6–4 Safety Enhancement of High-Temperature Gas-Cooled Reactors

— Nuclear and Thermal Design of an Oxidation-Resistant Fuel-Loaded Reactor Core —

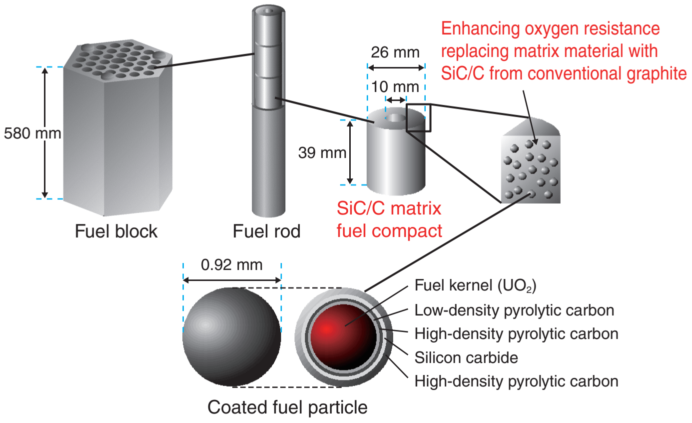
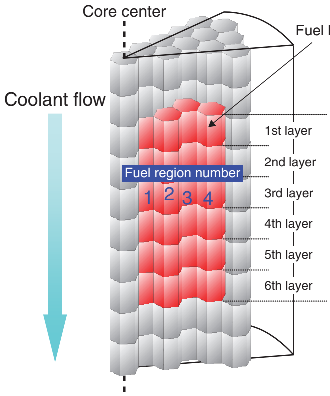


Fig.6-7 HTGR-fuel example with enhanced oxidation resistance
A conventional fuel compact is fabricated by sintering coated-fuel particles, which consist of fuel kernels coated with ceramic layers, with a graphite matrix. The oxidation resistance of the fuel compact is enhanced by replacing the matrix material with a compound of silicon and graphite (SiC/C).



Fuel location (wt%)

	Fuel region number			
	1	2	3	4
1st layer	8.0	8.0	10.0	10.0
2nd layer	8.0	8.0	10.0	10.0
3rd layer	5.4	5.4	8.0	8.0
4th layer	5.4	5.4	8.0	8.0
5th layer	5.4	5.4	5.4	5.4
6th layer	5.4	5.4	5.4	5.4

Fig.6-8 Fuel location to optimize power distribution
Higher-enrichment fuel is placed at the upper region, where the coolant temperature is low, and at the outer region, where neutron flux is low. Owing to this fuel location, the fuel temperature is uniform in the reactor core and consequently the maximum fuel temperature is suppressed.

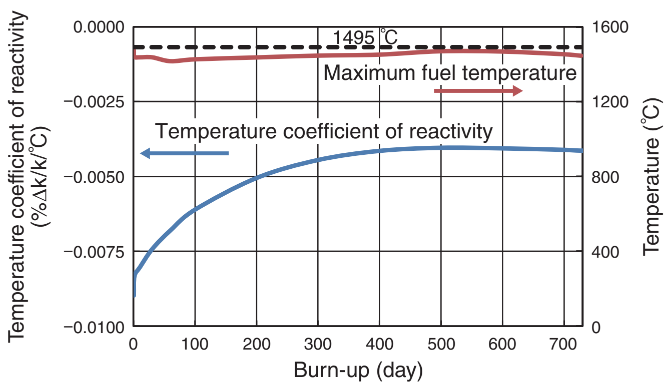


Fig.6-9 Calculation results for the temperature coefficient of reactivity and the maximum fuel temperature
The fuel temperature was confirmed to be lower than the limit of 1495 °C and the temperature coefficient of reactivity had a negative value during the burn-up period.

We are developing an oxidation-resistant fuel (SiC/C-matrix-fuel compact, Fig.6-7), which promises to enhance the safety of High Temperature Gas-cooled Reactors (HTGRs) by maintaining its integrity under air ingress. Such an ingress is considered as a characteristic accident due to pipe rupture, with a large amount of air (beyond any projections) entering into the reactor core. To introduce the SiC/C matrix-fuel compact to the HTGRs, the nuclear and thermal feasibility of the reactor core should be confirmed in addition to establishing fuel-fabrication technology. Nuclear and thermal design was performed for the SiC/C-matrix-fuel-compact-loaded HTGR based on the conceptual design of HTR50S so as to achieve the same performance, namely a 50-MW thermal power and 730-day (2-year) burn-up period.

The most important issues facing the nuclear and thermal design of HTR50S are determination of fuel specifications (such as the degree of enrichment needed for the required period of operation) and optimization of the power distribution to suppress the fuel temperature below the limit during the burn-up period. Silicon, which is included in the SiC/C matrix, more easily yields neutron-capture reactions than graphite and almost fails to moderate neutrons. Owing to these nuclear characteristics, the reactivity of the reactor core decreases when the fuel-compact-matrix material is replaced with SiC/C from conventional graphite. To compensate for the reactivity decrease, the average fuel enrichment was determined to be 1.1wt% higher than the original HTR50S. Optimization of the power distribution was performed using three kinds of fuel enrichment as with the original HTGR50S, and the fuel location was determined as shown in Fig.6-8.

The excess reactivity and power distribution were calculated by performing burn-up calculations with the whole-core model, and the fuel temperature was calculated using the power-distribution results. It was confirmed that the reactor core has enough excess reactivity to operate for 730 days with 50 MW of thermal power, and the fuel temperature is kept below the limit during the burn-up period (Fig.6-9). Additionally, the shutdown margin was confirmed to be larger than 1%Δk/k and it was observed that the reactor could be safely stopped. The temperature coefficient of reactivity was found to have a negative value, leading to a self-stabilizing characteristic. Based on the above results, the nuclear and thermal feasibility of the SiC/C-matrix-fuel-compact-loaded HTGR were confirmed.

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Reference

Aihara, J., Goto, M. et al., Nuclear Thermal Design of High Temperature Gas-Cooled Reactor with SiC/C Mixed Matrix Fuel Compacts, Proceedings of 8th International Topical Meeting on High Temperature Reactor Technology (HTR 2016), Las Vegas, Nevada, USA, 2016, p.814-822, in CD-ROM.