduction to 90% and back to 100%, showing curve of helium gas and SG pressures.

![Figure 5](image5.png)  
**Figure 5.** Power reduction to 90% and back to 100%, showing curve of core helium flow, SG feed water and steam flows.

3.2. Sample run of accident - loss of helium circulation ATWS
A helium circulator trip without scram (ATWS) is run. Starting from 100% power, the reactor scram signal on low circulation flow rate is bypassed so the rod control system is locked. We would observe the circulator flow coasts down to natural circulation. The temperatures of the coolant and fuel increase. The large mass and heat capacitance of the fuel and graphite moderator absorb a large amount of core heat. Feedback by the negative moderator temperature and Doppler coefficients reduces the fission
power to a lower level.

The fuel average temperature reaches a peak about 1520°C and then slowly decreases afterwards. The steam generator continues to remove the core heat without adverse consequence. Figure 6, Figure 7, and Figure 8, showing the curve of core helium flow and percentage of reactor core power, temperature profile of coolant gas (inlet/outlet), fuel centerline and peak clad, and percentage of reactor core power, at the loss of helium circulation at ATWS condition.

![Figure 6. Loss of helium circulation at ATWS showing core helium flow (kg/s) and percentage of reactor core power.](image1)

![Figure 7. Loss of helium circulation at ATWS showing temperature profile of coolant gas (inlet/outlet), fuel centerline and peak clad.](image2)

### 3.3. Scaling power to low power test unit

By completing the 200 MWt HTR model, BATAN’s staff created a reduced size 10 MWt (~3 MWe) model. Note in the following mimic the full power is 10 MWt and core neutron flux is $3.10^{14}$ n cm$^{-2}$s$^{-1}$, about one-twentieth of the 200 MWt power unit. A number of normal operation and transient runs have also been conducted. They all perform in reasonable manner. The PCTRAN/HTR software tool proves to be useful in BATAN’s pursuit for nuclear power generation [12].

A study case in using a reduced size 10 MWt PCTRAN/HTR model which is similar with RDE, the software is used simulation of a small break loss of coolant accident (SBLOCA), the simulation result is shown in Figure 8 and Figure 9. Temperature of helium for SBLOCA of 8% pipe break is shown in Figure 8, the highest temperature of HTR coolant is 1150 °C and the maximum fuel centerline temperature is 1296 °C whiles the helium pressure after 8% pipe break is 8 MPa (shown in Figure 9). A sample case of safety analysis simulation to demonstrate the inherent safety features of HTR was done for helium pipe break malfunction scenario. The analysis was done for the variation of primary coolant pipe break i.e. from 0,1% - 0,5 % and 1% - 10 % helium gas leakages, while the reactor was operated at the maximum constant power of 10 MWt for 5000 s. The result shows that the highest temperature of HTR fuel is 1296 °C, and this temperature during the SBLOCA is much lower than the fuel temperature limit of HTR-10 type i.e. 1620°C [13].

The simulation results are consistent with other sophisticated computer code calculations used in benchmarking of HTR safety analysis [9,10,11].
4. Conclusion

The national program on experimental power reactor (RDE) project need a human resources development mainly for its operating personnel. In order to train its personnel, BATAN has cooperated with Micro-Simulation Technology of USA to develop a 200 MWt PC-based simulation model PCTRAN/HTR. A simulation on normal operation of startup, power maneuvering, shutdown and accidents including pipe breaks and complete loss of AC power have been conducted. A sample case of safety analysis simulation to demonstrate the inherent safety features of HTR was done for helium pipe break malfunction scenario. The simulation for the variation of primary coolant pipe break (SBLOCA) i.e. from 0.1% - 0.5% and 1% - 10% helium gas leakages, while the reactor was operated at the maximum constant power of 10 MWt for 5000 s. The result shows that the highest temperature of HTR coolant and fuel centerline are 1150 °C and 1296 °C respectively, and the helium pressure after 8% pipe break is 8 MPa. The fuel temperature limit of HTR-10 type is 1620 °C, therefore, owing to HTR’s unique fuel property and helium gas characteristic, the core will never fail under any possible circumstances.

5. References

[1] Badan Tenaga Nuklir Nasional Rencana Pembangunan RDE di Indonesia http://www.batan.go.id/index.php/id/rencana-pembangunan-rde-di-indonesia
[2] Saito S, Tanaka T and Sudo Y 1994 Design of high temperature engineering test reactor (HTTR) (Japan: Japan Atomic Energy Research Inst).
[3] Al-Mugrabi M and Po L C 1997 IAEA activities in advanced reactor simulation.
[4] Adi A, Syarip S and Elisabeth S 2016 The development of Kartini reactor data acquisition system to support Nuclear Training Centre (NTC) (International Conference: and The 3rd International Conference on Nano Electronics Research and Education (3rd ICNERE) and The 8th International Conference on Electrical, Electronics, Communications, Controls and Informatics System (8th EECCIS)).
[5] Syarip, Puradwi I W and Tegas S 2015 Evaluation on the utilization of Kartini research reactor for education and training programs (Denpasar: 2nd Seminar Nuclear Energy Technology).
[6] IAEA-TECDOC-1411. 2004 Use of control room simulators for training of nuclear power plant personnel (Vienna: IAEA).
[7] Malkawi S A and Al-araidah O B 2013 Students’ assessment of interactive distance
experimentation in nuclear reactor physics laboratory education *European Journal of Engineering Education* **38** 5 512-518.

[8] Kadak A C 2016 The status of the US high-temperature gas reactors *Engineering* **2** 119–123

[9] Sui Z, Sun J, Wei C and Ma Y 2014 The engineering simulation system for HTR-PM *Nuclear Engineering Design* **271** 479–86.

[10] Minggang L and Yujie D 2014 The ATWS analysis of one control rod withdraw out of the HTR-10GTcore in addition with bypass valve failure *Nuclear Engineering and Design* **271** 459–464.

[11] IAEA-TECDOC-1694 2013 *Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to the PBMR-400, PBMM, GT-MHR, HTR-10 and the ASTRA Critical Facility* (Vienna: IAEA).

[12] Syarip, Khoirul A and Dwi P 2016 Post reactor scram control rods position adjustment analysis for the Indonesian experimental power reactor concept *Ganendra Journal of Nuclear Science and Technology, (Jurnal Iptek Nuklir Ganendra)* **19** 2 83-93

[13] Fubing C, Yujie D and Zuoyi Z 2015 Temperature response of the HTR-10 during the power ascension test *Hindawi Publishing Corporation Science and Technology of Nuclear Installations* **2015** 1-13.

**Acknowledgments**

The authors wish to thank Dr. Susilo Widodo, Director of PSTA BATAN and his staffs for their sup