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Venneri et al.

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(54) **COMPOSITE MODERATOR FOR NUCLEAR REACTOR SYSTEMS**

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G21C 5/12 (2006.01)
G21C 5/18 (2006.01)
G21C 3/60 (2006.01)

(52) **U.S. Cl.**
CPC ***G21C 5/12*** (2013.01); ***G21C 5/18*** (2013.01); *G21C 3/60* (2013.01)

(58) **Field of Classification Search**
CPC G21C 5/12; G21C 5/18
(Continued)

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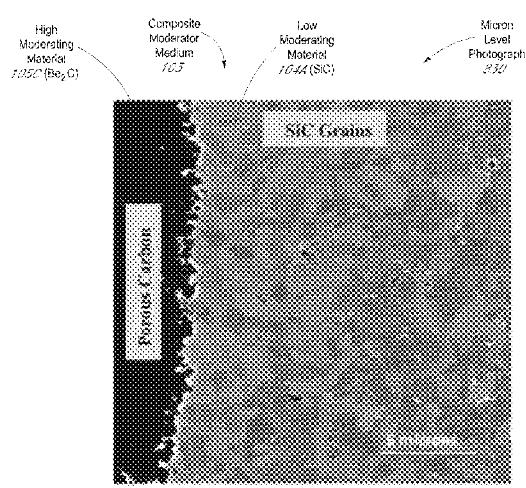
(57) **ABSTRACT**

A composite moderator medium for nuclear reactor systems and a method of fabricating a composite moderator block formed of the composite moderator medium. The composite moderator medium includes two or more moderators, such as a low moderating material and a high moderating material. The high moderating material has a higher neutron slowing down power compared to the low moderating material. The low moderating material includes a moderating matrix of silicon carbide or magnesium oxide. The high moderating material is dispersed within the moderating matrix and includes beryllium, boron, or a compound thereof. The high moderating material is encapsulated within the low moderating material such that the high moderating material is not exposed outside of the low moderating material. The method can include selecting a sintering aid and a weight percent of the sintering aid in a composite moderator mixture based on the low moderating material and spark plasma sintering.

9 Claims, 12 Drawing Sheets



(58) **Field of Classification Search**
USPC 376/350, 458, 904
See application file for complete search history.

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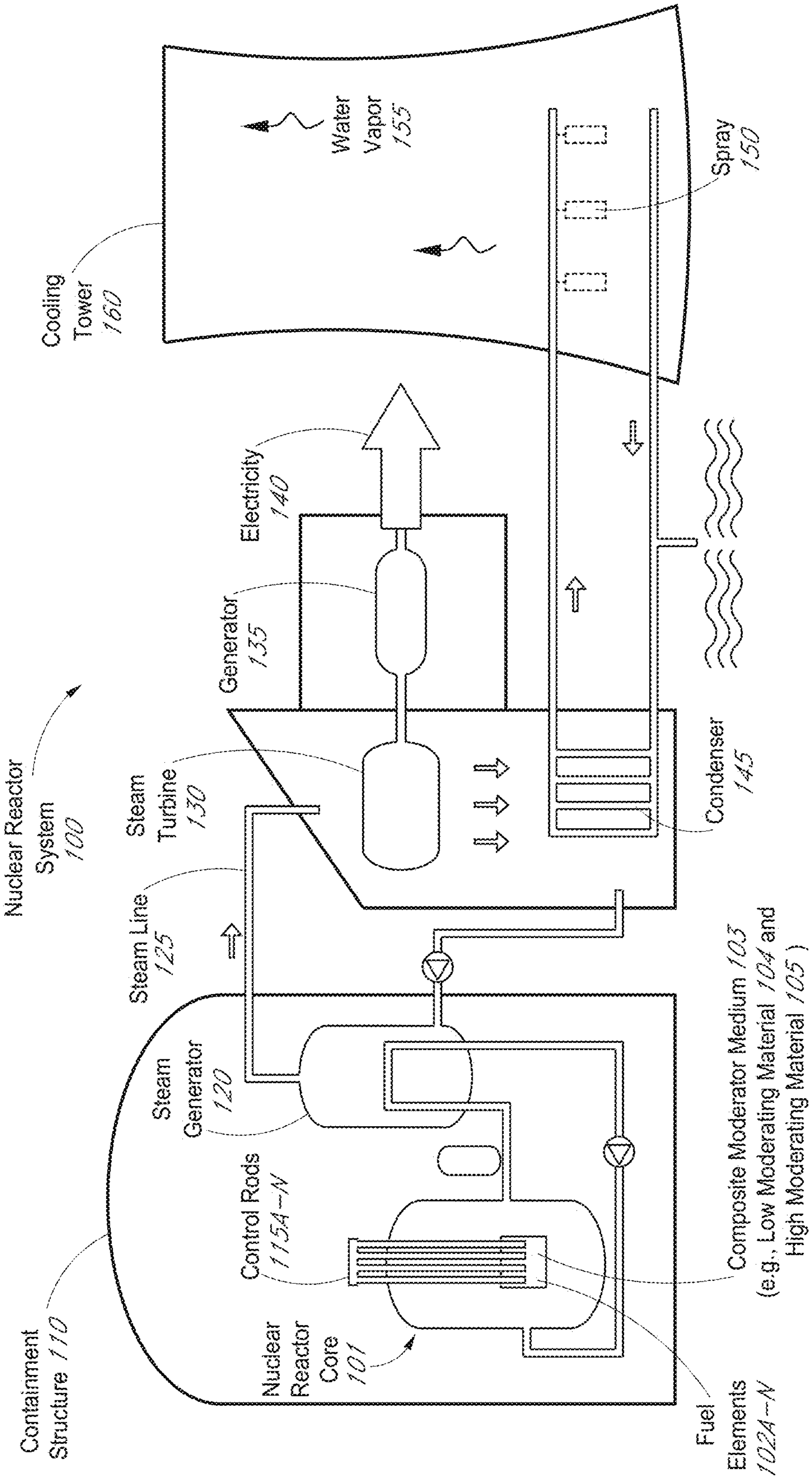

Nuclear Reactor System 100
Containment Structure 110
Nuclear Reactor Core 101
Control Rods 115A–N
Steam Generator 120
Steam Line 125
Steam Turbine 130
Generator 135
Electricity 140
Condenser 145
Spray 150
Water Vapor 155
Cooling Tower 160
Fuel Elements 102A–N
Composite Moderator Medium 103
(e.g., Low Moderating Material 104 and
High Moderating Material 105)

FIG. 1

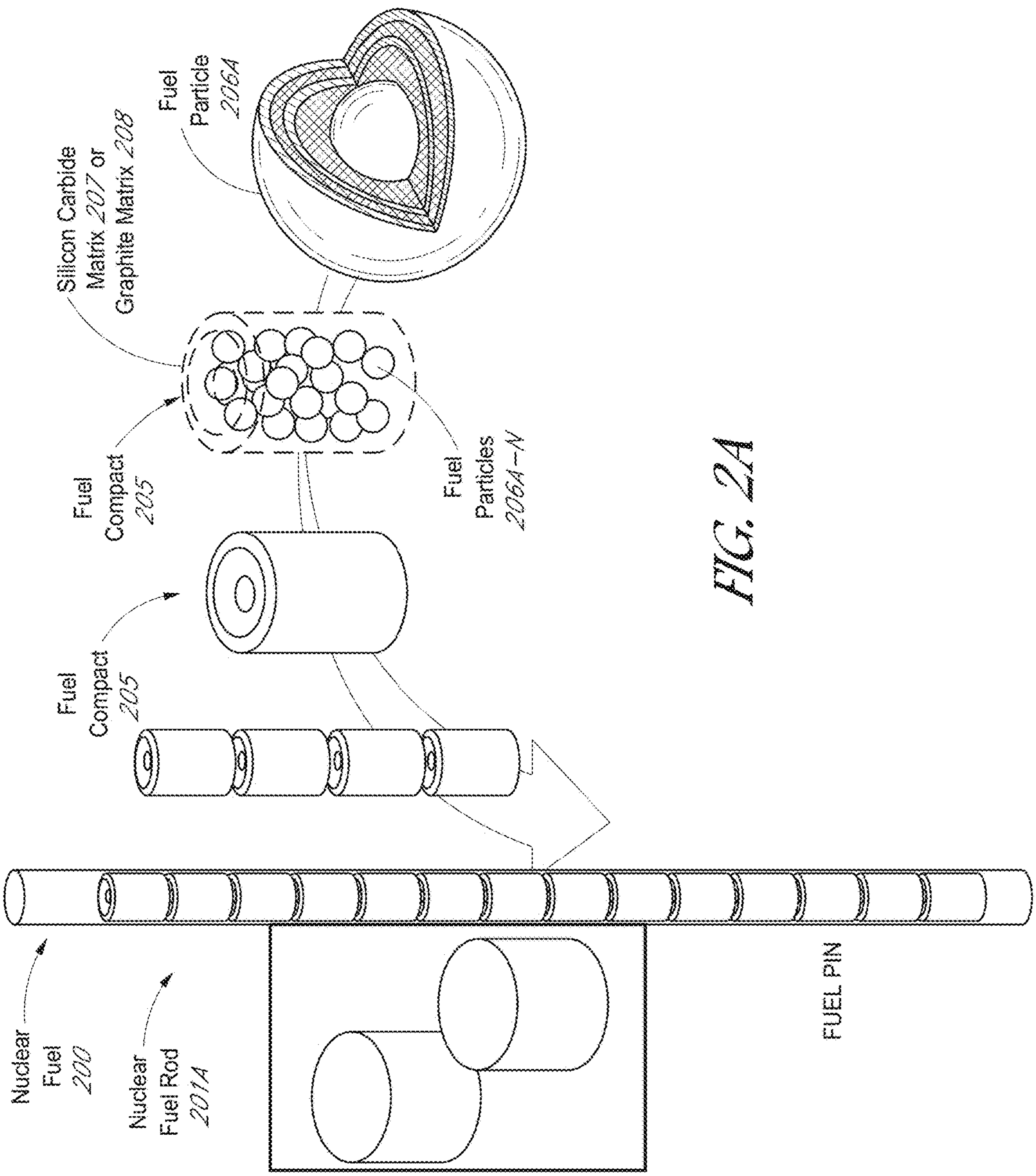
Silicon Carbide Matrix 207 or Graphite Matrix 208
Fuel Particle 206A
Fuel Compact 205
Fuel Particles 206A–N
Fuel Compact 205
Nuclear Fuel 200
Nuclear Fuel Rod 201A
FUEL PIN

FIG. 2A

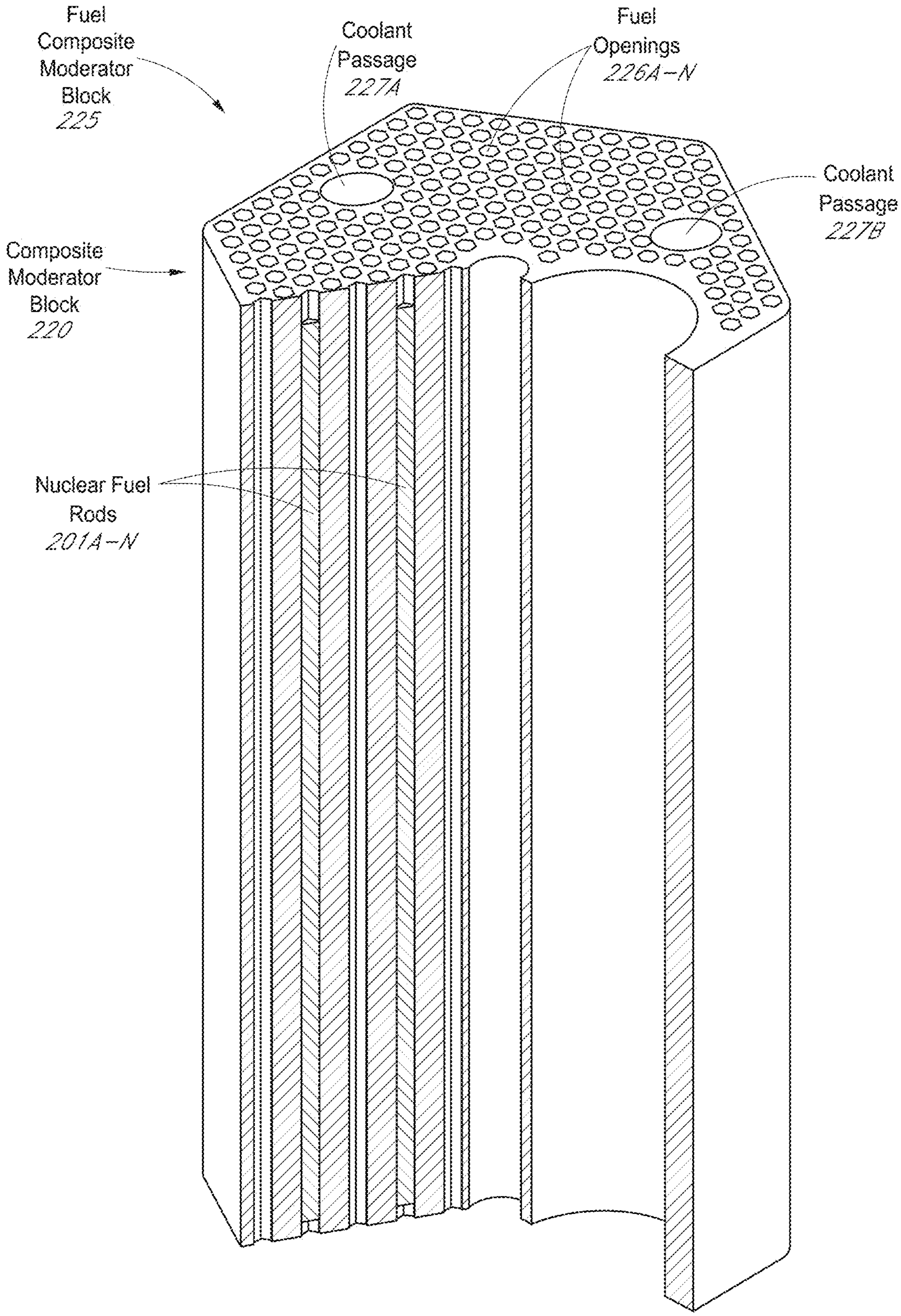
Fuel
Composite
Moderator
Block
225
Coolant
Passage
227A
Fuel
Openings
226A-N
Coolant
Passage
227B
Composite
Moderator
Block
220
Nuclear Fuel
Rods
201A-N

FIG. 2B

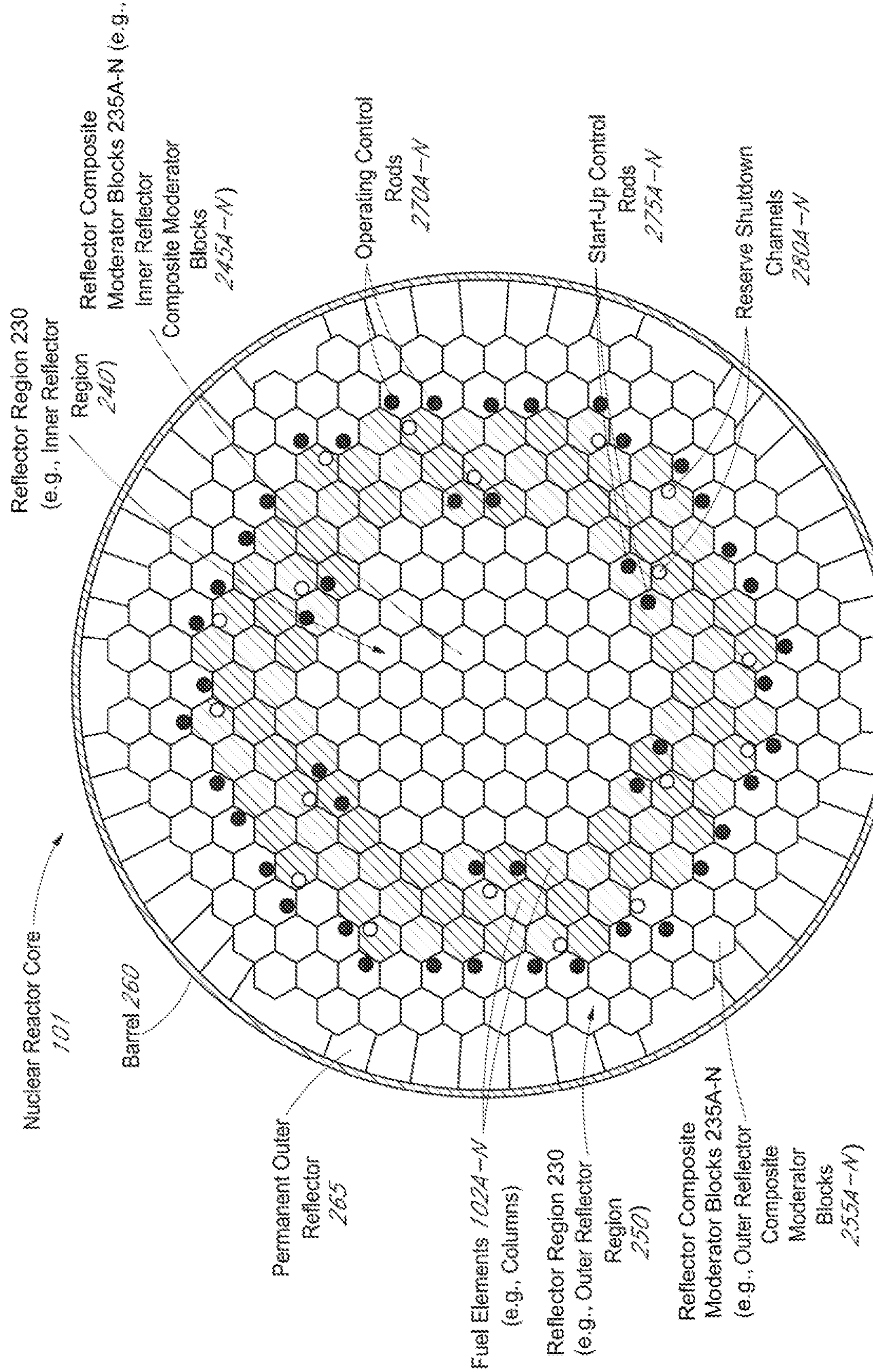
Nuclear Reactor Core 101
Barrel 260
Permanent Outer Reflector 265
Fuel Elements 102A–N (e.g., Columns)
Reflector Region 230 (e.g., Outer Reflector Region 250)
Reflector Composite Moderator Blocks 235A-N (e.g., Outer Reflector Composite Moderator Blocks 255A–N)
Reflector Region 230 (e.g., Inner Reflector Region 240)
Reflector Composite Moderator Blocks 235A-N (e.g., Inner Reflector Composite Moderator Blocks 245A–N)
Operating Control Rods 270A–N
Start-Up Control Rods 275A–N
Reserve Shutdown Channels 280A–N

FIG. 2C

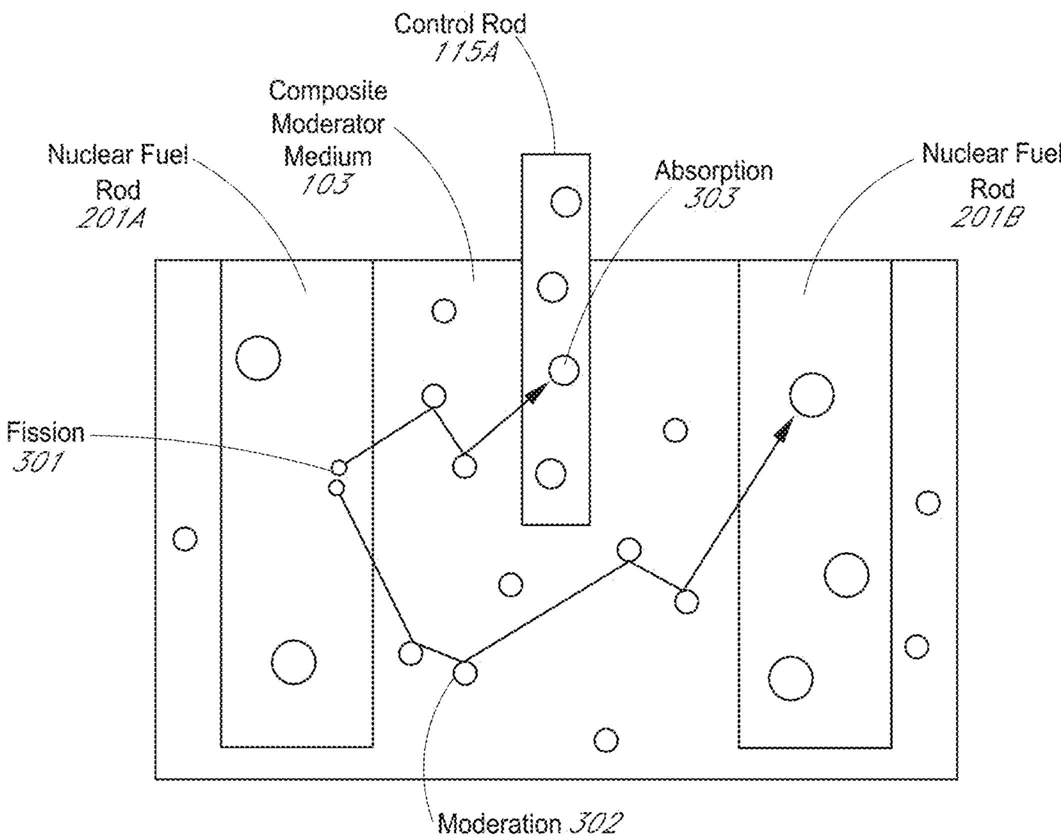
Control Rod
115A
Composite
Moderator
Medium
103
Nuclear Fuel
Rod
201A
Absorption
303
Nuclear Fuel
Rod
201B
Fission
301
Moderation 302

FIG. 3

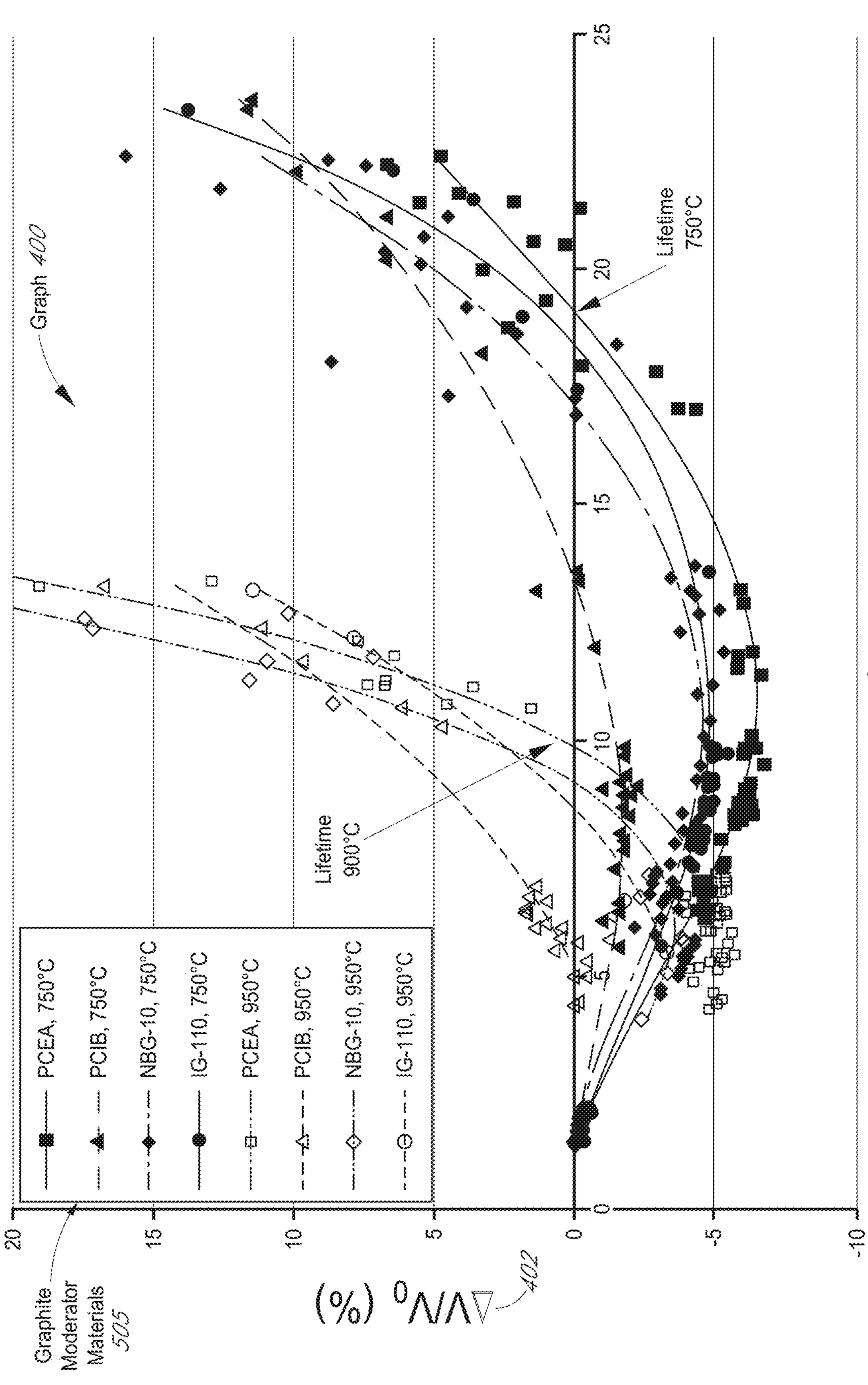
Graph 400
PCEA, 750°C
PCIB, 750°C
NBG-10, 750°C
IG-110, 750°C
PCEA, 950°C
PCIB, 950°C
NBG-10, 950°C
IG-110, 950°C
Lifetime 750°C
Lifetime 900°C
Graphite Moderator Materials 505
ΔV/V0 (%)
402
dpa
401
20
15
10
5
0
-5
-10
25

FIG. 4

Table 500

		510 Slowing Down Power $\xi\Sigma_s$	515 T_{melt} (T_{decomp}) °C	520 Density (g/cc)	525 Chemical Reactivity (O_2, N_2, H_2O)	530 Crystal	535 Irrad. Perf. (~500 °C, >20 dpa)	540 Therm. Cond. (@~650 °C) W/m-K
505	Graphite	0.077	>(3000)	1.8	O2:Rx@>800C	Hexag.	Poor	~20
104A	SiC	0.044	(2860)	3.2	Very Low	Cubic	Excellent	~80
104B	MgO	0.060	2853	3.58	Mildly Hyg.	Cubic	Excellent	~50
105A	Be_2B	0.120	1504	1.89		Cubic	unknown	unknown
105B	Be_4B	0.136	1160	1.94		Tetrag.	unknown	unknown
105C	Be_2C	0.125	(~2100)	1.9	N:Rx@>1100C O2:Rx@>800C	Cubic	unknown	~22
105D	$ZrBe_{13}$	0.129	1525	2.72		Cubic	unknown	~40
105E	$TiBe_{12}$	0.138	1925	2.26		Hexag.	unknown	
105F	BeO	0.124	2507	3.015	Very Low	Hexag.	Very Bad, anisotropic	~30
105G	$^{11}B_4C$	0.087	2350	2.52		Rhomb.	unknown	~12-15
105H	Be	0.16	1287	1.85	O2:Rx@>500C	Hexag.	Bad	~60

FIG. 5

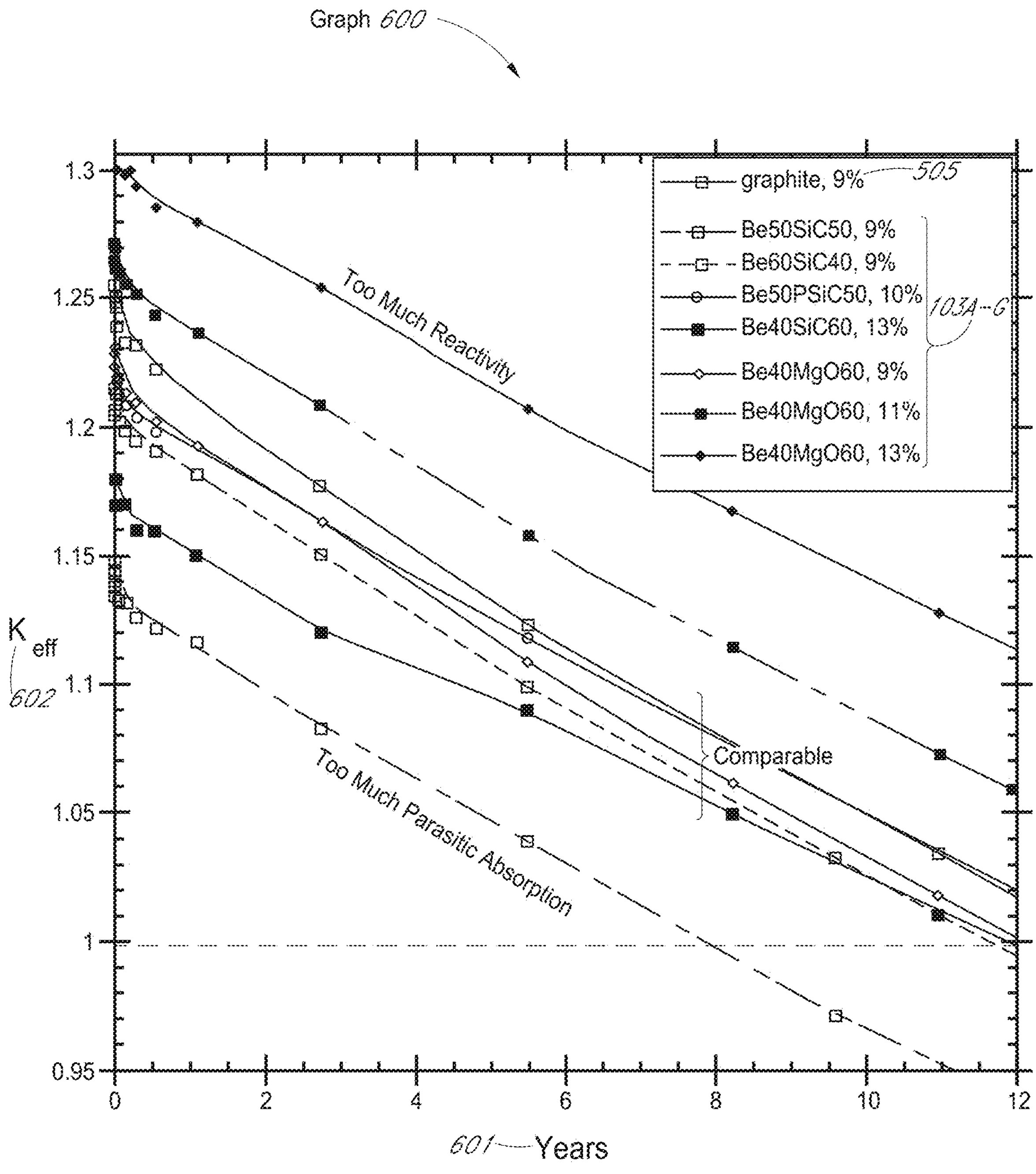
Graph 600
1.3
1.25
1.2
1.15
1.1
1.05
1
0.95
0
2
4
6
8
10
12
Keff
602
601
Years
graphite, 9%
505
Be50SiC50, 9%
Be60SiC40, 9%
Be50PSiC50, 10%
Be40SiC60, 13%
Be40MgO60, 9%
Be40MgO60, 11%
Be40MgO60, 13%
103A-G
Too Much Reactivity
Comparable
Too Much Parasitic Absorption

FIG. 6

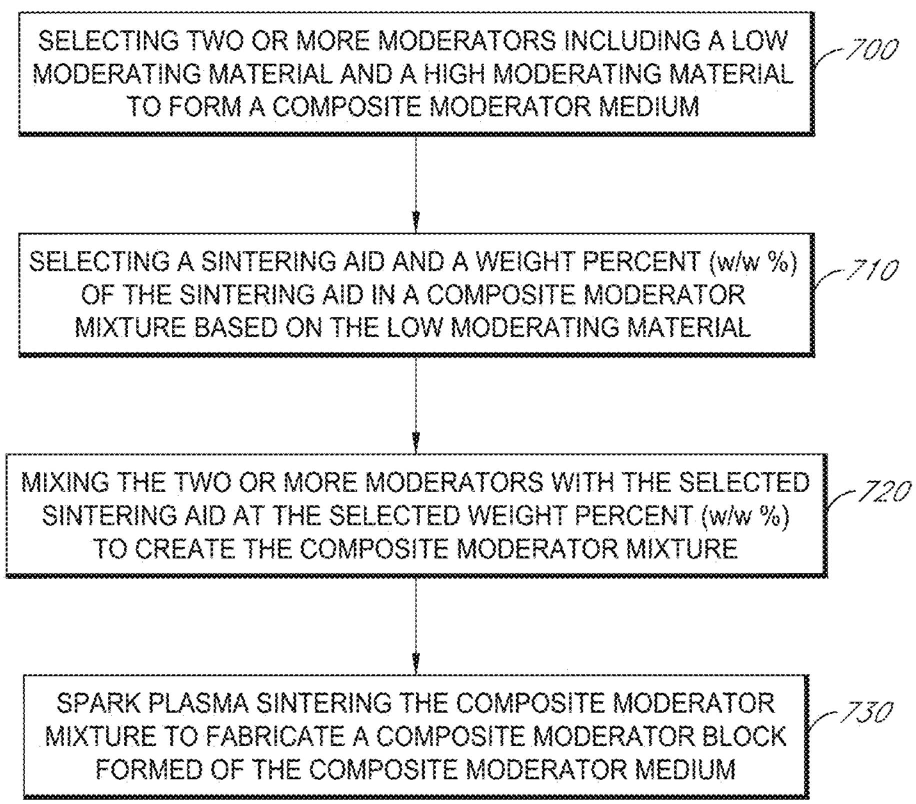
SELECTING TWO OR MORE MODERATORS INCLUDING A LOW MODERATING MATERIAL AND A HIGH MODERATING MATERIAL TO FORM A COMPOSITE MODERATOR MEDIUM
700
SELECTING A SINTERING AID AND A WEIGHT PERCENT (w/w %) OF THE SINTERING AID IN A COMPOSITE MODERATOR MIXTURE BASED ON THE LOW MODERATING MATERIAL
710
MIXING THE TWO OR MORE MODERATORS WITH THE SELECTED SINTERING AID AT THE SELECTED WEIGHT PERCENT (w/w %) TO CREATE THE COMPOSITE MODERATOR MIXTURE
720
SPARK PLASMA SINTERING THE COMPOSITE MODERATOR MIXTURE TO FABRICATE A COMPOSITE MODERATOR BLOCK FORMED OF THE COMPOSITE MODERATOR MEDIUM
730

FIG. 7

Spark Plasma Sintering (SPS) Photograph 800

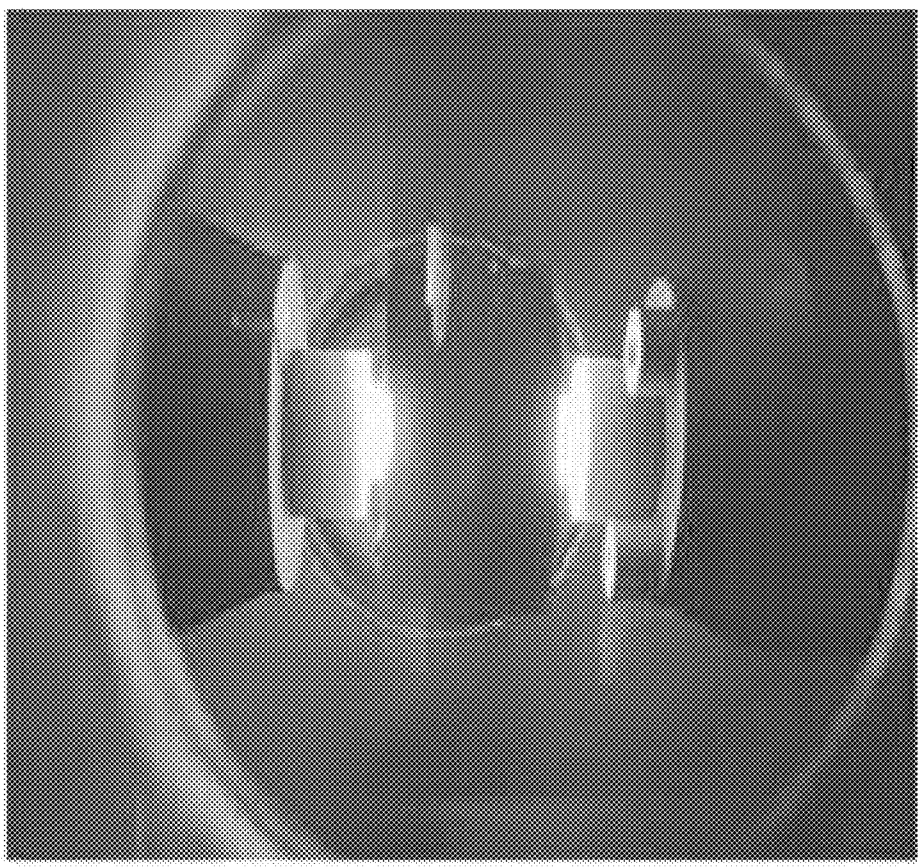

FIG. 8A

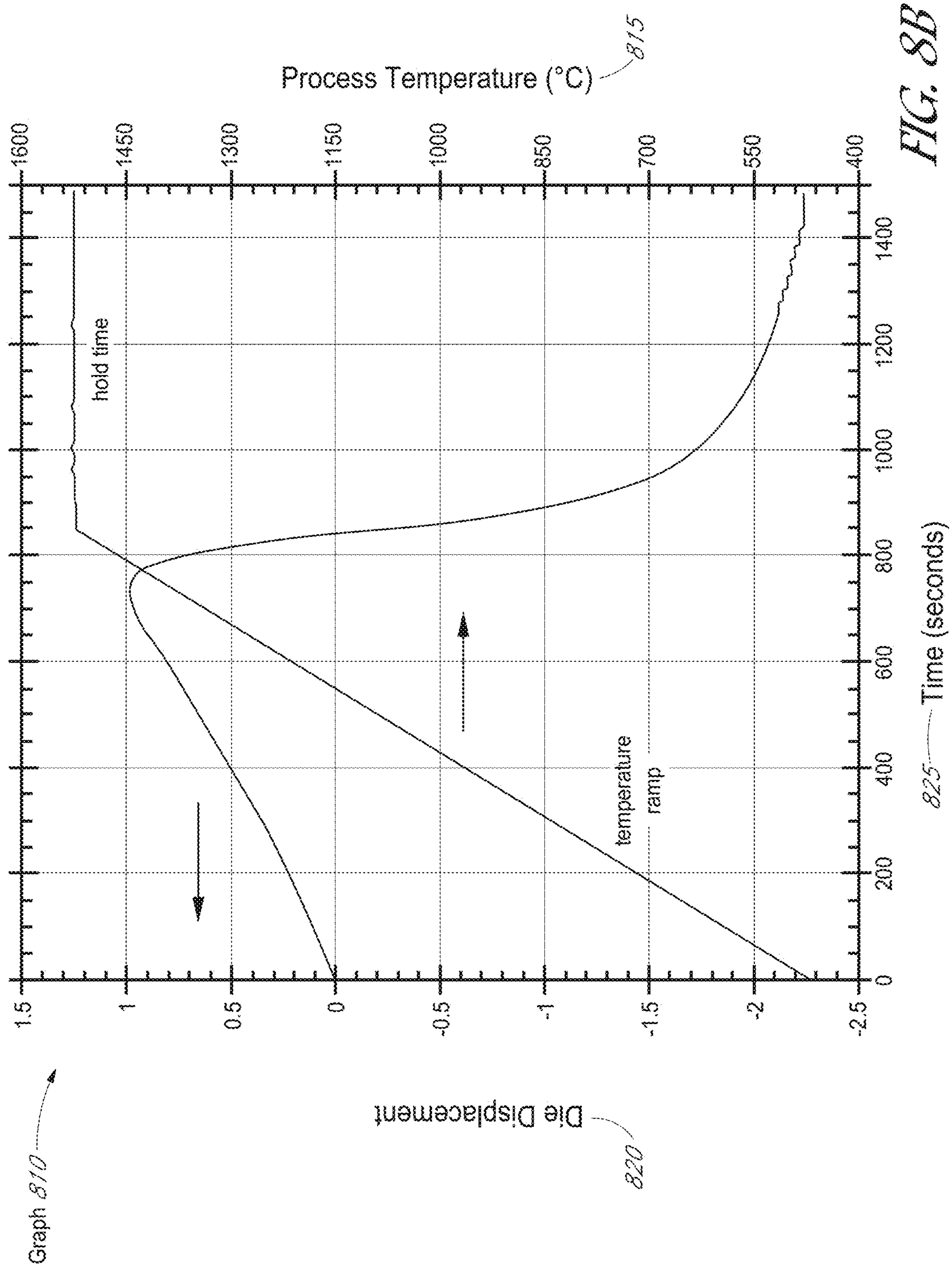
Process Temperature (°C)
815
1600
1450
1300
1150
1000
850
700
550
400
hold time
temperature ramp
0
200
400
600
800
1000
1200
1400
Time (seconds)
825
1.5
1
0.5
0
-0.5
-1
-1.5
-2
-2.5
Die Displacement
820
Graph 810

FIG. 8B

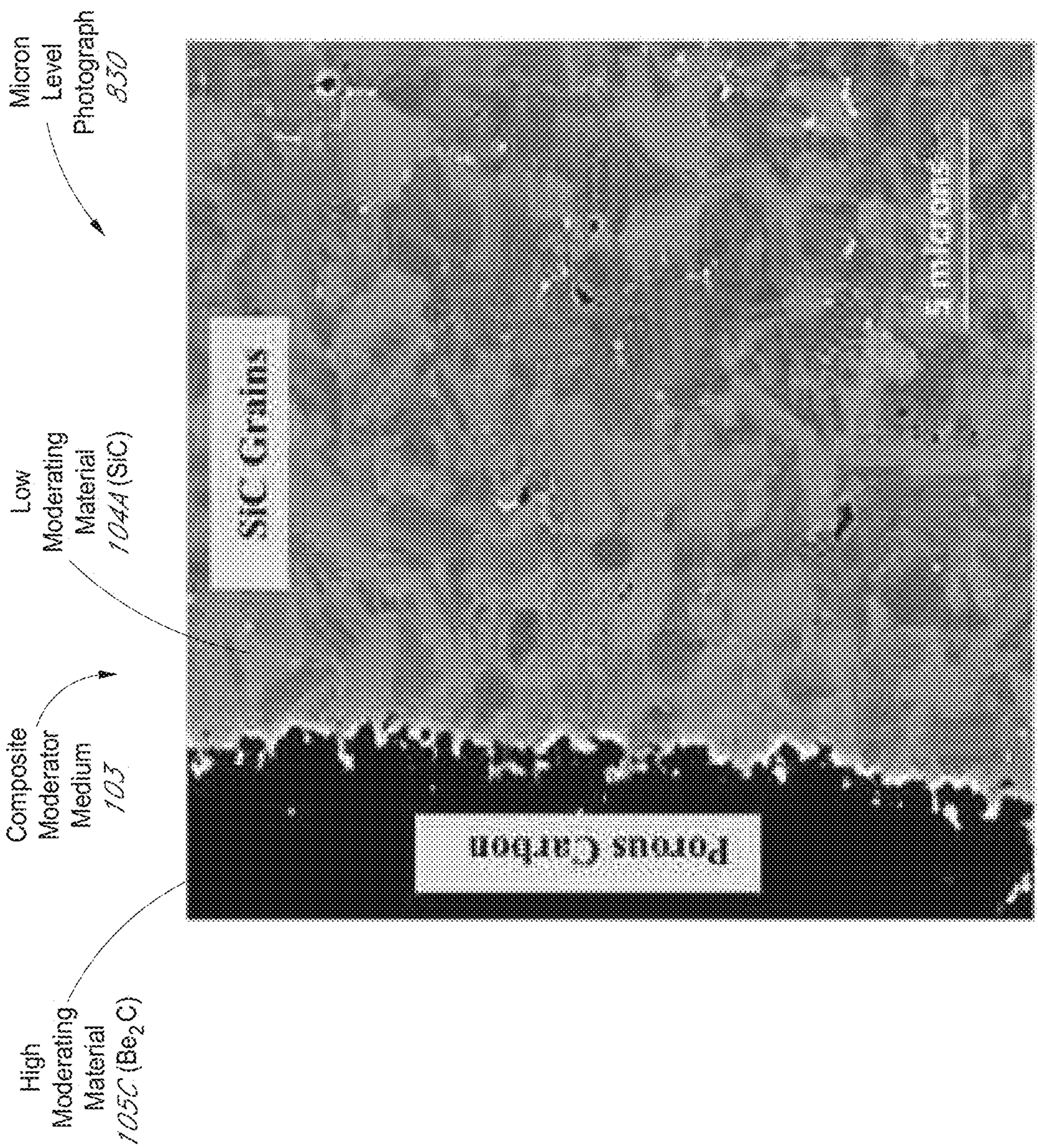
Micron Level Photograph 830
Low Moderating Material 104A (SiC)
Composite Moderator Medium 103
High Moderating Material 105C (Be_2C)
SiC Grains
Porous Carbon
5 microns

FIG. 8C

COMPOSITE MODERATOR FOR NUCLEAR REACTOR SYSTEMS

CROSS-REFERENCE TO RELATED APPLICATIONS

This application claims priority to U.S. Provisional Patent Application No. 62/619,925, filed on Jan. 22, 2018, titled "Composite Moderators for Nuclear Power Systems," the entire disclosure of which is incorporated by reference herein.

TECHNICAL FIELD

The present subject matter relates to examples of nuclear systems and nuclear reactor systems, which include a composite moderator. The present subject matter also encompasses a method for fabricating the composite moderator.

BACKGROUND

Nuclear fission reactors include thermal or fast type reactors. Currently, almost all operating reactors are thermal and thus require a moderator to slow down fast neutrons so that nuclear fission can continue. Materials used for moderation need to have a very specific set of properties. First, a moderator cannot absorb neutrons itself. Conventionally, this means that the moderator should have a low neutron absorption cross-section. However, the moderator should be able to slow down neutrons to an acceptable speed. Thus, in an ideal moderator the neutron scattering cross-section is high. This neutron scattering is a measure of how likely a neutron will interact with an atom of the moderator. If the collisions between neutrons and nuclei are elastic collisions, it implies that the closer in size the nucleus of an atom is to a neutron, the more the neutron will be slowed. For this reason, lighter elements tend to be more efficient moderators.

Commonly utilized moderators, such as light water (H_2O), heavy water (D_2O), and graphite (C) have a low neutron absorption cross-section but a comparatively large neutron scattering cross-section. Neutron scattering cross-sections (σ_s) for light water, heavy water, and graphite are: 49, 10.6, and 4.7 barns, respectively. Neutron absorption cross-sections (σ_s) for light water, heavy water, and graphite are: 0.66, 0.0013, and 0.0035 barns, respectively. The moderators vary in terms of their moderating abilities, as well as in their costs.

Currently, operating thermal nuclear reactors utilize a single moderator material that is monolithic as a neutron moderator. The monolithic moderator material is typically dug out of the ground. To determine the best-suited monolithic moderator material for a nuclear reactor core, engineers analyze whether the neutron properties of the monolithic moderator material are suitable for a nuclear reactor, last for a relatively long time, and then perform an optimization. Graphite is one type of neutron moderator that is commonly utilized in nuclear reactors. Graphite is a crystalline form of the element carbon with atoms arranged in a hexagonal structure that is naturally occurring. Graphite is the most stable form of carbon under standard conditions.

However, a single monolithic moderating material has limitations, one of which is moderator lifetime that is the physical limitation of the crystals of the monolithic moderating material as the single monolithic moderating material undergoes nuclear radiation inside a nuclear reactor core. Moreover, a high moderating material, such as graphite, is unstable under nuclear radiation, which causes the high moderating material to eventually deteriorate structurally before the nuclear fuel reaches the fuel lifetime limit.

Nuclear graphite was initially developed as a moderator for the Chicago Pile nuclear reactor (i.e., the world's first nuclear reactor) and is the first and arguably most studied nuclear material. Today, a number of gas-cooled systems (e.g., prismatic or pebble-bed) and salt-cooled systems assume very large graphite nuclear reactor core loadings. While there has been continual refinement in methods to process graphite as a single moderating material, providing both higher purity and better—more isotropic forms of graphite—a hard moderator lifetime limit remains for graphite. Essentially, the physics of irradiation-induced anisotropic crystal swelling leads to gross dimensional change of the graphite moderator material, microcracking, and loss of integrity of the graphite moderator material.

Typical high-temperature gas-cooled reactors (HTGR) of approximately 200 megawatt electrical (MWe) have an associated graphite loading of approximately 600 tons. Newly anticipated salt-cooled systems will have similarly large graphite waste streams. Unfortunately, contaminated graphite poses serious waste issues for these nuclear reactor systems as evidenced by the approximately 250,000 tons of graphite waste disposed of to date. While the level of contamination is dependent on the nuclear reactor system, nuclear fuel, and nuclear fuel quality, carbon-14 (^{14}C) and ^3T contamination are unavoidable. This nuclear waste issue is compounded by the fact that the graphite moderator lifetime for high-power (high neutron influence) systems mandate in-service change-out of significant volumes of the nuclear reactor core.

SUMMARY

The various examples disclosed herein relate to composite moderator technologies for nuclear reactor systems, including nuclear reactor cores including a composite moderator and a method for fabricating the composite moderator. Several benefits are achieved with the composite moderator over individual moderator materials, such as graphite, water, and molten salt (e.g., FLiBE which combines lithium fluoride and beryllium fluoride). First, the composite moderator reduces nuclear waste compared to the individual moderator materials by serving for the fuel lifetime of the nuclear fuel without requiring change out from the nuclear reactor core. Second, the composite moderator is dimensionally irradiation stable (i.e., undergoes less structural deterioration). Third, the composite moderator improves safety characteristics by eliminating the current graphite oxidation issue.

In a first example, a nuclear reactor system includes a nuclear reactor core. The nuclear reactor core includes an array of fuel elements and a composite moderator medium formed of two or more moderators. The two or more moderators include a low moderating material and a high moderating material. The high moderating material has a higher neutron slowing down power compared to the low moderating material.

In a second example, a method includes selecting two or more moderators including a low moderating material and a high moderating material to form a composite moderator medium. The method further includes selecting a sintering aid and a weight percent (w/w %) of the sintering aid in a composite moderator mixture based on the low moderating material. The method further includes mixing the two or more moderators with the selected sintering aid at the selected weight percent (w/w %) to create the composite moderator mixture. The method further includes spark plasma sintering the composite moderator mixture to fabricate a composite moderator block formed of the composite moderator medium.

Additional objects, advantages and novel features of the examples will be set forth in part in the description which follows, and in part will become apparent to those skilled in the art upon examination of the following and the accompanying drawings or may be learned by production or operation of the examples. The objects and advantages of the present subject matter may be realized and attained by means of the methodologies, instrumentalities and combinations particularly pointed out in the appended claims.

BRIEF DESCRIPTION OF THE DRAWINGS

The drawing figures depict one or more implementations in accordance with the present concepts, by way of example only, not by way of limitations. In the figures, like reference numerals refer to the same or similar elements.

FIG. **1** is an illustration of a nuclear reactor system that depicts a nuclear reactor core, control rods, and other components of the assembly.

FIG. **2**A is an illustration of fuel particles and a fuel compact of nuclear fuel utilized in the nuclear reactor core of FIG. **1**.

FIG. **2**B is an illustration of a fuel composite moderator block of the nuclear reactor core of FIG. **1**, which includes the nuclear fuel of FIG. **2**A and is formed of a composite moderator medium.

FIG. **2**C is a cross-sectional view of the nuclear reactor core and components, including an array of fuel elements and reflector composite moderator blocks formed of the composite moderator medium.

FIG. **3** is an enlarged plane view of a portion of the fuel composite moderator block of FIG. **2**B depicting the nuclear fuel enclosed by the composite moderator medium.

FIG. **4** is a graph illustrating dimensional change of a graphite moderator material over time while in a nuclear reactor core.

FIG. **5** is a table depicting properties, including neutron slowing down power, of a graphite moderator material compared with two types of low moderating materials of the composite moderator medium and eight types of high moderating materials of the composite moderator medium.

FIG. **6** is a graph illustrating reactivity over time of the nuclear reactor core that includes the graphite moderator material of FIG. **5** compared with seven different types of composite moderate media.

FIG. **7** is a flowchart of a method that can be implemented to fabricate a composite moderator block of the composite moderator medium.

FIG. **8**A is a processing photograph of the method of FIG. **7** in which spark plasma sintering (SPS) is utilized to fabricate the composite moderator block.

FIG. **8**B is a graph illustrating a processing temperature and pressure (die displacement) over time during sparking plasma sintering of the method of FIG. **7**.

FIG. **8**C is a micron level photograph of the composite moderator medium showing the low moderating material encapsulating the high moderating material.

PARTS LISTING

100 Nuclear Reactor System
101 Nuclear Reactor Core
102A-N Fuel Elements
103 Composite Moderator Medium
104 Low Moderating Material
105 High Moderating Material
110 Containment Structure
115A-N Control Rods
120 Steam Generator
125 Steam Line
130 Steam Turbine
135 Generator
140 Electricity
145 Condenser
150 Spray
155 Water Vapor
160 Cooling Tower
200 Nuclear Fuel
201A-N Nuclear Fuel Rods
205 Fuel Compact
206A-N Fuel Particles
207 Silicon Carbide Matrix
208 Graphite Matrix
220 Composite Moderator Block
225 Fuel Composite Moderator Block
226A-N Fuel Openings
227A-B Coolant Passages
228 Coolant
230 Reflector Region
235A-N Reflector Composite Moderator Blocks
240 Inner Reflector Region
245A-N Inner Reflector Composite Moderator Blocks
250 Outer Reflector Region
255A-N Outer Reflector Composite Moderator Blocks
260 Barrel
265 Permanent Outer Reflector
270A-N Operating Control Rods
275A-N Start-Up Control Rods
280A-N Reserve Shutdown Channels

DETAILED DESCRIPTION

In the following detailed description, numerous specific details are set forth by way of examples in order to provide a thorough understanding of the relevant teachings. However, it should be apparent to those skilled in the art that the present teachings may be practiced without such details. In other instances, well known methods, procedures, components, and/or circuitry have been described at a relatively high-level, without detail, in order to avoid unnecessarily obscuring aspects of the present teachings.

To deal with nuclear radiation, a composite moderator medium (e.g., solid material) is engineered that includes several moderating materials that change in a direction that collectively last a longer duration than a single moderator material. The composite moderator medium enables nuclear reactor cores to have an extended lifetime without swap out of the moderator material and to be more compact compared to graphite moderator material, for example. In addition, the composite moderator medium can be deployed in various nuclear reactor system implementations, such as a terrestrial land reactor for electricity generation or a high temperature nuclear thermal propulsion (NTP) system (e.g., compact space nuclear reactor).

Reference now is made in detail to the examples illustrated in the accompanying drawings and discussed below.

FIG. **1** is an illustration of a nuclear reactor system **100** that depicts a nuclear reactor core **101**, control rods **115**A-N, and other components of the assembly. In the example, the nuclear reactor system **100** includes a nuclear reactor **101** in which a controlled nuclear chain reactions occurs, and energy is released. In this example, the nuclear reactor system **100** is a nuclear power plant in a terrestrial land application. However, nuclear reactors and the composite moderator technologies can be utilized in a space environment, such as in a nuclear thermal propulsion (NTP) system. In such an NTP system, a generated thrust propels a vehicle that houses, is formed integrally with, connects, or attaches to the nuclear reactor core **101**, such as a rocket, drone, unmanned air vehicle (UAV), aircraft, spacecraft, missile, etc. In addition, the NTP system can be used in the propulsion of submarines or ships.

Nuclear reactor core **101** includes an array of fuel elements **102**A-N and a composite moderator medium **103**. The nuclear reactor core **101** is a nuclear fission reactor core that includes nuclear fuel to generate megawatts or more of thermal power (MWt). A plurality of circumferential control rods **115**A-N may surround the array of fuel elements **102**A-N to change reactivity of the nuclear reactor core **101** by rotating the control rods **115**A-N. Containment structure **110** houses the nuclear reactor core **101**, control rods **115**A-N, and a steam generator **120**. Control rods **115**A-N may be positioned in an area of the reflector regions **240**, **250** (see FIG. **2**C) of the nuclear reactor core **101** to regulate the neutron population and reactor power level during operation by changing reactivity of the nuclear reactor core **101**.

Control rods **115**A-N are composed of chemical elements such as boron, silver, indium, and cadmium that are capable of absorbing many neutrons without themselves fissioning. Nuclear reactor core **101** creates thermal energy, which is released as heat. Other components of the nuclear reactor system **100** convert the thermal energy into a useful form of energy, such as electricity **140**. In the example, the nuclear reactor core **101** provides thermal energy to the steam generator **120**, which extracts thermal energy into steam line **125**, which turns a steam turbine **130**. Steam turbine **130** drives the generator **135**, which then converts the thermal energy into electricity **140**. Subsequently, the thermal expansion cycle repeats.

In the example nuclear reactor system **100**, a condenser **145** produces a coolant, such as a high-pressure liquid or gas, for feeding the nuclear reactor core **101** and cooling the components of the nuclear reactor system **100**. For example, during the expansion cycle, the coolant stored in a cooling tower **160** can be drawn through the nuclear reactor core **101** to cool the nuclear reactor core **101**. Heat from the coolant may be extracted into a cooling tower **160** as spray **150** and released as water vapor **155** from the cooling tower **160**. Of note, some of the coolant may be returned, for example, bled from the nuclear reactor core **101** via a bypass to turn the steam turbine **130**. In some examples, the nuclear reactor system **100** can be used in a molten salt loop application.

The neutron chain reaction in the nuclear reactor core **101** is critical—a single neutron from each fission nucleus results in fission of another nucleus—the chain reaction must be controlled. The composite moderator medium **103** is formed of two or more moderators, which effectively regulates the criticality and provides an extended moderator lifetime that can match the nuclear fuel lifetime. The two or more moderators include a low moderating material **104** and a high moderating material **105**. The high moderating material **105** has a higher neutron slowing down power compared to the low moderating material **104**, which can correlate to the neutron absorption cross-section and the neutron scattering cross-section. The composite moderator medium **103** in the nuclear reactor core **101** slows down the fast neutrons (produced by splitting atoms in fissile compounds like uranium-235), to make them more effective in the nuclear fission chain reaction. This slowing or moderation of the neutrons allows the neutrons to be more easily absorbed by fissile nuclei, creating more fission events. The two or more moderators can be adapted to a very specific set of properties depending on the implementation environment of the nuclear reactor core **101** (e.g., electricity generation or NTP).

As will be further explained in FIGS. **5**-**6**, the low moderating material **104** includes a moderating matrix of silicon carbide (SiC) **104**A or magnesium oxide (MgO) **104**B. The high moderating material **105** is dispersed within the moderating matrix and includes beryllium (Be) **105**H, boron (B), or a compound thereof. More specifically, the high moderating material **105** includes at least one of beryllium boride (Be_2B **105**A, Be_4B **105**B, BeB_2, or BeB_6), beryllium carbide (Be_2C **105**C), zirconium beryllide ($ZrBe_{13}$ **105**D), titanium beryllide ($TiBe_{12}$ **105**E), beryllium oxide (BeO **105**F), or boron carbide ($^{11}B_4C$ **105**G). The high moderating material **105** is encapsulated within the low moderating material **104** such that the high moderating material **105** is not exposed outside of the low moderating material **104**.

FIG. **2**A is an illustration of fuel particles **206**A-N and a fuel compact **205** (e.g., fuel pellet) of nuclear fuel **200** utilized in the nuclear reactor core **101** of FIG. **1**. In one example, the nuclear fuel **200** includes a fuel compact **205** comprised of tristructural-isotropic (TRISO) fuel particles **206**A-N embedded inside a silicon carbide matrix **207**. In another example, the nuclear fuel **200** includes tristructural-isotropic (TRISO) fuel particles **206**A-N embedded inside a graphite matrix **208** to create fuel pellets. TRISO fuel particles **206**A-N include a fuel kernel composed of UC or uranium oxycarbide (UCO) in the center, coated with one or more layers surrounding one or more isotropic materials. As shown in FIG. **2**A, TRISO fuel particles **206**A-N include four layers of three isotropic materials. In that example, the four layers are: (1) a porous buffer layer made of carbon, followed by (2) a dense inner layer of pyrolytic carbon (PyC), followed by (3) a ceramic layer of SiC to retain fission products at elevated temperatures and to give the TRISO fuel particle **206**A a strong structural integrity, followed by (4) a dense outer layer of PyC.

TRISO fuel particles **206**A-N are designed not to crack due to the stresses or fission gas pressure at temperatures beyond 1,600° C., and therefore can contain the fuel in the worst of accident scenarios. TRISO fuel particles **206**A-N were designed for use in high-temperature gas-cooled reactors (HTGR) like the example cross-section of the nuclear reactor core **101** shown in FIG. **2**C, to be operating at temperatures much higher than the temperatures of LWRs. The fuel compacts **205** can be loaded into fuel pins or rods, cladded, and stacked inside the numerous columns of fuel elements **102**A-N. Of the possible matrix **207**, **208** materials for the TRISO fuel particles **206**A-N, silicon carbide (SiC) offers good irradiation behavior, and fabrication. SiC has excellent oxidation resistance due to rapid formation of a dense, adherent silicon dioxide (SiO_2) surface scale on exposure to air at elevated temperature, which prevents further oxidation.

The use of coated fuel particles **206**A-N makes it more difficult to achieve high heavy metal density in the nuclear fuel **200**, since the net heavy metal density within a fuel particle **206** falls rapidly with increasing coating thickness. This fact requires that the coating thickness to kernel diameter ratio be kept as small as possible while maintaining utility as a fission product barrier. It is, however, clear that the use of dispersion fuels in LWRs will demand higher enrichment and a lower power density. The most likely fissile particle types for composite nuclear fuels are uranium/plutonium carbides (UC or PuC) and uranium/plutonium nitrides (UN or PuN) due to the combination of high melting temperature and high actinide density. Uranium silicides could provide an even higher density of fissile uranium, but may be unstable under the expected fabrication and operation conditions. Other types of fuel particles **206**A-N can be utilized including QUADRISO fuel, which includes one or more burnable neutron poisons surrounding the fuel kernel of TRISO particles, such as erbium oxide, to better manage excess reactivity, as well CerMet fuel (e.g., ceramic fuel particles **206**A-N, such as uranium oxide), embedded in a metal matrix), etc.

In some examples of nuclear fuel **200**, uranium dioxide (UO_2) powder is compacted to create a cylindrical shaped fuel compact **205** and sintered at high temperatures to produce ceramic nuclear fuel pellets with a high density and well defined physical properties and chemical composition. A grinding process is used to achieve a uniform cylindrical geometry with narrow tolerances.

FIG. **2**B is an illustration of a fuel composite moderator block **225** of the nuclear reactor core **101** of FIG. **1**, which includes the nuclear fuel **200** of FIG. **2**A and is formed of a composite moderator medium **103**. Each of the fuel elements **102**A-N (as shown in FIG. **2**C) includes a composite moderator block **220** formed of the composite moderator medium **103** and nuclear fuel **200**. The fuel composite moderator block **225** includes fuel openings **226**A-N. The nuclear fuel **200** is disposed inside the fuel openings **226**A-N, such that the nuclear fuel **200** is enclosed by the composite moderator medium **103**. The fuel composite moderator block **225** further includes coolant passages **227**A-B to flow a coolant **228**, such as a gas or liquid.

Many such fuel compacts **205** (as shown in FIG. **2**A) are stacked and filled into the depicted nuclear fuel rods **201**A-N (e.g., sealed tubes). Cladding is an outer layer of the nuclear fuel rods **201**A-N that prevents radioactive fission fragments from escaping from the nuclear fuel **200** into the coolant **228** and contaminating the coolant **228**. The metal used for the cladding of the nuclear fuel rods **201**A-N depends on the design of the nuclear reactor core **101**, but can include stainless steel, magnesium with aluminum, or a zirconium alloy which, in addition to being highly corrosion-resistant, has low neutron absorption. The finished nuclear fuel rods **201**A-N are grouped into fuel assemblies that are used to build up the nuclear reactor core **101**, as described in FIG. **2**C.

FIG. **2**C is a cross-sectional view of the nuclear reactor core **101** and components, including an array of fuel elements **102**A-N and various reflector composite moderator blocks **245**A-N, **255**A-N, formed of the composite moderator medium **103**. Typically, the nuclear reactor core **101** includes at least one reflector region **230** (shown as inner reflector region **240** and outer reflector region **250**) that includes reflector composite moderator blocks **235**A-N formed of the composite moderator medium **103**. In the example arrangement of FIG. **2**C, nuclear reactor core **101** includes an inner reflector region **240** and an outer reflector region **250**. The inner reflector region **240** includes inner reflector composite moderator blocks **245**A-N. The outer reflector region **250** includes outer reflector composite moderator blocks **255**A-N. The inner reflector composite moderator blocks **245**A-N and the outer reflector composite moderator blocks **255**A-N are formed of the composite moderator medium **103**. The array of fuel elements **102**A-N, which forms a hexagonal shaped fuel block region, surrounds the inner reflector region **240**. The outer reflector region **250** surrounds the array of fuel elements **102**A-N. Hence, the fuel elements **102**A-N are interspersed between the inner reflector region **240** and the outer reflector region **250**.

In FIG. **2**C, the nuclear reactor core **101** is a prismatic high-temperature gas nuclear reactor core **101**. Of course, the composite moderator technologies can be utilized within any nuclear reactor core **101** that is not gas based. In the example gas nuclear reactor core **101**, the composite moderator block **220** is a block of composite moderator medium **103**, which is a solid material formed of both the low moderating material **104** and the high moderating material **105**. Composite moderator block **220** may be prismatic shaped (e.g., hexagonally shaped) and include various openings (holes) drilled in, such as a coolant passages **227**A-B to flow the coolant **228** and fuel openings **226**A-N to put the nuclear fuel **200** inside. Many nuclear fuel rods **201**A-N of the nuclear fuel **200** are dropped into each composite moderator block **220** to create each fuel composite moderator block **225** (e.g., a fuel bundle). The fuel composite moderator blocks **225**A-N (e.g., fuel bundles) are then loaded in the nuclear reactor core **101**.

Many composite moderator blocks **220** are fueled, which are shown as fuel composite moderator blocks **225**A-N. The majority of the composite moderator blocks **220** are not fueled (i.e., no nuclear fuel **200** is present), and thus do not include fuel openings **226**A-N. These reflector composite moderator blocks **235**A-N (shown as the inner reflector composite moderator blocks **245**A-N and outer reflector composite moderator blocks **255**A-N) include coolant passages **227**A-B to flow the coolant **228**.

The array of fuel elements **102**A-N includes hundreds of hexagonal shaped fuel composite moderator blocks **225**A-N, of which there are **102** fuel columns in the example, where each fuel column is ten (10) fuel composite moderator blocks **225**A-J high. The cross-section of the nuclear reactor core **101** is approximately six meters wide and each composite moderator block **220** is approximately 30 cm wide. In the middle of the cross-section of the nuclear reactor core **101** are the inner reflector composite moderator blocks **245**A-N, which includes many columns of inner reflector composite moderator blocks **245**A stacked ten per column. On the outside of the cross-section of the nuclear reactor core **101** are the outer reflector composite moderator blocks **255**A-N, which includes many columns of outer reflector composite moderator blocks **255**A-N stacked ten per column. Typically the control rods **115**A-N, columns of fuel composite moderator blocks **225**A-N, inner reflector composite moderator blocks **245**A-N, and outer reflector composite moderator blocks **255**A-N are the same length; however, it should be understood that the lengths can differ depending on the implementation.

Control rods **115**A-N as shown in FIG. **1** include both operating control rods **270**A-N and start-up control rods **275**A-N, which are inserted through the top of the nuclear reactor core **101** and pass through a subset of the composite moderator blocks **220**. Thirty-six (36) operating control rods **270**A-N pass through a subset of the outer reflector composite moderator blocks **255**A-N. Twelve (12) start-up control rods **275**A-N pass through a subset of fuel composite moderator blocks **225**A-N. The control rods **270**A-N, **275**A-N absorb neutrons. Barrel **260**, such as a metal core support formed of steel, surrounds the bundled collection of array of fuel elements **102**A-N, inner reflector region **240**, and outer reflector region **250** of the nuclear reactor core **101** on the periphery of the nuclear reactor core **101**. A permanent outer reflector **265**, which can be formed of the composite moderator medium **103**, is disposed between the outer reflector region **250** and the barrel **260**. The permanent outer reflector **265** includes partially hexagonally shaped filler elements which surround the perimeter of the outer reflector region **250** that make up the nuclear reactor core **101**. Eighteen (18) reserve shutdown channels **280**A-N are positioned in area within the array of fuel elements **102**A-N.

Generally, the composite moderator blocks **220**A-N utilized in the fuel composite moderator blocks **225**A-N, inner reflector composite moderator blocks **245**A-N, and outer reflector composite moderator blocks **255**A-N are all formed of the same composite moderator medium **103**, have the same profile shape (e.g., helical), and have coolant passages **227**A-B to flow the coolant **228**. However, the composite moderator block **220** for the inner reflector composite moderator blocks **245**A-N and outer reflector composite moderator blocks **255**A-N does not include drilled-in fuel openings **226**A-N.

Thus, from one-hundred feet away, the nuclear reactor core **101** appears to have many large hexagonal composite moderator blocks **220**A-N that look nearly identical, but the fuel block region (e.g., central hexagonal shaped part of the nuclear reactor core **101**) that contains the array of fuel elements **102**A-N happens to include nuclear fuel **200** disposed inside the composite moderator blocks **220**A-N.

FIG. **3** is an enlarged plane view of a portion of the fuel composite moderator block **225** of FIG. **2**B depicting the nuclear fuel **200** enclosed by the composite moderator medium **103**. Between blocks of fissile material of two nuclear fuel rods **201**A-B, a single control rod **115**A strongly absorbs **303** neutrons. The deeper the control rod **115**A is inserted in between the fissile material of nuclear fuel rods **201**A-B, the more difficult for the neutrons produced from fission **301** to collide, resulting in a more limited chain reaction, and a decrease in thermal energy production. As shown, composite moderator medium **103** fills the voids between fuel rods **201**A-B, reducing neutron energy by slowing (moderating **302**) the neutrons. Without the composite moderator medium **103**, the neutrons will move too fast and thus have a very low probability of causing fission **301** of ^{235}U and so these neutrons will pass by many, many nuclei before being absorbed and inducing fission.

FIG. **4** is a graph **400** illustrating dimensional change of a graphite moderator material **505** over time while in a nuclear reactor core **101**. In the graph **400**, displacement per atom (dpa) **401** is shown on the x axis over lifetime in the nuclear reactor core and delta ΔV/V (%) **402** is dimensional change of the graphite moderator material **505**. Large dimensional changes occur in nuclear graphite moderator material **505** due to neutron irradiation with implied lifetime in the 10-20 dpa (displacement per atom) range. Typically, high-power high-temperature gas-cooled reactors (HTGRs) need the nuclear reactor core graphite changed once or twice during plant lifetime.

Four different types of graphite moderator materials **505** are plotted in the graph **400**, at both 750° Celsius (C.) and 900° Celsius (C.). The temperature of the graphite moderator material **505** within the nuclear reactor core is a function of position, which has great variability. The graphite moderator material **505** can be 750° Celsius (C.) at or near the bottom of the nuclear reactor core and 900° Celsius (C.) at or near the top of the nuclear reactor core. Hence, the graphite moderator material **505** has a different lifetime depending on the particular position or placement in the nuclear reactor core.

As shown in the graph **400**, while in the nuclear reactor core, the graphite moderator material undergoes a big dimensional change and then falls apart and that is called "moderator lifetime." While this swelling mechanism is somewhat compensated by nuclear graphite's native porosity (essentially all nuclear graphite materials are approximately 18% porous), the moderator lifetime is associated with the point at which graphite returns to its original nil-swelling value, or the zero-point for the two traces inset into the graph of FIG. **4** for a range of the four plotted types of nuclear graphite moderator materials **505**. Thus, the problem with the graphite moderator material **505** is a very limited and defined moderator lifetime from radiation damage during operating time in the nuclear reactor core.

FIG. **5** is a table **500** depicting properties, including neutron slowing down power, of a graphite moderator material **505** compared with two types of low moderating materials **104**A-B of the composite moderator medium **103** and eight types of high moderating materials **105**A-H of the composite moderator medium **103**. As can be seen, the goal is to place (e.g., encapsulate) a high moderating material **105**A-H with poor radiation performance inside a low moderating material **104**A-B with good radiation performance, so the moderator lifetime of the composite moderator medium **103** is significantly extended compared to the graphite moderator material **505**. As can be seen in the table **500**, the slowing down power **510** of the low moderating materials **104**A-B are relatively lower than the high moderating materials **105**A-H. Other comparative properties included in the table **500** are melting temperature (T_{melt}) **515**, density **520**, chemical reactivity **525**, crystal **530**, irradiation performance **535**, and thermal conductivity **540**.

The composite moderator medium **103** has substantially lower dimensional change without the need for replacement. Composite moderator medium **103** is a material that can last the fuel lifetime of the nuclear reactor core **101**. This is achieved by matching the neutron moderation through a two-phase structure while increasing irradiation stability through the use of a superior moderating matrix material for the low moderating material **104**, such as SiC **104**A or MgO **104**B.

The choice of a second high moderating material **105** for an encapsulated or entrained phase in the moderating matrix phase of the low moderating material **104** is driven by the need to enhance moderation, as understood by inspection of the table **500** of FIG. **5**. A simple metric of moderation is the product of average logarithmic decrement ξ□ and the probability for that interaction to take place (the macroscopic cross section $\Sigma_{\square}$.), known as slowing down power **510**. Accordingly, the composite moderator medium **103** includes the high moderating material **105**A-H captured inside the low moderating material **104**A-B (e.g., a moderating matrix to entrain the high moderating material **105**A-H). The low moderating material **104**A-B is actually relatively stable under radiation. The first example low moderating material in the table is silicon carbide (SiC) **104**A, which can be chemical vapor deposition (CVD) SiC, for example.

The high moderating material **105**A-H is put inside the moderating matrix of the low moderating material **104**A-B. The moderator matrix of low moderating material **104**A-B is weakly moderating (silicon carbide or magnesium oxide), but the high moderating material **105**A-H H has large moderating capabilities. All of the high moderating materials **105**A-H are better than graphite in slowing down power **510**, and all of the low moderating materials **104**A-B are not as good as graphite moderator material **505** in slowing down power **510**. The goal is that on average the moderating ability (slowing down power **510**) of the low moderating material **104**A-B and the high moderating material **105**A-H add together to produce a composite moderator medium **103** that is more stable under radiation and lasts longer inside the nuclear reactor core **101** than the graphite moderator material **505**. In some examples of the composite moderator medium **103**, the moderating matrix of low moderating material **104**A-B is matched up with a high moderating material **105**A-H that is a beryllium (Be) or boron (B) compound.

To successfully replace the nuclear graphite moderator material **505**, the composite moderation medium **103**, has a similar moderating ability and achieves a longer moderator lifetime than the nuclear graphite moderator material **505**. Composite moderator medium **103** may be thought of as two-phase or more structures (e.g., components), fibrous structures, or alloys. Silicon carbide (SiC) as a low moderating material **104**A has been demonstrated to survive to greater than **100** displacement per atom (dpa) with a nominal "saturated" volume change, unlike the graphite moderator material **505** as shown in FIG. **4**. An example of such a structure presented here is a 45% volume fraction high moderating material **105**A-H into a SiC host moderating matrix through spark plasma sintering (SPS) processing as described in FIGS. **7** and **8**A-C. A second host moderating matrix is magnesium oxide (MgO), similarly processed. Manufacturing the engineered composite moderator medium **103** by this rapid, advanced manufacturing SPS technique suggests economic fabrication. Similar fabrication techniques could be hot pressing and sintering.

FIG. **6** is a graph **600** illustrating reactivity over time of the nuclear reactor core that includes the graphite moderator material **505** of FIG. **5** compared with seven different types of composite moderate media **103**A-G. The graph **600** demonstrates that as a function of time—years **601** shown on the X axis—some of the composite moderator media **103**A-G are better on day one and some are worse, as measured by reactivity coefficient—K_{eff} **602** shown on the Y axis. Composite moderator media **103**A-G can perform better or worse than graphite **505** depending on the selection of low moderating materials **104**A-B and high moderating materials **105**A-H. There are two versions of lifetime: (1) fuel lifetime—nuclear fuel **200** degrades and burns out and the nuclear reactor shuts down; and (2) moderator lifetime—the moderator falls apart and a nuclear regulating authority determines the nuclear reactor is too dangerous and must be turned off.

When K_{eff} **602** falls below 1, then the initial loading of nuclear fuel **200** has reached the fuel lifetime. The graph **600** does not show moderator lifetime of composite moderator media **103**A-G, rather the graph **600** shows fuel lifetime and that the composite moderator media **103**A-G are just as effective as the graphite moderator material **505**. Graph **600** also shows that moderation can be adjusted (i.e., increased or decreased) based on the selected composite moderating media **103**A-G.

Although not shown in FIG. **6**, the moderator lifetime for all of the composite moderator media **103**A-G is significantly extended over the graphite moderator material **505**, for example, to match the fuel lifetime without replacement. The problem with the graphite moderator material **505** is the extreme expense of opening up the nuclear reactor core **101** to replace the graphite moderator material, which can be around $100 million. The composite moderator media **103**A-G may be a greater upfront investment, but will save the expense needed to replace the graphite moderator material **505** in the nuclear reactor core **101** down the road. Even if more expensive on day one, the composite moderator media **103**A-G will be less expensive in the future by not requiring replacement of moderator elements, which graphite moderator material **505** requires. Typically, all but the low volume permanent outer reflector **265** (see FIG. **2**C) of a nuclear reactor core with a graphite moderator material **505** is replaced as the graphite moderator material **505** becomes compromised. Such change out is unavoidable unless the graphite moderator material **505** is replaced with a higher performance composite moderator medium **103**.

Graph **600** presents representative calculations for a representative reactor type: a 35V % SiC Matrix Fuel (FCM® nuclear fuel) in a small modular prismatic high-temperature gas-cooled reactor (HTGR). In the graph **600**, graphite moderator material **505** is implemented in a graphite moderated nuclear reactor fueled by 9% enriched UO_2 TRISO FCM® nuclear fuel. As shown, due to the larger absorption of silicon carbide (SiC), some BeSiC types of composite moderator media **103**A, **103**D implemented in prismatic HTGR nuclear reactor cores may have an unacceptably large impact on nuclear reactor core life. Meanwhile reducing the moderating matrix fraction of SiC or increasing the porosity of the composite moderating media **103**B-C, or increasing the TRISO fuel particle enrichment provides comparable lifetime to graphite moderator material **505** in the prismatic HTGR. Moreover, MgO-based composite moderator media **103**E-G are seen to have very good neutronic performance.

Graph **600** presents the neutronic impact of utilizing composite moderators in a typical small modular high temperature gas-cooled reactor (HTGR). Specifically K_{eff} **602** is a measure of the core reactivity, predicted as a function of years **601** and compared with the standard nuclear graphite moderator material **505**. A number of cases are provided with variables of type of moderating matrix of the low moderating material **104** (100% dense SiC, 100% dense MgO and 20% porous SiC) and variable uranium enrichment of the nuclear fuel **200**. In the example of FIG. **6**, the nuclear fuel **200** is comprised of fuel compacts **205** of tristructural-isotropic (TRISO) fuel particles **206**A-N embedded inside a silicon carbide matrix **207**. Also variable is the volume fraction of the moderating matrix and beryllium phase. Shown are curves for overly parasitic moderators, solutions with too much moderation and fertile fuel (too much reactivity), and a suite of options that are comparable to nuclear graphite moderator material **505**.

FIG. **7** is a flowchart of a method that can be implemented to fabricate a composite moderator block **220** of the composite moderator medium **103**. Beginning in step **700**, the method includes selecting two or more moderators including a low moderating material **104** and a high moderating material **105** to form a composite moderator medium **103**. This includes selecting powders for the low moderating material **104** and a high moderating material **105**. Generally described, the low moderating material **104** includes silicon carbide (SiC) **104**A or magnesium oxide (MgO) **104**B. The high moderating material **105** includes beryllium (Be **105**H), boron (B), or a compound thereof. More specifically, the high moderating material **105** includes at least one of beryllium boride (Be_2B **105**A, Be_4B **105**B, BeB_2, or BeB_6), beryllium carbide (Be_2C **105**C), zirconium beryllide ($ZrBe_{13}$ **105**D), titanium beryllide ($TiBe_{12}$ **105**E), beryllium oxide (BeO **105**F), or boron carbide ($^{11}B_4C$ **105**G).

Continuing to step **710**, the method further includes selecting a sintering aid and a weight percent (w/w % or wt %) or weight fraction, of the sintering aid in a composite moderator mixture based on the low moderating material **104**. This includes selecting one or more appropriate sintering aids and weight percent or fraction depending on the combination of powders for the low moderating material **104** and the high moderating material **105**. Sintering aids are eutectic powers, such as oxides (e.g., yttria and alumina) for silicon carbide, and lithium for magnesium oxide. Mass percent or mass fraction can be used instead of weight percent or weight fraction to express a mixing ratio of the composite moderator mixture.

Sintering aids vary depending on, for example, the low moderating material **104**. Sintering aids include various oxides, such as yttria (Y_2O_3) known as yttrium oxide, or alumina (Al_2O_3) known as aluminum oxide; and lithium. In a first example, in which the low moderating material **104** includes silicon carbide (SiC) **104**A, then the sintering aid includes yttria (Y_2O_3) or alumina (Al_2O_3). In this first example, the selected weight percent (w/w %) of the sintering aid in the composite moderator mixture is 3 to 10 weight percent (w/w %), and more preferably 4 to 10 w/w %, of yttria or alumina. In a second example, in which the low moderating material **104** includes magnesium oxide (MgO) **104**B, then the sintering aid includes lithium. In this second example, the selected weight percent (w/w %) of the sintering aid in the composite moderator mixture is 3 to 10 weight percent (w/w %) of lithium.

Proceeding to step **720**, the method further includes mixing the two or more moderators with the selected sintering aid at the selected weight percent (w/w %) to create the composite moderator mixture. Finishing now in step **730**, the method further includes spark plasma sintering (SPS) the composite moderator mixture to fabricate a composite moderator block **220** formed of the composite moderator medium **103**. SPS uses additives (e.g., sintering aids) to suppress sintering temperatures, which brings down processing temperature and pressure required to carry out a process run. The sintering aids reduce temperature and time to carry out the process, which advantageously minimizes vaporization loss of the high moderating material **105** (e.g., beryllium and boron compounds) of the composite moderator medium **103**.

The step of sparking plasma sintering the composite moderator mixture includes: pouring the composite moderator mixture in a mandrel; and pressing a die into the mandrel to apply a processing temperature and pressure to the composite moderator mixture to fabricate the composite moderator block **220** formed of the composite moderator medium **103**. The die is like a piston that applies the processing temperature and pressure to the composite moderator mixture. Processing temperatures vary depending on, for example, the low moderating material **104**.

Returning to the first example, in which the low moderating material **104** includes silicon carbide (SiC) **104**A and/or the sintering aid includes yttria (Y_2O_3) or alumina (Al_2O_3), then the processing temperature is in a range between 1,400° Celsius (C.) to 1,800° Celsius (C.). At the end of the SPS processing of the silicon carbide low moderating matrix **104**A, the yttria or alumina partially vaporizes. Therefore, the yttria or alumina may be partially present in the composite moderator block **220** and may be detectable in trace amounts after SPS processing in the composite moderator medium **103**.

Returning to the second example, in which the low moderating material **104** includes magnesium oxide (MgO) **104**B and/or the sintering aid includes lithium, then the processing temperature is in a range between 1,300° Celsius (C.) to 1,600° Celsius (C.). The lithium sintering aid completely vaporizes at the end of the SPS processing of the magnesium oxide low moderating material **104**B. Therefore, the lithium is not present in the composite moderator block **220** and is not typically detectable (i.e., the lithium leaves like a fugitive additive).

Spark plasma sintering (SPS), also known as field assisted sintering technique (FAST) or pulsed electric current sintering (PECS), is a sintering technique. The main characteristic of SPS is that the pulsed or unpulsed DC or AC current directly passes through the graphite die, as well as the powder compact, in case of conductive samples. Joule heating has been found to play a dominant role in the densification of powder compacts, which results in achieving near theoretical density at lower sintering temperature compared to conventional sintering techniques. The heat generation is internal, in contrast to the conventional hot pressing, where the heat is provided by external heating elements. This facilitates a very high heating or cooling rate (up to 1,000 Kelvin per minute), hence the sintering process is very fast (within a few minutes). The general speed of the SPS process ensures it has the potential of densifying powders with nanosize or nanostructure, while avoiding coarsening which accompanies standard densification routes. SPS is a good method for preparation of ceramics based on nanoparticles with enhanced magnetic, magnetoelectric, piezoelectric, thermoelectric, optical, or biomedical properties.

FIG. **8**A is a processing photograph **800** of the method of FIG. **7** in which spark plasma sintering (SPS) is utilized to fabricate the composite moderator block **220**. The SPS processing photograph **800** shows the SPS procedure looking through a direct current sintering (DCS) window at spark plasma sintering (SPS) materials to produce the composite moderator block **220**. In this case, a graphite punch (glowing) within a graphite die containing the powders is shown through the DCS window of FIG. **8**A. SPS is an advanced manufacturing technique that enables fabrication at a much lower processing temperature to rapidly produce the composite moderator block **220**. The low moderating material **104** powder and high moderating material **105** powder are mixed up and sintering aids are added. SPS allows the powders to be heated up rapidly. In order to encapsulate the high moderating material **105**, the low moderating material **104** is solidified before the high moderating material **105** (e.g., beryllium compounds) evaporates during SPS to form the composite moderator block **220** of the composite moderator medium **103**.

Composite moderator medium **103** processing can be carried out using high-vacuum direct-current sintering (Sinterland LABOX 3010KF) of relatively pure SiC powders. Current powders being consolidated to high-density include nano-SiC powders in the 35-100 nanometer (nm) range and Acheson-derived SiC powders in the 0.2-2 micrometer (μm) range. All materials are kinetically stable, ensuring thorough, impurity free dispersion, cold pressed sintered in the spark plasma sintering apparatus.

FIG. **8**B is a graph **810** illustrating a processing temperature **815** and pressure (die displacement) **820** over time **825** during sparking plasma sintering (SPS) of the method of FIG. **7**. The graph **810** shows the temperature **815**, pressure **820** (e.g., die displacement), and time **825** to do a process run with SPS to fabricate the composite moderator block **220**. The graph **810** provides a relative time-temperature trace for the SPS procedure, indicating processing temperatures in the 1,500° C. range for SiC as the low moderating material **104**A. Processing temperatures for the MgO as the low moderating material **104**B can be in the range of 1,300-1,600° C. Run of the mill processing of a low moderating material **104**, such as silicon carbide (SiC), would take several hours for a furnace to attain the needed temperature of well above 2,000° C. to fabricate the composite moderator block **220**. With SPS processing, ten minutes at a processing temperature from 1,600°-1,800° C. enables fabrication of the composite moderator block **220** of the composite moderator medium **103**.

In order to minimize any hygroscopic tendencies of the sintering additives, zirconium or zirconia may be added. Processing temperatures for SiC as the low moderating material **104**A can be in the range of 1,400-1,800° C. with sintering additives of alumina or yttria added to enhance densification.

FIG. **8**C is a micron level photograph **830** of a polished cross-section of the composite moderator medium **103** showing the low moderating material **104**A (SiC) encapsulating the high moderating material **105**C (Be_2C). The micron level photograph **830** is a microscopic view of the composite moderator medium **103** of the composite moderator block **220** that is produced and shows the low moderating material **104**A crystal microstructure near the interface of the high moderating material **105**C. The silicon carbide moderating matrix of the low moderating material **104**A completely encapsulates (covers) the porous carbon coating of the beryllium material of the high moderating material **105**C (Be_2C). Since beryllium is toxic, encapsulation around the beryllium by the non-toxic silicon carbide low moderating material **104**A is advantageous because exposure to the toxic high moderating material **105**C is eliminated. The porous carbon interlayer is coated on the beryllium of the high moderating material **105**C and thus is located in between silicon carbide low moderating material **104**A. The silicon carbide moderating matrix (SiC) of the low moderating material **104**A is completely densified around high moderating material **105**C.

The micron level photograph **830** of FIG. **8**C presents an image of a two-phase SiC matrix composite (moderator second phase volume fraction approximately 35%). As depicted in FIG. **8**B, sintering in excess of 1500° C. was applied with a hold time of approximately 10 minutes, achieving a near full-density SiC moderating matrix density for a relatively small (8 mm diameter) part.

Various composite moderators have been disclosed for use in nuclear reactors, including advanced nuclear fission reactor applications. The composite moderator is, for example, a high moderating material **105** (e.g., beryllium containing phase) contained in a low moderating material **104** (e.g., continuum or a radiation-stable matrix phase of SiC or MgO). Similar neutronic moderation to nuclear graphite moderator material **505** can be provided through the use of the high moderating material **105**, while providing a number of safety, economic, and waste reduction benefits conveyed by the use of the low moderating material **104**. Thus, the composite moderators can replace nuclear graphite moderator material **505** and have superior moderator lifetime and increased safety and waste disposal attributes. Example fabrication processing includes the use of eutectic powder during spark plasma sintering (SPS) of the low moderating material **104** (e.g., radiation-stable matrices of silicon carbide and magnesium oxide) and the high moderating material **105**.

As described above, a method is disclosed for fabricating a composite moderator (e.g., composite moderator block **220**) formed of a composite moderator medium **103** for a nuclear reactor core **101**. The method includes producing a composite moderator medium **103** (two-phase composite moderator) that includes a high moderating material **105** (e.g., second captured phase) within a continuum of a low moderating material **104** (e.g., first matrix phase). The low moderating material **104** is either SiC or MgO. The high moderating material **105** is either a beryllium containing compound, such as Be2C, BeO, or beryllium metal with a shell of Be2C or BeO. The interphase between the low moderating material **104** and the high moderating material **105** is a porous compliant structure capable of absorbing helium produced through n-alpha reactions. The composite moderator medium **103** is a lifetime component of the nuclear reactor core **101**.

It will be understood that the terms and expressions used herein have the ordinary meaning as is accorded to such terms and expressions with respect to their corresponding respective areas of inquiry and study except where specific meanings have otherwise been set forth herein. Relational terms such as first and second and the like may be used solely to distinguish one entity or action from another without necessarily requiring or implying any actual such relationship or order between such entities or actions. The terms "comprises," "comprising," "includes," "including," or any other variation thereof, are intended to cover a non-exclusive inclusion, such that a process, method, article, or apparatus that comprises or includes a list of elements or steps does not include only those elements or steps but may include other elements or steps not expressly listed or inherent to such process, method, article, or apparatus. An element preceded by "a" or "an" does not, without further constraints, preclude the existence of additional identical elements in the process, method, article, or apparatus that comprises the element.

Unless otherwise stated, any and all measurements, values, ratings, positions, magnitudes, sizes, and other specifications that are set forth in this specification, including in the claims that follow, are approximate, not exact. Such amounts are intended to have a reasonable range that is consistent with the functions to which they relate and with what is customary in the art to which they pertain. For example, unless expressly stated otherwise, a parameter value or the like may vary by as much as ±10% from the stated amount.

In addition, in the foregoing Detailed Description, it can be seen that various features are grouped together in various examples for the purpose of streamlining the disclosure. This method of disclosure is not to be interpreted as reflecting an intention that the claimed examples require more features than are expressly recited in each claim. Rather, as the following claims reflect, the subject matter to be protected lies in less than all features of any single disclosed example. Thus the following claims are hereby incorporated into the Detailed Description, with each claim standing on its own as a separately claimed subject matter.

While the foregoing has described what are considered to be the best mode and/or other examples, it is understood that various modifications may be made therein and that the subject matter disclosed herein may be implemented in various forms and examples, and that they may be applied in numerous applications, only some of which have been described herein. It is intended by the following claims to claim any and all modifications and variations that fall within the true scope of the present concepts.

What is claimed is:

1. A nuclear reactor system comprising:

a nuclear reactor core including:

an array of fuel elements; and

a composite moderator medium formed of two or more moderators;

wherein:

the two or more moderators include a low moderating material and a high moderating material;

the high moderating material has a higher neutron slowing down power compared to the low moderating material;

the low moderating material includes a moderating matrix of silicon carbide (SiC) or magnesium oxide (MgO); and

the high moderating material is dispersed within the moderating matrix and includes beryllium (Be), boron (B), or a compound thereof.

2. The nuclear reactor system of claim **1**, wherein the high moderating material includes at least one of beryllium boride (Be_2B, Be_4B, BeB_2, or BeB_6), beryllium carbide (Be_2C), zirconium beryllide ($ZrBe_{13}$), titanium beryllide ($TiBe_{12}$), beryllium oxide (BeO), or boron carbide (B_4C).

3. The nuclear reactor system of claim **1**, wherein the high moderating material is adjacent the low moderating material.

4. The nuclear reactor system of claim **1**, wherein:

each of the fuel elements includes a fuel composite moderator block formed of the composite moderator medium and nuclear fuel;

the fuel composite moderator block includes fuel openings; and

the nuclear fuel is disposed inside the fuel openings, such that the nuclear fuel is enclosed by the composite moderator medium.

5. The nuclear reactor system of claim **4**, wherein:

the fuel composite moderator block further includes a coolant passage to flow a coolant gas or liquid.

6. The nuclear reactor system of claim **4**, wherein:

the nuclear fuel includes a fuel compact comprised of:

tristructural-isotropic (TRISO) fuel particles embedded inside a silicon carbide matrix; or

tristructural-isotropic (TRISO) fuel particles embedded inside a graphite matrix.

7. The nuclear reactor system of claim **1**, wherein the nuclear reactor core further includes at least one reflector region that includes reflector blocks formed of the composite moderator medium.

8. The nuclear reactor system of claim **4**, wherein:

the nuclear reactor core includes an inner reflector region and an outer reflector region;

the inner reflector region includes inner reflector blocks;

the outer reflector region includes outer reflector blocks; and

the inner reflector blocks and the outer reflector blocks are formed of the composite moderator medium.

9. The nuclear reactor system of claim **8**, wherein:

the array of fuel elements surrounds the inner reflector region; and

the outer reflector region surrounds the array of fuel elements.

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