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Development Strategy for Thorium Molten Salt Reactor Materials

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Abstract: As one of the six General IV nuclear reactors, the molten salt reactor (MSR) has received increasing attention around the world owing to its high energy conversion efficiency, inherent safety features, and use of thorium fuel. However, the major components of MSRs are facing great challenges owing to the extreme environments in the MSR such as high temperatures, corrosive molten fluoride salt, and neutron irradiation. In this paper, requirements for the MSR materials are analyzed; the development history, current status, and existing problems of the key MSR materials (including alloy structure materials and nuclear graphite) are depicted; key technical problems to be solved during the current construction of experimental MSRs are pointed out; and the development strategy for the MSR materials is proposed based on the research and development plan for MSRs.
Keywords: molten salt reactor; high temperature materials; material-molten salt compatibility; irradiation effect

1 Material requirements for molten salt reactors

The manufacturing materials of the reactor components, such as core vessels, core moderator, reactor loop, heat exchangers, molten salt pumps, and control rods, will directly affect the safety of the staff and safety of the property in the surrounding environment of the reactor. Therefore, these manufacturing materials are the key elements for all types of nuclear reactors. For a long time, the components in nuclear reactors were manufactured using well-established engineering materials in other fields such as steels, superalloys, and carbon materials. The nuclear fuels used for molten salt reactors are very different from the solid nuclear fuels used for current nuclear reactors in operation or under construction and the other next-generation advanced reactors. In molten salt reactors, nuclear fuels are mixed with and transported by a high temperature molten fluoride salt. Therefore, the components in the core and the manufacturing materials will have direct contact with the high temperature molten fluoride salt, nuclear fuel, and fission products. In this case, the manufacturing materials are exposed to an extreme environment with neutron irradiation, high temperature, stress, and corrosive molten salt. Thus, the materials used for molten salt reactors should be resistive to high temperature, resistive to molten salt corrosion, resistive to neutron irradiation, exhibit a certain degree of strength, and exhibit good manufacturability.

As the cooling agent used in molten salt reactors has a boiling point of more than 1400 ℃, molten salt reactors have the potential to operate stably over a long time at atmospheric pressure and a high temperature of 1000 ℃. However, there is still no suitable alloy structural material that can meet the requirements for use in a molten salt environment at such a high temperature over a long time. Currently, the available alloy structural material can only function properly in a molten salt environment at 700 ℃ over a long time.

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(1) Increasing the working temperature will intensify the corrosion from molten salt on an alloy structural material. At this stage, we expect that the average corrosion rate of an alloy structural material in molten salt reactors under a high temperature fluoride salt environment at 700 °C is less than 25 μm per year.

(2) The molten salt reactor operates at atmospheric pressure and does not require the complex high pressure system found in a pressurized water reactor. As a consequence, the mechanical property requirements are less stringent for the alloy structural material used in molten salt reactors than that used in pressurized water reactors. Nevertheless, excellent mechanical properties are still preferred for the alloy structural material used in molten salt reactors to reduce the weight of the component and to improve the overall economic value of the reactor. In particular, the alloy structural materials used in molten salt reactors should possess an excellent high temperature mechanical property such as a strong resistance to high temperature creep.

(3) Since the main components of molten salt reactor are in direct contact with the fuel salt (containing nuclear fuel and fission products), the alloy structural materials used to manufacture the components should be inert to the fission products. In other words, these materials will not react with the fission products (e.g., Te), which would degrade the performance of the alloy structural material.

(4) The alloy structural materials should possess excellent manufacturability and weldability so that it can be processed, joined, and assembled into various components.

(5) Nuclear graphite is used as the moderator in the core of molten salt reactor. The nuclear graphite should exhibit high purity, high isotropy, excellent mechanical properties, and high thermal conductivity.

(6) The irradiation resistance property of nuclear graphite determines the service life of the reactor. Therefore, the nuclear graphite used in molten salt reactors should possess an excellent resistance to neutron irradiation (10 to 30 dpa).

(7) The nuclear graphite in molten salt reactors will have direct contact with the fuel salt. Once the nuclear fuel penetrates to the graphite with the molten salt through the surface pores, it will create local hotspots and damage the graphite. To prevent this from happening, the size of the micropore at the surface of the nuclear graphite in molten salt reactors must be smaller than 1 μm.

2 Material development for molten salt reactors: status quo and future trends

Similar to those of the pressurized water reactor, the components in the molten salt reactor are manufactured from a variety of materials. Conventional components used in a molten salt reactor (i.e., the components that are not in direct contact with molten salt and free from irradiation) are generally manufactured using convectional engineering materials. A detailed discussion of these conventional materials will be skipped in this paper. The following sections will provide a detailed description of the special materials used in molten salt reactors, i.e., the UNS N10003 alloy and nuclear graphite.

2.1 Alloy structural material—UNS N10003 alloy

The key components in a molten salt reactor (including pressure vessel, loop pipe, and heat exchanger pipe) must be able to function stably and effectively over a long time in an extreme environment with high temperatures, corrosive molten salt, and neutron irradiation. However, it is very challenging to find a well-established engineering material that satisfies all the above requirements. Therefore, identifying a suitable structural material is a prominent technical difficulty in the development of molten salt reactor. During the 1950–60s, the Oak Ridge National Lab (ORNL) in the US developed a nickel-based alloy for the Aircraft Nuclear Propulsion (ANP) program and the Molten Salt Reactor Experiment (MSRE). This alloy, known as the UNS N10003 alloy (has already been listed in the high temperature pressure vessel material catalog by ASME standard; the one produced by Haynes International, Inc. in the US is called Hastelloy N alloy, and the one produced in China is called GH3535), is the only structural alloy in the world that has served in a molten salt reactor. After extensive screening and testing work, the researchers at ORNL found that a series of well-established alloys such as Inconel 600 and Hastelloy B cannot meet the high corrosion resistance requirements in the molten salt environment. In 1956, ORNL finalized the main elements in Hastelloy N alloy as Ni-17Mo-7Cr. In 1961, ORNL conducted a civilian-level molten salt reactor experiment (MSRE). During this period, Hastelloy N alloy was primarily used for assessing the corrosion level of molten salt and for measuring mechanical properties. The MSRE reached a critical condition in 1965 and was terminated in 1969. During this time, the Hastelloy N alloy played an important role in ensuring proper operation of the MSRE. Ever since the research plan of the MSRE was terminated, no further investigation or improvement has been made to the Hastelloy N alloy [1]. In the early stages, the Hastelloy N alloy was only produced commercially in the world by Haynes International, Inc. This material is categorized as a nuclear control material by the US government

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and must be approved before being exported to China.

ORNL and Haynes conducted a series of studies on the manufacturing process of the Hastelloy N alloy from the 1950s to1970s and finalized a complete set of manufacturing procedures including forging, hot rolling, and hot extrusion. Specifically, the processing parameters for each step were confirmed through research. These include the operating temperature range and the forging reduction, which were confirmed during the forging step; the operating temperature range, the rolling reduction, and rolling speed, which were confirmed during the hot rolling step; and the operating temperature range, extrusion ratio, and extrusion rate, which were confirmed during the hot extrusion step.

A distinct advantage of the Hastelloy N alloy is its excellent resistance to high temperature corrosion in a molten salt environment (e.g., the corrosion rate in pure FLiNaK salt is less than 25 μm per year). At the same time, the Hastelloy N alloy exhibits excellent high-temperature mechanical properties as well as a strong resistance to neutron irradiation. Owing to these merits, using the Hastelloy N alloy as the structural material in the experimental reactor can satisfy the design service life requirement of 30 years [1].

While the US government approved the export of the Hastelloy N alloy to China for constructing molten salt reactors several years ago, this policy is strongly dependent on the unstable relationship between the US and China. To eliminate the reliance on other countries for nuclear power development in China in the long run, our research team has started research on the domestic production of molten salt corrosion resistive alloy (GH3535) following the initiation of the “thorium-based molten salt reactor nuclear energy system—TMSR” special project, a science and technology pilot project announced by the Chinese Academy of Sciences. Since 2011, with funding provided by the Chinese Academy of Sciences to support the Generation IV fission reactor nuclear power system (TMSR) special project, we optimized the element content (C and Si) in the Hastelloy N alloy through collaboration with domestic research institutes (Institute of Metal Research of the Chinese Academy of Sciences) and manufacturers (Fushun Special Steel Co., Ltd., Baoshan Iron and Steel Co., Ltd., Jiuli Group Co., Ltd., and Chengdu Aerospace Superalloy Technology Co., Ltd.) in China [2–5]. Furthermore, we finalized the technical conditions for the key processing procedures (homogenization, processing window, welding, and heat treatment). Currently, we already possess the capability to manufacture large-scale GH3535 alloy ingots (10 t) after conducting pilot, small-scale, medium-scale, and large-scale production in a laboratory. Furthermore, we successfully developed a rolled alloy ring of a large size (ϕ3000 × 400), alloy pipes of different sizes (ϕ14–ϕ168), alloy plates, head, and bends. The sizes of these parts can basically satisfy the requirement for different components in the experimental reactor. The technical conditions for alloy welding were also determined through experiments. These conditions laid the foundation for the manufacture of experimental reactor components and the construction of the reactors. The development and mass-production of the GH3535 alloy not only ensure the smooth progress of the TMSR special project, but also create the necessary conditions that allow molten salt reactors to be used as demonstrative reactors and commercial reactors. Recently, the Hastelloy N alloy was listed by the US government among the items prohibited from being exported to China. This scenario demonstrates that the development of alloy structural materials for molten salt reactors locally during the early stages of the TMSR special project is a predictive and forward-looking move.

The different mechanical properties and molten salt corrosion resistance of the GH3535 alloy were tested in an experimental system. The test results indicate that the comprehensive performance of the GH3535 alloy produced in China is comparable to that of the Hastelloy N alloy imported from other countries (Table 1 and Table 2). This finding suggests that the alloy produced in China can be used as an alternative to the imported alloy used to build molten salt reactors.

Currently, apart from the research and development organizations affiliated with the Innovation Center of Advanced Nuclear Energy Technologies of the Chinese Academy of Sciences (Shanghai Institute of Applied Physics, and Institute of Metal Research of the Chinese Academy of Sciences), other well-known manufactures in China (Fushun Special Steel Co., Ltd., Baoshan Iron and Steel Co., Ltd., Jiuli Group Co., Ltd., and Chengdu Aerospace Superalloy Technology Co., Ltd.) have also participated in the research and development of this alloy. Many foreign countries (e.g., the US, France, Russia, and Czech Republic) are also actively conducting similar research, as shown in Fig. 1. However, except in the US and China, the alloy structural material remains at the stage of laboratory research in other countries, without having yet reached the capacity for mass production.

Table 1. Impact energy of the Hastelloy N alloy and GH3535 alloy at 650 °C and 700 °C.

Testing temperature (°C)	Impact energy of different testing materials (J)		
	Hastelloy N alloy	GH3535 alloy (before optimization)	GH3535 alloy (after optimization)
650	121.7	132.0	236.0
700	130.7	129.3	236.3

Table 2. Mass loss per unit area, corrosion depth, and Cr diffusion depth for the GH3535 and Hastelloy N alloys.

Experimental materials	Sample number	Temperature (°C)	Crucible material	Mass loss per unit area ($mg \cdot cm^{-2}$)	Corrosion depth (μm)	Cr diffusion depth (μm)
GH3535	11	650	Graphite crucible	4.71±0.22	51	58
Hastelloy N	21		Pure nickel crucible	4.25	55	88
GH3535	13			3.95	45	70
Hastelloy N	22			4.17	56	87
GH3535	19	700		4.59±0.10	55	92
Hastelloy N	25			4.39	51	80
GH3535	111			1.98±0.12	22	40
Hastelloy N	26			2.14	24	48
GH3535	113			7.03±0.24	53	88
Hastelloy N	27			7.02	58	85

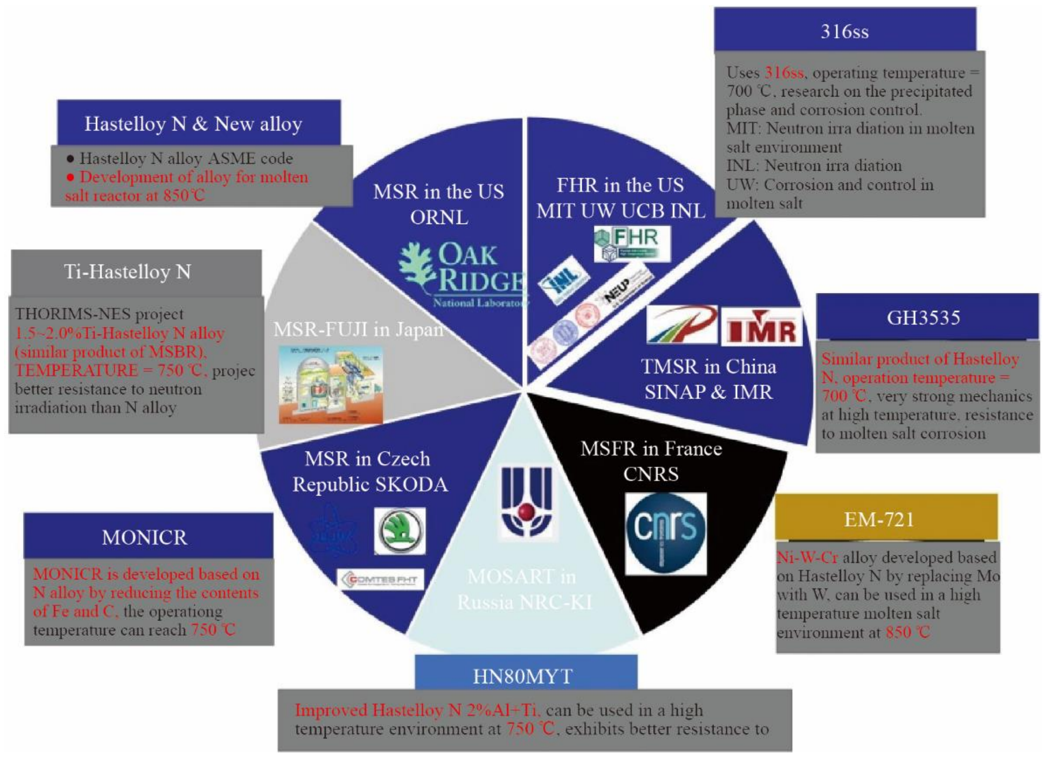

Fig. 1. Summary of research studies on the alloy structural material of molten salt reactors in China and abroad.

2.2 Nuclear graphite

Nuclear graphite is an excellent neutron moderator and reflector material. It has an excellent slowing ratio, high chemical stability, strong radiation resistance, and low cost. The application of nuclear graphite in gas-cooled reactors is already quite mature. Currently, studies on nuclear graphite conducted around the world are mostly based on its application in high-temperature gas-cooled reactors. For molten salt reactors, the nuclear graphite is used as the moderator

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and reflector. At the same time, it also functions as the flow channel for molten salt. Therefore, the nuclear graphite has to be in direct contact with the molten salt.

Nuclear graphite is a porous material. If the micropore size at the surface of the graphite is large enough, molten salt in contact with the graphite may seep into the interior of the graphite through the pore and cause serious consequences. On one hand, the penetration of molten salt into the graphite may alter the performance of the graphite, particularly the irradiation damage behavior. On the other hand, the fuel after infiltrating the graphite conveyed by the molten salt may create local hotspots inside the graphite and reduce its service life. Therefore, for it to be used in molten salt reactors, nuclear graphite must be able to prevent molten salt infiltration in addition to satisfying the conventional requirements (high purity, high strength, high thermal conductivity, high isotropy, and resistance to neutron irradiation). At present, no country has developed a mature nuclear graphite capable of blocking the infiltration of molten salt.

ORNL has developed a CGB nuclear graphite specifically made for molten salt reactors under the given conditions. This nuclear graphite has a large density and an extremely small micropore size. The experimental results of the MSRE demonstrated that CGB nuclear graphite can sufficiently satisfy the requirements for application in molten salt reactors [1]. However, the production of this nuclear graphite was discontinued after the termination of the molten salt reactor program in the US. As a consequence, the research and development of nuclear graphite for molten salt reactors came to an end. In addition, owing to the limited synthesis techniques of graphite at the time, the CGB nuclear graphite produced by ORNL is anisotropic and fails to satisfy the current requirements for nuclear graphite (highly isotropic). The development of gas-cooled reactors has greatly promoted the development of nuclear graphite. The continuous improvement of the synthesis technique has also led to substantial performance enhancement of nuclear graphite. Currently, the nuclear graphite used in gas-cooled reactors has already been developed to the third generation (Figs. 2 and 3).

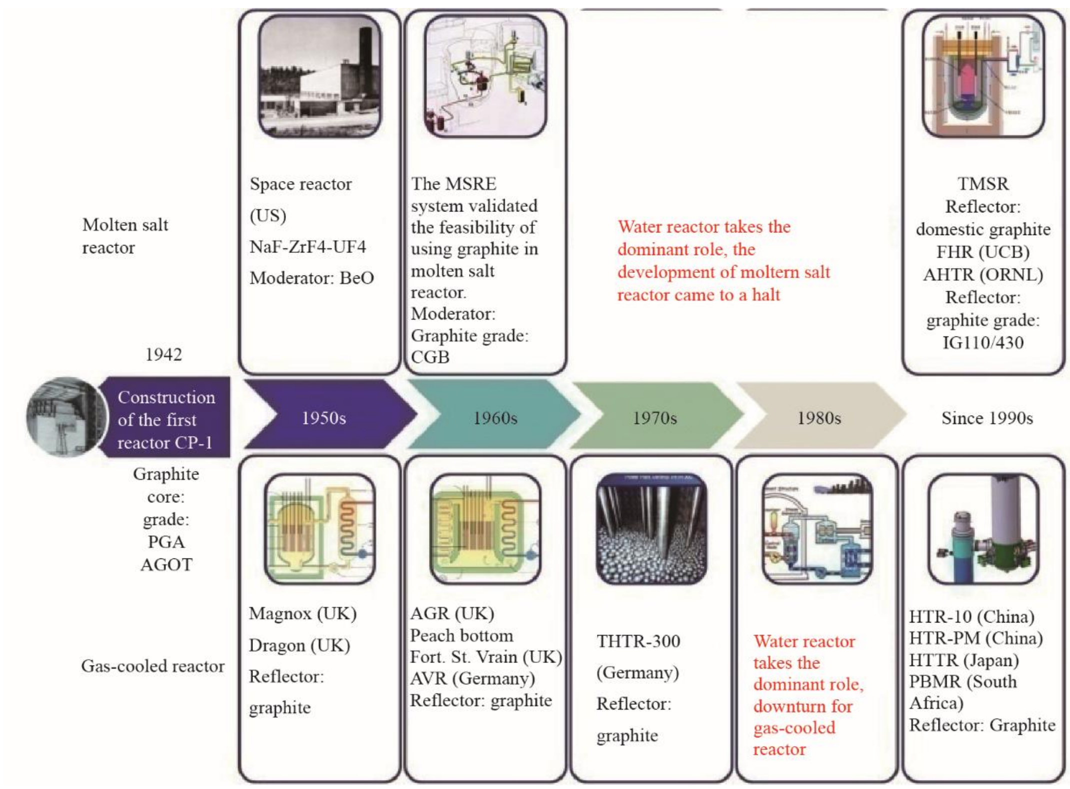

Fig. 2. Development history of nuclear graphite in molten salt reactors and gas-cooled reactors.

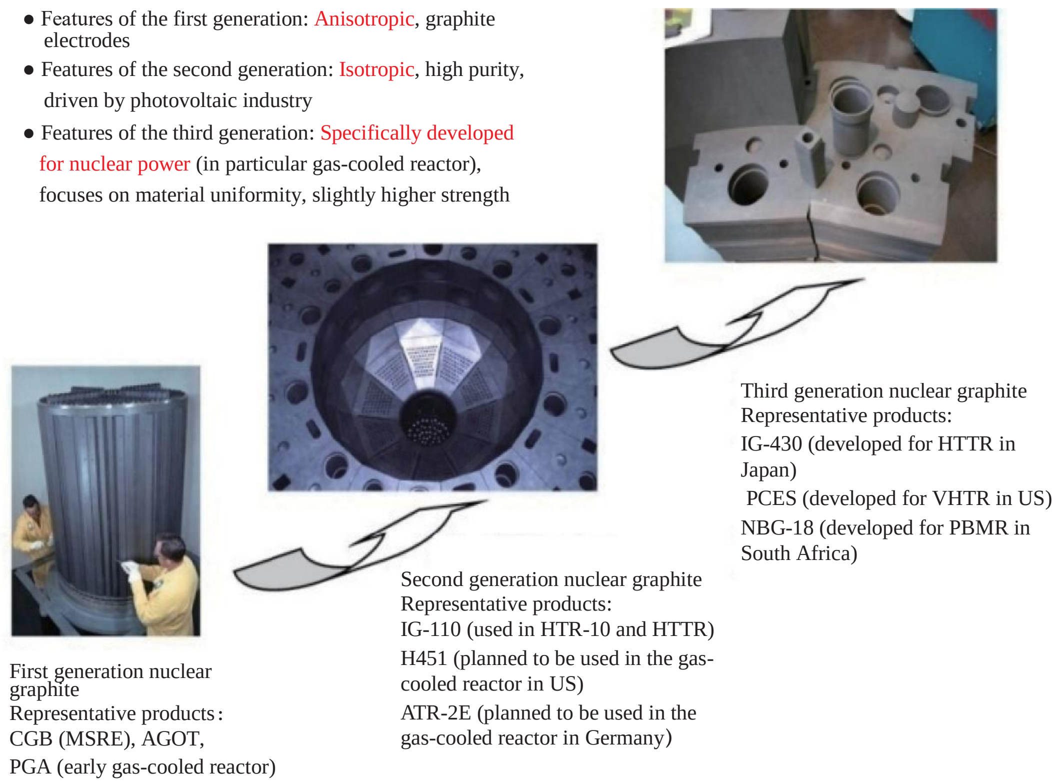

Fig. 3. Development history of nuclear graphite.

As the molten salt reactor and gas-cooled reactor place different demands on nuclear graphite, the nuclear graphite developed for gas-cooled reactors cannot be used directly in molten salt reactors. To obtain the appropriate nuclear graphite specifically for use in molten salt reactors, our team has developed two types of ultrafine graphite through collaboration with the carbon material research organization in China (Shanxi Institute of Coal Chemistry of the Chinese Academy of Sciences) and well-known carbon companies. The first type of graphite is NG-CT-50, which was developed collaboratively with Chengdu Carbon Co., Ltd. of the Fangda Group (Fig. 4). The second type of graphite is T220, which was developed collaboratively with Sinosteel Group New Materials (Zhejiang) Co., Ltd. Both graphite samples have a pore size of less than 1 μm and a dense structure. These features can effectively prevent molten salt from infiltrating the graphite. The dimensions of these nuclear graphite products also satisfy the manufacturing requirements of the components used in the TMSR experimental reactor. In addition, the regular properties of these two grades of domestic nuclear graphite and their compatibility with molten salt were tested in detail (Table 3). Apart from a low fracture toughness and an unassessed resistance to neutron irradiation, all other properties of these nuclear graphite products have met the design requirements for constructing the experimental reactor. The authors of this paper and their team have been working on the development of materials for molten salt reactors over the last several years. During this process, the team has collaborated extensively and established strong working relations with peers both in China and abroad (Figs. 5 and 6). The R&D efforts in the field of molten salt reactor materials and cooperation with domestic and foreign peers have enabled the nuclear material R&D team at the Innovation Center of Advanced Nuclear Energy Technology of the Chinese Academy of Sciences to grow rapidly.

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Fig. 4. NG-CT-50 ultrafine graphite blanks.

Table 3. Performance parameters of nuclear graphite NG-CT-50 and T220 graphite for molten salt reactors.

Property	Specifications	NG-CT-50	T220
Density/(g·cm^{-3})	⩾1.75	1.77	1.87
Most probable pore size/μm	⩽1.0	0.86	0.95
Strength of extension/MPa	⩾22	25.9	33.2
Compressive strength/MPa	⩾65	71.1	124.1
Bending strength/MPa	⩾30	36.3	35.5
Breaking tenacity K_{IC}/MPa·m$^{1/2}$	~1.0	0.71	0.85
Thermal coefficient of expansion (room temperature of −600°C)/1/°C × 10^{-6}	⩽5.5	3.6	4.6
Anisotropy coefficient α_{AG}/α_{WG}	⩽1.1	1.08	1.08
Heat conductivity at room temperature /w/(m·°C)	⩾80	90.6	83.4
Critical infiltration pressure/atm	⩾6.0	6.5	6.5

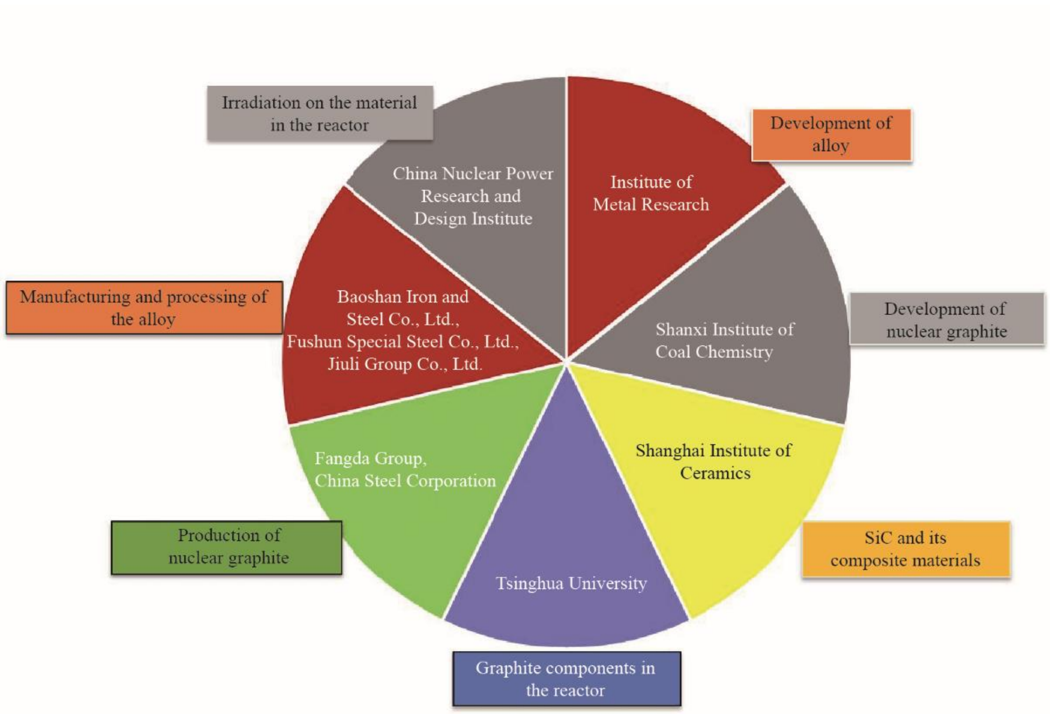

Fig. 5. A summary of the domestic research collaborations for material development for molten salt reactors in China.

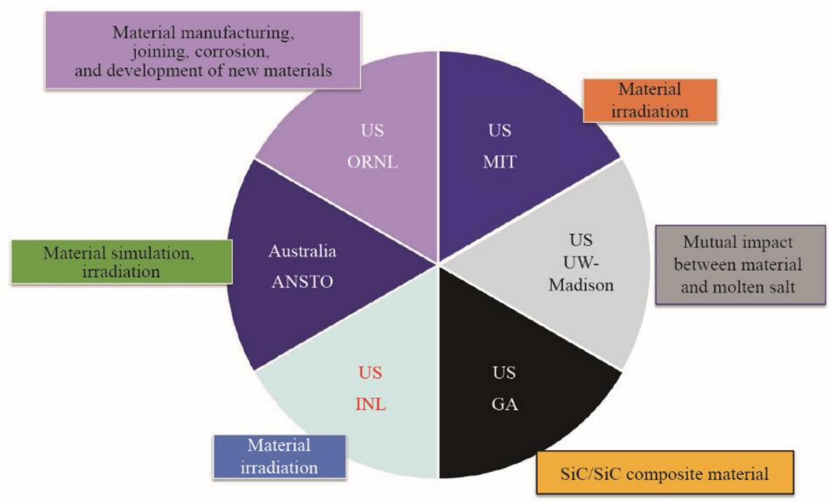

Fig. 6. Summary of international research collaborations for material research for molten salt reactors.

3 Major issues in current material development and future trends

3.1 Structural material

The molten salt reactor is a high temperature reactor. In principle, only the licensed building material for molten salt reactors can be used as the structural material. According to regulations described by ASME-NH, only four types of alloy material (including stainless steel) can be used for constructing a high temperature nuclear reactor, and this list does not include the UNS N10003 alloy. The UNS N10003 alloy is the only licensed material for building pressure vessels according to ASME standards. Therefore, the primary issue at this stage is the lack of sufficient data relevant to the UNS N10003 alloy, which prevents it from being listed by ASME-NH as a permitted building material for high temperature nuclear reactors.

At present, the missing data for the UNS N10003 alloy include the high temperature creep, creep fatigue, and anti-neutron irradiation performance over a long period of time. However, gathering these data requires a large amount of time and financial support.

The fuels used in molten salt reactors, unlike those used in other types of reactors, have direct contact with the components inside the reactor. Therefore, special attention should be paid to the interaction between the fission product and the building material of the molten salt reactor. Another issue of the UNS N10003 alloy is that the fission product (such as Te) may cause the alloy to become brittle. In other words, the fission product such as Te will diffuse into the alloy through the grain boundaries. This behavior will cause the grain boundaries of the alloy to become brittle and fracture (Fig. 7) [1,6,7]. While existing studies have shown that the embrittlement of the alloy by Te diffusion can be inhibited by adjusting the pH value of the fuel salt (Fig. 8) [6], further experiments are still required to validate this point.

Furthermore, the UNS N10003 alloy contains a large amount of Ni, which makes the material relatively expensive. Furthermore, the maximum operating temperature of the UNS N10003 alloy is only 700 °C, which restricts the potential of the molten salt reactor to operate at higher temperatures.

The future development trends will focus on resolving the aforementioned issues. On the one hand, more data will be gathered for the USN N10003 alloy so that it can be listed as a permitted alloy material in high temperature reactors by the ASME standard. On the other hand, new high temperature materials resistant to molten salt corrosion will be developed using modern material science and technology. These materials include novel high temperature/molten salt corrosion resistive alloys, bimetallic composite materials (e.g., stainless steel–nickel alloy composite), and C/C composite materials. The overall target is to improve the high temperature strength, resistance to molten salt corrosion, and resistance to neutron irradiation of the alloy material. These performance enhancements will enable the molten salt reactor to operate

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at a higher temperature with a longer component service life and better economic value.

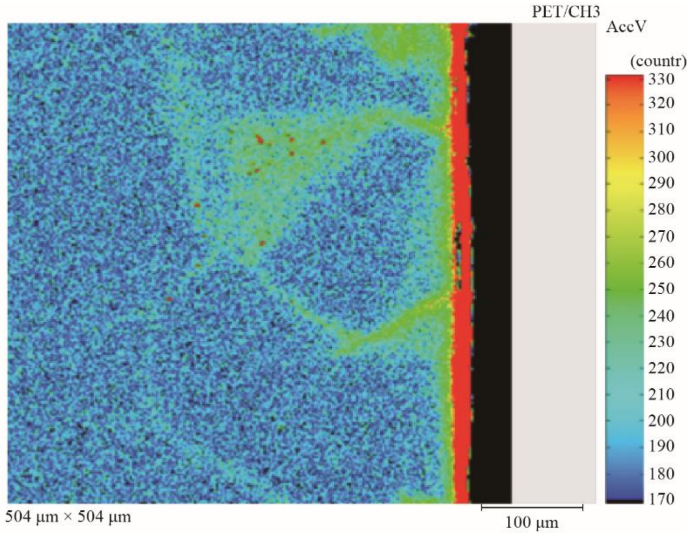

Fig. 7. Diffusion of Te in the Ni alloy along the crystal grain.

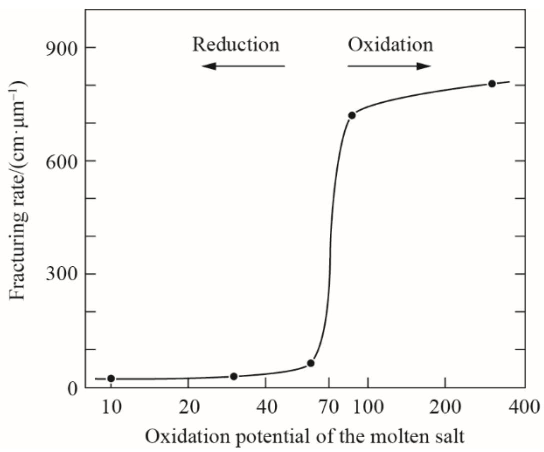

Fig. 8. Relationship between the fracture rate of the alloy induced by Te and the oxidation potential of the molten salt.

3.2 Nuclear graphite

The nuclear graphite used in molten salt reactors must be able to block molten salt infiltration. This feature requires the micropore size of the graphite material to be less than 1 μm. Such a requirement can only be satisfied for graphite materials prepared using ultrafine particle aggregates. However, graphite with an ultrafine grain is often very brittle and prone to cracking. They generally have a low fracture toughness and low yield. In addition, it is very challenging to manufacture such brittle graphite products in large sizes. Therefore, reducing the brittleness of ultrafine-grain graphite (e.g., improving the plasticity of graphite by optimizing the preparation process or optimizing the design of the reactor components to allow brittle graphite to satisfy the manufacturing requirements) is a major issue that remains to be resolved for the nuclear graphite used in molten salt reactors.

The service life of molten salt reactors is dependent on the neutron irradiation lifespan of the nuclear graphite. Graphite with ultrafine and coarse grains exhibits different degrees of compactness; thus, the causative behavior of graphite with ultra-fine particles under a neutron irradiation environment is very different from that of graphite with coarse particles. In particular, the neutron irradiation lifespan may be reduced, which is another major issue affecting the service life of nuclear graphite in molten salt reactors.

The future development trends of nuclear graphite will focus on improving the preparation process of graphite on the basis of current studies on nuclear graphite. Specifically, future research will seek to resolve the issue of brittleness of ultrafine-grained graphite and obtain graphite products with a long lifespan through irradiation screening. These approaches will ultimately improve the economic value of the molten salt reactor.

4 Material development gap between China and foreign countries

During the 1950s and 1960s, ORNL developed an alloy resistant to molten salt corrosion. This alloy, named Hastelloy N, is now produced by Haynes International Inc. in the US. In the past few years, the collaborative efforts with different research organizations and manufacturers have enabled China to produce its own Hastelloy N alloy. This alloy is locally known as GH3535. Currently, the performance of the domestically produced GH3535 alloy is comparable to that of Hastelloy N. In addition, China is in an internationally leading position in terms of the dimensions and parameters as well as performance data collection of the alloy products.

Nuclear graphite produced in foreign countries is developed for gas-cooled reactors. Thus far, only China has developed two types of ultrafine-grained graphite specifically for molten salt reactors. Currently, China is taking the lead in the development of nuclear graphite for molten salt reactors.

5 Goals and general principles of material development for molten salt reactors

The overall goal of material development for molten salt reactors is oriented towards the development needs of thorium-based molten salt high temperature reactors (such as experimental reactors, small-scale modular reactors, demonstrative reactors, special reactors, and commercial reactors). Based on the material development progress and capability in China, our objective is to resolve the critical technical issues encountered in the current material development for molten salt reactors. In addition, reserve material technologies for molten salt reactors should also be taken into consideration to develop novel, next-generation, and advanced molten salt reactor materials with better high temperature tolerance, stronger irradiation resistance, and better economic value. During this process, a high-quality nuclear material research team will be established along with a world-class nuclear material evaluation and test platform and R&D base. The specific targets include further improvement of the alloy profiles, improvement of the manufacturing process for nuclear graphite components, and collection of performance data of the materials. These activities will support the construction of a series of reactors at different stages including a simulated molten salt reactor (to be finished by 2019), a 2 MW experimental reactor (to be finished by 2020), a 20-MW research reactor (to be finished by 2025), and a 100 MW demonstrative reactor (to be finished by 2030). Through collaborations with research institutes and manufacturers in China, an “industry–university–research” model of an integrated molten salt material development and manufacturing base is expected to be formed.

For alloy structural materials, the first objective is to improve the UNS N10003 alloy. The second task is to develop substituting materials such as stainless steel–nickel alloy composite materials (stainless steel substrate with its surface coated or welded with nickel alloy). This composite material has both the mechanical properties of stainless steel and the molten salt corrosion resistance of nickel alloy. Compared to the UNS N10003 alloy, the bimetallic composite material has a more mature manufacturing technology, better economic value, and more comprehensive material data. Furthermore, stainless steel used as a base material is listed by ASME-NH as one of the four permitted building materials for high temperature nuclear reactors. Regardless of whether the UNS N10003 alloy or stainless steel is used, the maximum operating temperature of molten salt reactors is restricted at around 700 ℃. To further increase the working temperature and the energy conversion efficiency of the molten salt reactor, other types of high temperature materials must also be considered. The next stage of material development will focus on novel materials for high temperature molten salt reactors. These materials include novel high temperature alloys with an operating temperature greater than 850 ℃, C/C composite materials with an operating temperature greater than 1000 ℃, and SiC/SiC composite materials.

The development of nuclear graphite will focus on addressing the key scientific and technological problems, that is, the brittleness of ultrafine-grained graphite and its uncertain behavior under neutron irradiation. The goal is to improve the material design and the preparation process to reduce the brittleness and increase the neutron irradiation lifespan of ultrafine-grained graphite. These performance enhancements will extend the service life and improve the economic value of molten salt reactors.

6 Roadmap of material development for molten salt reactors in China

According to the strategic development plan of thorium-based molten salt reactors in China, the research and development of materials must be conducted in accordance with the construction requirements of molten salt reactors at different stages (e.g., stages of experimental reactor, small-scale modular reactor, demonstrative reactor, and commercial reactor) (Fig. 9). It is expected that the construction of all GH3535 alloy components, graphite moderators, and graphite reflectors in the experimental reactor will be completed by 2020. In addition, these materials should have already

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undergone neutron irradiation evaluation experiments, relevant performance tests, and passed the key material safety analysis for experimental reactors. By 2025, the manufacturing process of stainless steel–nickel composite materials should be finalized, and its resistance to molten salt corrosion should also be tested. At the same time, the feasibility of using stainless steel–nickel composite materials in the 20 MW research reactor and future demonstrative reactor should be confirmed. By 2030, novel alloy structural materials and C/C composite material components should be developed with an operating temperature of 850 ℃. Also, the development of large-scale components for the demonstrative reactor should be completed. Finally, these novel materials should have undergone the relevant performance tests and passed the material safety analysis inspection for demonstrative reactors.

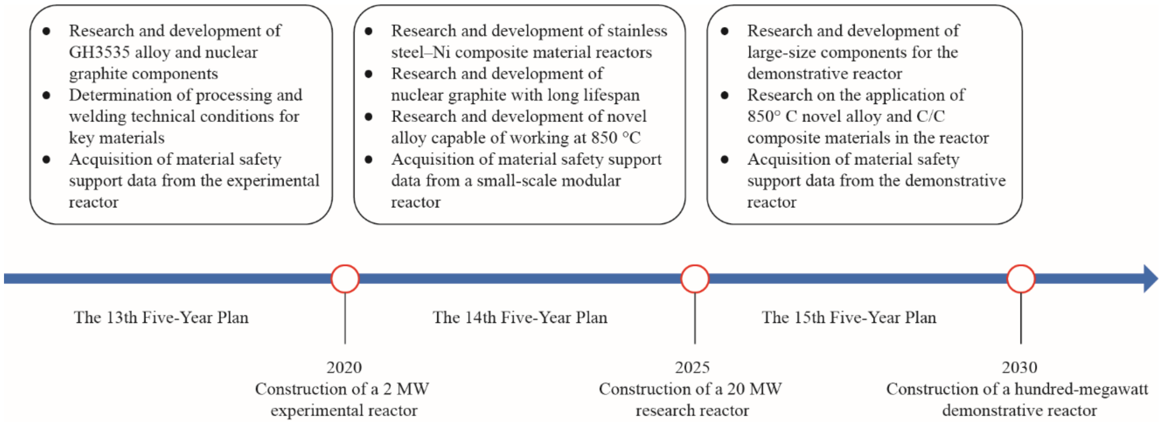

Fig. 9. Material development roadmap for molten salt reactors.

7 Key points and main tasks of material development for molten salt reactors during the 13th Five-Year Plan period

According to the material development plan for molten salt reactors, future tasks should emphasize the following three aspects: material development, performance evaluation, and safety analysis.

In terms of material development, research and development efforts will focus on the key materials used in the experimental reactor such as the GH3535 alloy and ultrafine-grained nuclear graphite. Based on early research findings, the future research activities will seek to determine and finalize the preparation, processing, welding, and heat treatment techniques of alloy structural materials and their various profiles. In addition, the preparation and processing techniques of ultrafine-grained nuclear graphite and its components should also be determined and finalized. Furthermore, research on the materials used for the small-scale modular reactor and demonstrative reactor should be initiated early according to the development plan for molten salt reactors. In other words, a detailed analysis should be performed on the operating conditions of the molten salt reactors (high temperature, molten salt corrosion, and neutron irradiation) and the corresponding material behavior (including degradation and failure patterns and mechanisms of the material) under such conditions. The overall target is to develop economic materials for molten salt reactors with a strong resistance to high temperature, molten salt corrosion, and irradiation. Optimization analysis should also be performed on the material composition (raw material), structural design, and component parameters to improve the safety and economic value of the material. By establishing a relationship between material composition, structure, and performance, research should explore the material preparation processes and optimize the processing parameters. The specific material development plan is given as follows. The feasibility of using (1) stainless steel–Ni alloy bimetallic composite material, (2) novel high temperature/molten salt corrosion resistive alloy structural materials, and (3) carbon-based composite materials in molten salt reactors will be explored preliminarily through collaboration with research institutes and manufacturers in China. The research on ultrafine-grained nuclear graphite with a long lifespan will also be initiated. These research activities will lay a solid foundation for future applications in the small-scale modular reactor and demonstrative reactor.

In terms of material performance evaluation, a series of mechanical property and thermal property tests should be performed on the material according to the operating conditions and service life of the experimental reactor. To ensure the test data are reliable, it is necessary to first determine the test standards of each test item, establish test specifications and instruction guides, and calibrate the testing equipment based on the regulations periodically. The testers must abide by the test standards, specifications, and operating procedures strictly when performing the tests. Important experimental results must be compared with the test results obtained from another independent laboratory. Finally, irradiation and post-irradiation tests (PIE) of the material should be conducted in the reactor to determine the irradiation failure criterion of the material.

Two important aspects should be considered in terms of material safety analysis. First, it is necessary to ensure that the preparation, processing, welding, and heat treatment processes of all materials and profiles are conducted in accordance with finalized procedures strictly. At the same time, all material performance tests must be performed in strict accordance with test standards, specifications, and work instructions. A material database for molten salt reactors needs to be established. Secondly, it is necessary to assess the impact of component size, shape, and working environment on the material behavior. The thermal properties of the material (such as the resistance to neutron irradiation for nuclear graphite) also play a decisive role in determining the lifespan of the core components. All of the aforementioned factors are important for the safety analysis of the materials and components built from these materials. We will complete the safety analysis of the experimental reactor materials and approve relevant assessments after taking the above factors into full consideration.

During the "13th Five-Year Plan," the focus of molten salt reactor material development is in the research and development of materials and components for the experimental reactor. The main tasks include (1) finalizing the preparation, processing, and welding techniques for the material and components to be used in the simulated reactor and experimental reactor; (2) conducting performance tests of the UNS N10003 alloy, establishing a database, and analyzing and reviewing the safety analysis inspection for the experimental reactor; (3) conducting neutron irradiation tests on ultrafine-grained graphite and approving the safety analysis inspection for the experimental reactor; and (4) initiating research work on stainless steel–Ni alloy bimetallic composite material and next-generation high performance molten salt reactor materials (e.g., novel high temperature molten salt corrosion resistive alloy/composite materials, radiation resistive nanostructured materials, nuclear graphite materials with long lifespan) early to lay a solid foundation for the small-scale modular reactor and demonstrative reactor in the late stage.

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