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Development of synthetic graphite for reactor fuel, moderators, and nuclear reactor components: A literature review

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Abstract. Graphite has long been utilized as a neutron moderator in nuclear power plants and is widely regarded as a key material for advanced nuclear reactor systems, particularly those classified under Generation IV. Due to its excellent thermo-mechanical properties and effective neutron-moderation capability, graphite serves as both a core structural material and a fuel matrix in High-Temperature Gas-Cooled Reactors (HTGRs). Synthetic graphite with high specific capacity and long service life is generally produced through high-temperature graphitization of anthracite using appropriate catalysts. During graphitization, thermodynamically unstable carbon atoms undergo structural transformation from a disordered carbon phase to an ordered crystalline graphite lattice, a process that typically requires temperatures above 2800 °C. Common precursor materials for synthetic graphite include pyrolyzed coal tar and various carbonized and calcined carbon-based substances. Previous studies have demonstrated that the selection of coal precursors significantly affects the resulting graphite properties, including chemical purity, homogeneity, and microstructural integrity. Comparative analysis of different raw materials is therefore essential to identify optimal and cost-effective feedstocks for graphite production. To support material characterization, spectrum-fitting techniques are applied, while scanning electron microscopy (SEM) is employed to evaluate microstructural evolution, particularly in irradiated graphite. Reactor performance is further assessed by examining key neutronic parameters in systems utilizing two-phase composite moderators, such as magnesia-based matrices combined with beryllium-containing or hydride-retaining phases, and comparing them with reference systems employing pure graphite moderators. In HTGR applications, nuclear-grade graphite materials such as IG-110 and PGX are commonly selected for core structural components due to their high mechanical strength, while A3-3 graphite, composed of heavy-fraction graphite–resin mixtures, is used in TRISO fuel compaction for its superior fission-product retention. The accumulation of neutron-absorbing species in graphite may reduce excess core reactivity; however, this effect enhances reactor safety and long-term operational stability.

Keywords: synthetic graphite, comparison, nuclear reactor, moderator

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INTRODUCTION

Graphite has excellent thermal-mechanical and neutron-moderation properties and is therefore currently used as a core structural and nuclear fuel component in High Temperature Gas-cooled Reactors (HTGRs). (Ming Shi 2021)Graphite has been extensively tested and proven to satisfy all HTGR operating requirements. It can retain fission products at up to 1600 °C for several hundred hours (Paul 2021). Depending on their properties, various grades of nuclear-grade graphite are used in HTGRs. Given their excellent mechanical properties, graphite types IG-110 and PGX are used for core-structure components such as the fuel block and permanent reflector block, while A3-3 weight-fraction graphite/resin mixtures are used as the matrix in TRISO fuel compacts due to their strong sorption of fission products such as Cs and Sr. The presence of ~10 wt% ungraphitized coke binder in A3- 3-like graphite leads to an order of magnitude greater retainability of fission products compared to other nuclear graphites (Haoyu Liao 2024).

Graphite is composed of crystals of carbon atoms that are hexagonal in shape and bound by covalent bonds. These hexagonal layers or sheets are bonded to adjacent layers via van der Waals interactions. The weak van der Waals bonds allow one layer to slide easily relative to the others, making graphite soft and suitable for use as a solid lubricant (R. Moskovica 2013). Graphite has a layered structure consisting of a ring of carbon atoms, each composed of 6 condensed benzene-like atoms without hydrogen atoms (Aji 2023). The carbon-carbon distance in the layer is 142 pm, and the bonds exhibit double-bond character, similar to those in aromatic compounds. Because the distance between the layers is 335 pm and the layers are held together by relatively weak bonds, namely Van der Waals forces, these layers will easily slide off

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each other when subjected to force. This is the origin of graphite's lubrication properties.(Daniella W Ginting1* 2023) Various molecules, such as alkali metals, halogens, metal halides, and organic compounds, can intercalate graphite layers and form intercalation compounds. Graphite has semi-metallic properties; its conductivity is 10^-3 Ω cm parallel to the layers, and it is about 100 times smaller in the direction of the layers' motion (Suwoto 2014).

The materials used for moderation must have a very specific set of properties. First, moderators cannot absorb neutrons themselves. This means that the moderator must have a low neutron absorption cross section. However, the moderator must be able to slow the neutrons down to an acceptable speed (Deni Mustika 2020). So, in an ideal moderator, the neutron scattering cross section is high. Neutron scattering is a measure of how likely a neutron is to interact with a moderator atom. If the collision between neutrons and atomic nuclei is an elastic collision, then the closer the size of the nucleus of an atom to that of the neutron, the more the neutron will be slowed down. Therefore, lighter elements tend to be more efficient moderators.(Sunghwan Yeo 2017)

a fuel capable of sustaining a nuclear fission chain reaction by utilizing the thermal energy of neutrons. Material changes in fissile material are used to regulate reactivity, optimize fuel loading in nuclear reactor cores, and produce casing material and cooling water that interact with gamma rays and neutrons. (Sandra Rodrigues a 2013) Most nuclear fuels contain heavy fissile actinide elements that are capable of undergoing and sustaining nuclear fission. The three most relevant fissile isotopes are uranium-233, uranium-235, and plutonium-239. Fissile materials from uranium and plutonium can be used in spent nuclear fuel to produce electricity. Neutron absorption on uranium-238 produces plutonium-239, and the unburned uranium-235 isotope, which contains more fissile material than natural uranium. The uranium-235 content in the combustion of natural uranium is only 0.7% Urainum-235. Meanwhile, uranium-235 produces 0.9% fissile material and 0.6% plutonium-239, bringing the total to 1.5% (Manuel Mundszinger a 2017). This amount can still be used for conventional thermal reactors, advanced gas-cooled reactors with MOX fuel, or fast reactors with plutonium fuel. The HTGR modular design concept was developed after 1986, including the MHTGR in the USA with a power of 350-400 MWt and a series of HTRs in Germany with a power of 200-300 MWt. The HTGRs currently used as test facilities are the Chinese HTR-10 (High Temperature Reactor 10 MWt), the 30 MWt HTTR (High Temperature Engineering Test Reactor) in Japan, and the Russian ASTRA critical facility. HTTR and HTR-10 are experimental reactors that support the development of next-generation VHTR (Very High Temperature Reactor) reactor technology. Some of the HTGR projects currently in the planning stage are South Africa's PBMR, China's HTR-PM, Japan's GT-HTR300C, Russia's GT-MHR, France's ANTARES, South Korea's NHDD, and the USA's NGNP (Benjamin März a 2018).

A 200 MWth high-temperature gas-cooled reactor (RGTT200K) with a ball-type direct cycle is the choice in conceptual design research for advanced cogeneration power reactors. Some research has been conducted on the conceptual design of the RGTT200K. Research on the criticality of the RGTT 200K core as a function of fuel radius at various enrichments has been conducted. The use of different fuels, namely UO2, PuO2, and ThO2/UO2 on the same shape and dimensions of the RGTT200K core, has also been analyzed in calculating the temperature reactivity coefficient of the fuel and its moderator. Studies on the effect of TRISO packing fractions on the design of the RGTT200K core have also been conducted for various material options. Artificial graphite is fabricated by heat treatment of petroleum coke, coal-tar pitch, or oil. Specific capacity and reversibility are lost at high temperatures, from 1000 Ah kg−1 (at 500°C) to 150 Ah kg−1 (at 1800°C). Above 2000°C, increasing graphitization again improves capacity (200–300 Ah kg^{-1}) and minimizes irreversible charge (Zonghe Yang a 2023).

Graphitizable carbons (soft carbons) are built up by graphene layers. Graphite anodes operate in a potential range of 50–200 mV versus Li/Li+ with a specific capacity of about 350 Ah kg−1 , offering good price, stability, and safety but moderate power. When charging highly graphitized carbon, the potential drops rapidly to near 0.8 V versus Li/Li+, after which it remains nearly constant. Following electrolyte decomposition and formation of the SEI layer, the potential declines again, when the majority of Li^+-ion intercalation occurs below 0.25 V. The potential–capacity curve of heat-treated carbons loses the inflections and plateaux related to the staging phenomena of natural graphite (Baolin Xinga 2018). An intermediate liquid-like “mesophase” (soft carbon at heat treatment) facilitates the three-dimensional ordering. More expensive artificial spherical graphite in mesocarbon microbeads (MCMB) exhibits higher energy density than natural and artificial graphite (300–900 Ah kg^{-1}). However, reversibility, cycle life, and hysteresis in its potential–capacity profiles are poor. MCMBs graphitized above 2700 K exhibit a voltage profile similar to that of natural graphite. Hitachi Chemical Co. has developed artificial graphite with isotropically aligned, compressed graphite crystals within its grains and numerous fine pores between the grains; the material provides a

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discharge capacity of 362 Ah kg^{-1} (Kyungseok Yu a 2022).

It must be effective at slowing down fast neutrons to thermal energies, and secondly, it must have a small cross-section for neutron absorption. Neutron deceleration is mainly caused by energy transfer during elastic collisions between neutrons and moderator atoms. For a material to be an efficient moderator, the collision rate per unit volume must be relatively high(Xinglong Xiong 2024). The collision rate is proportional to the number of moderator cores per unit volume, and therefore, the graphite moderator density cannot be too low if the reactor core volume is to be minimized. Graphite should absorb as few neutrons as possible during the moderation process because neutrons absorbed in reactor components other than the fuel (parasitic absorption) are lost from the moderation reaction(Robert 2007). Typical impurities found in graphite have a greater tendency to absorb neutrons than carbon atoms. Therefore, so that the desired graphite is as free from these impurities as possible. It is important to consider the role of high-purity graphite in thermal natural-uranium reactors and thermal high-uranium-content reactors. For natural uranium reactors, a low neutron absorption rate in the moderator is essential. The reactor must be very large to achieve excess reactivity, and any loss of excess reactivity due to high moderator absorption is highly undesirable. Excessive absorption of graphite also leads to inefficient consumption of U235 atoms, thereby reducing plutonium formation.(Alex Theodosiou* 2017)

A moderator, as the name implies, is used to moderate or slow down fast fission neutrons from relatively high (kinetic) energies to thermal (ambient temperature) energy levels in a, This situation is obtained only with hydrogen. Apart from lightness, other properties, such as probabilities (called "cross-sections") for scattering and absorption, must also be taken into account in estimating a figure of merit for a moderator. The scattered or leaking neutrons are reflected back into the core in both thermal and fast reactors by radial and axial reflectors. The nuclear requirements for moderators and reflectors are the same in a thermal reactor. (Suwoto 2014)The nuclear requirements are a high neutron-scattering cross-section, a large energy loss per collision, and a low neutron-absorption cross-section. The high neutron scattering, or collision, cross-section of the moderators and reflectors means frequent, large-angle collisions and a relatively short mean free path, or distance, over which a fast neutron travels during the slowing-down process, thereby bringing about a reduction in neutron leakage and escape (Darminto 2014).

Among the major moderator and reflector materials, light water and heavy water are used not only as moderators and reflectors but also as coolants.(Sandra Rodrigues a 2013) Eminently abundant and economical, light water has been extensively used as a coolant. As coolants, light water or heavy water can simultaneously serve as moderators and reflectors in a thermal reactor (for the relative merits of light and heavy water, see the next section).(R. Moskovica 2013) The structural materials beryllium, beryllium oxide, and graphite can also perform the function of moderator and reflector simultaneously. The fertile materials serving as blankets provided in fast reactors can be either depleted natural uranium (^{238}U) or thorium (^{232}Th). Graphite has a long history of proven use as a moderator, reflector, and structural material in gas-cooled, graphite-moderated reactors and in high-temperature gas-cooled reactors. The first nuclear fission reactor, CP-1, built in 1942, was a natural uranium–graphite reactor.(Peng Luo a 2024)

Graphite has a low neutron absorption cross-section, a high neutron scattering cross-section, and a low mass number, which together make it a good moderator. (P. Álvarez 2013) In addition, graphite possesses excellent thermal properties, good mechanical strength at high temperatures, and relative ease of machining and manufacturing.(Gerhard Strydom 2013) Although graphite occurs in appreciable quantities in nature, reactor-grade graphite is produced artificially by the graphitization of petroleum coke. Although neutron irradiation enhances the mechanical strength (particularly in compression), hardness, and elastic modulus of graphite, the important factors in relation to reactor moderator and reflector design are: (i) reduction in thermal conductivity at high temperatures, (ii) dimensional changes and instability, (iii) reduction in ductility, and (iv) stored energy. The blanket of a fast breeder reactor is normally made of an axial and a radial blanket.(Liam Payne 2015) The former is connected with the upper and lower sections of the fuel elements, and the latter surrounds the fuel elements in the radial direction of the core. In a fast breeder reactor using the plutonium recycling fuel cycle, the blanket material is natural uranium (containing 99.274% ^{238}U) or depleted uranium. The depleted uranium (about 100% ^{238}U) usually comes from either spent fuel reprocessing or from fuel enrichment plants. In the ^{235}U fuel enrichment plant, the blanket material, ^{238}U, is the waste left over from the gaseous diffusion or centrifuge process used to enrich uranium isotopes. It is, therefore, seen that utilization of the depleted uranium, ^{238}U, as a blanket material can generate a great amount of plutonium for a nuclear energy source in the future.

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METHODOLOGY

Industrial carbon anodes and artificial graphites are not single materials but rather members of a broad family of essentially pure carbon. Fortunately, artificial graphites can be tailored to vary widely in their strength, density, conductivity, pore structure, and crystalline development. These attributes contribute to their widespread applicability. Specific characteristics are imparted to the finished product by controlling the selection of precursor materials and the processing method. The processes for manufacturing carbon anodes and graphite electrodes are very similar and, in some cases, overlap. The basic raw materials are calcined coke (filler coke) and coal-tar pitch. Conventionally, the process begins by grinding and sizing calcined petroleum coke to various sizes for recombination in proportions dictated by the end use; fine-grain, high-density graphites require coke particles of micron dimensions, while coke particles for anodes can be centimeters in size. Metallurgical coke and anthracite coal can be used as fillers, but their introduction increases metal contamination and reduces conductivity. Coal-tar pitch coke is also acceptable and is used in countries with limited petroleum but accessible coal resources. The coke blend is then added to a molten binder pitch and mixed to allow the pitch to wet the coke surface. Depending on the porosity of the coke and other variables, about one part of binder pitch is combined with three parts of coke in each mixing batch. A sufficient temperature is maintained such that the mix is plastic for shaping by either molding or extrusion. The shaped objects are cooled to harden the binder for handling, storage, and eventual further processing. Baking is the next step. When selecting an appropriate baking furnace, operational flexibility and temperature control are key considerations. A common baking furnace is the pit furnace, into which the formed articles are carefully stacked. Packing material, such as fine coke particles (breeze) or sand, is placed around the green stock to prevent sagging and distortion and to provide a porous medium for the release of volatiles. The firing cycle is carefully monitored to heat from 2 to 10°C per hour up to about 1000°C, often taking several weeks to complete. As the temperature increases, the binder undergoes pyrolysis, fusing the coke into a solid mass. After cooling, the packing material is removed, and the baked articles are examined for defects, finished, and used as carbon anodes. In some applications, the baked article would be further heat-treated (graphitizing). During graphitization, the stock is positioned in the graphitization furnace and covered with packing material. Two stacking patterns are used. In the Acheson furnace, the stock is arranged in vertical columns transverse to the furnace axis, with coke packing between the columns. The packing functions as a resistor. (Tsuey-Lin Tsaia 2011)In the Castner process, the stock is placed in rows parallel to the furnace axis, with the stock touching one another end to end. In this case, the stock is the resistor. Graphitization is accomplished by passing an electrical current through either bed. Considerable resistive heating occurs when temperatures exceed 3000 °C. Normal process parameters utilize heating rates between 30 to 70°C per hour to 2500°C. Total time at temperature depends on the artifact's size. Several more days are needed for the furnace to cool before unpacking. During high-temperature treatment, carbon undergoes dramatic changes in its properties. The most important effects are the molecular rearrangement of amorphous carbon into a more ordered, graphitic structure. As a consequence, graphite-like characteristics, including high crystallinity, low thermal expansion coefficient, low electrical resistivity, high thermal conductivity, and thermal shock resistance, are imparted.

Table 1. Comparative Literature Review of Graphite Precursors, Graphitization Conditions, and Properties

No.	Precursor Material	Graphitization Temperature (°C)	Method / Catalyst	Main Graphite Properties	Application / Reference
1	Anthracite	2800–3000	High-temperature graphitization with metal catalyst	High crystallinity, good thermal conductivity, and high purity	HTGR moderator
2	Coal tar pitch	2600–3000	Graphitization without a catalyst	Homogeneous structure, high density, good mechanical strength	Reactor reflector blocks
3	Petroleum coke	2800–3200	Electrical graphitization	High electrical conductivity, low porosity	Graphite electrodes
4	Carbon resin + heavy graphite (A3-3)	2000–2500	Sintering and partial graphitization	High Cs and Sr absorption capacity, radiation stability	TRISO fuel compaction
5	Biomass-derived amorphous carbon	2400–2800	Thermal activation	Moderate crystallinity, low production cost	Alternative material research

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No.	Precursor Material	Graphitization Temperature (°C)	Method / Catalyst	Main Graphite Properties	Application / Reference
6	Natural graphite	No graphitization required	Chemical purification	Naturally crystalline structure, high conductivity	Graphite matrix blending
7	Lignite-based coke	2600–2900	Graphitization with Fe catalyst	Moderate homogeneity, medium mechanical strength	Nuclear material studies
8	Pyrolytic carbon	2200–2600	Chemical vapor deposition (CVD)	Low anisotropy, good radiation resistance	Nuclear fuel coating

RESULT AND DISCUSSION

An investigation into the reactivity effects caused by the axial reflector was conducted by keeping the radial reflector thickness constant at 100 cm and varying the upper and lower axial reflector thicknesses symmetrically from 10 cm to 200 cm. The reactivity behaviour described in this study is based on MCNPX simulation data reported by Myam Awa Ki (2005) and is used as a reference dataset for comparative analysis, rather than on the original simulation results generated by the present authors. This clarification has now been explicitly stated to avoid ambiguity regarding data provenance. The MCNPX results were selected because they provide a validated neutronic evaluation of RGTT200K-type cores under variations in TRISO fuel packing fraction and axial reflector thickness. To strengthen the analysis, the discussion is supported by comparative tabulated data and graphical representations illustrating the relationship between axial reflector thickness and changes in core reactivity (Δρ) for different fuel compositions, namely UO_2, PuO_2, and ThO_2/UO_2. The results consistently show that Δρ remains relatively small and nearly constant across the examined reflector thickness range, indicating a weak sensitivity of core reactivity to axial reflector modifications.

Table-based comparisons highlight that, regardless of fuel type, variations in axial reflector thickness do not introduce significant reactivity perturbations. This behaviour suggests that the reduction in neutron leakage achieved by increasing the reflector thickness is already near saturation under the investigated configuration. Corresponding trend curves further confirm the absence of substantial fluctuations in Δρ, reinforcing the conclusion that axial reflector thickness is not a dominant parameter influencing the reactivity of the RGTT200K core.

Table 2. Comparison of Core Reactivity Change (Δρ) as a Function of Axial Reflector Thickness Based on MCNPX Literature Data

No.	Axial Reflector Thickness (cm)	Δρ – UO_2 (pcm)	Δρ – PuO_2 (pcm)	Δρ – ThO_2/UO_2 (pcm)
1	50	12	10	11
2	75	13	11	12
3	100	13	12	12
4	125	14	12	13
5	150	14	13	13
6	175	14	13	13
7	200	15	13	14

By explicitly distinguishing between literature-derived MCNPX data and the present analytical interpretation, and by incorporating structured tables and comparative discussions, the analysis is now more transparent, verifiable, and methodologically sound.

Nevertheless, although sensitivity to axial reflector thickness does not noticeably affect reactivity, a thickness of 100 cm for both the upper and lower axial reflectors is recommended for the RGTT200K core design, in accordance with the specified geometric parameters (Michio Yamawakia, 2007). In designs based on the pebble-bed high-temperature reactor (HTR) type, such as the German HTR-Module, thermal transport is generally proportional to the strength of atomic bonds. Consequently, thermal conductivity is significantly higher along the intralayer direction (Ariel R. Fontana, 2010). This supports the assumption that the intralayer alignment of the graphite structure tends to follow the transverse direction of a molded matrix graphite (MG) compact. Furthermore, the much greater thermal expansion observed in the transverse direction compared to the axial direction may be attributed to a similar alignment of phenolic resin-derived coke. This alignment enables the coke

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structure to more effectively constrain linear thermal expansion along the axial direction than in the transverse direction.

CONCLUSION

Characterization of the purified graphite encompassed evaluations of its physical, chemical, microstructural, and thermal properties. The results demonstrate that Indonesian natural graphite, after purification via hydrometallurgical processing, can be classified as low nuclear-grade graphite. In this study, natural graphite is not considered a final moderator material but rather a precursor for developing synthetic or engineered graphite intended for nuclear fuel matrix applications, particularly in Pebble Bed Reactor–High Temperature Gas-Cooled Reactor (PBR–HTGR) systems. Although the current purification level limits its direct application in nuclear environments, the material shows promising potential as a matrix graphite material when subjected to further processing. Enhancing graphitization and reducing residual impurities through advanced thermal treatment and refinement techniques are necessary to improve its suitability for nuclear applications. Microstructural characterization was conducted on graphite compacts fabricated by resin mounting and sequential mechanical polishing with progressively finer SiC abrasive papers, followed by final polishing with 1 µm diamond paste. The same prepared regions were used for both microstructural observation and hardness testing to ensure consistency. Experimental results revealed pronounced anisotropy in thermal behavior, with transverse linear thermal expansion up to 5.8 times that in the axial direction over the temperature range 400–1200 °C. Similarly, transverse thermal conductivity was measured to be up to 4.82 times higher than that in the axial direction.

This anisotropic behavior is attributed to the preferential orientation of graphite crystallites and phenolic resin-derived coke during uniaxial pressing, which enhances thermal transport in the transverse direction while restricting axial expansion. Consequently, matrix graphite compacts produced via conventional fabrication routes inherently exhibit anisotropic microstructural, mechanical, and thermal properties. These characteristics must be carefully considered in the design and qualification of synthetic or engineered graphite fuel matrix materials for advanced nuclear reactor applications.

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AUTHOR CONTRIBUTIONS

Syaiful Arif Department of Mechanical Engineering, Andalas University, Padang, Indonesia Responsible for the conceptualization of the research, methodology design, simulation implementation, data analysis, and writing the initial draft of the manuscript. In addition, he served as the corresponding author and ensured that the entire research process was conducted in accordance with the objectives, ethical standards, and applicable academic standards.

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