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(54) **Title:** INTEGRATED IN-VESSEL NEUTRON SHIELD

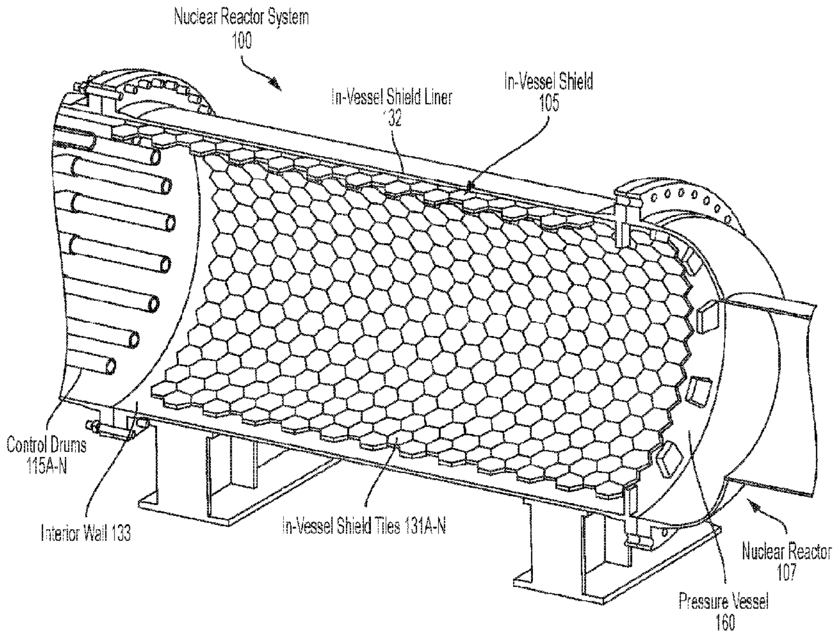

FIG. 1A

(57) **Abstract:** To reduce size and mass of a nuclear reactor system, an integrated in-vessel separates the role of a neutron reflector and a neutron shield. Nuclear reactor system includes a pressure vessel including an interior wall and a nuclear reactor core located within the interior wall of the pressure vessel. Nuclear reactor core includes a plurality of fuel elements and at least one moderator element. Nuclear reactor system includes a reflector located inside the pressure vessel that includes a plurality of reflector blocks laterally surrounding the plurality of fuel elements and the at least one moderator element. Nuclear reactor system includes the in-vessel shield located on the interior wall of the pressure vessel to surround the reflector blocks. In-vessel shield is formed of two or more neutron absorbing materials. The two more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material.

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Declarations under Rule 4.17:
— *as to applicant's entitlement to apply for and be granted a patent (Rule 4.17(ii))*
— *as to the applicant's entitlement to claim the priority of the earlier application (Rule 4.17(iii))*

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INTEGRATED IN-VESSEL NEUTRON SHIELD

Cross-Reference to Related Applications

[0001] This application claims priority to U.S. Provisional Patent Application No. 62/910,561, filed on October 4, 2019, titled “Nuclear System for Power Production in Space,” the entirety of which is incorporated by reference herein.

[0002] This application relates to International Application No. PCT/US2020/XXXXXX, filed on October 4, 2020, titled “Nuclear Reactor Core Architecture with Enhanced Heat Transfer and Safety,” the entirety of which is incorporated by reference herein. This application also relates to International Application No. PCT/US2020/XXXXXX, filed on October 4, 2020, titled “Automatic Shutdown Controller for Nuclear Reactor System with Control Drums,” the entirety of which is incorporated by reference herein.

Technical Field

[0003] The present subject matter relates to examples of nuclear reactor systems and nuclear reactors for power production and propulsion, e.g., in remote regions, such as outer space. The present subject matter also encompasses a nuclear reactor that includes an in-vessel neutron shield and a method for fabricating the in-vessel neutron shield.

Background

[0004] Nuclear fission reactors include thermal or fast type reactors. Currently, almost all operating nuclear fission reactors are thermal. Nuclear fission reactors include nuclear fuel inside a nuclear reactor core and a moderator to slow down fast neutrons so that nuclear fission can continue. Typically, the nuclear fuel is formed in cylindrical shaped fuel compacts or pellets. The fuel compacts are loaded into fuel pins or rods, cladded, and stacked inside the numerous columns of fuel elements in the nuclear reactor core.

[0005] The nuclear reactor core increases neutron fluence or neutron flux, which is the distance free neutrons move in a volume per time. Neutron fluence is necessary in nuclear reactors to generate heat and energy, but neutron fluence, particularly fast neutron fluence, or the neutron fluence of neutrons moving at a speed of 14,000km/s or higher, is very dangerous to both matter and human life. Therefore, the high fast neutron fluence is decreased to safe levels at distances further from the nuclear reactor core, ideally to safe values just beyond the nuclear reactor itself. Some components of a nuclear reactor, such as neutron reflectors, perform this action: reflectors direct free neutrons back toward the nuclear reactor core, thereby increasing neutron fluence inside the nuclear reactor, improving energy efficiency, and decreasing neutron fluence outside the nuclear reactor, making the surrounding area safer.

[0006] When building a conventional nuclear reactor system for conventional terrestrial land applications, e.g., a nuclear power plant, the size (e.g., space or volume) and mass of the nuclear reactor is not a major concern. Typically, the actual thickness of the neutron reflector to reflect neutrons back to the nuclear reactor core for energy efficiency is much greater than the required thickness needed to appropriately reduce fast neutron fluence to safe levels beyond the nuclear reactor. Therefore, the conventional nuclear reactor tends to have a very thick neutron reflector in order to reduce neutron fluence external to the nuclear reactor. Typically, the neutron reflector is much thicker than required for optimal neutronic performance needed to reflect the free neutrons back to the nuclear reactor core. Essentially, the neutron reflector additionally acts as a neutron shield.

[0007] However, the size and mass of a nuclear reactor system are important factors for nuclear thermal propulsion (NTP) and providing nuclear power (e.g., thermal and/or electrical power) in remote region applications including outer space, celestial bodies, planetary bodies, and remotes regions on Earth. For example, the mass of the nuclear reactor system directly affects performance, such as power per mass, in the NTP and space applications; and may add unnecessary manufacturing cost. Moreover, in the remote region applications, a smaller (more compact) form factor reduces construction costs and increases transportability while allowing operation of the nuclear reactor at high power densities. Accordingly, improvements to the conventional implementation of a neutron reflector as a space-inefficient neutron shield are needed.

Summary

[0008] The various examples disclosed herein relate to nuclear technologies for nuclear reactor systems both for space or terrestrial land applications. The nuclear reactor system 100 advantageously reduces the size and mass of the nuclear reactor 107 by implementing integrated in-vessel shield 105 to separate the role of neutron reflector and the neutron shield and still obtain optimal neutronic performance. The integrated in-vessel shield 105 brings radiation shielding into the pressure vessel 160 of the nuclear reactor 107, which allows for reduction of mass of the nuclear reactor system 100 and achieves a space and mass efficient neutron shield.

[0009] Integrated in-vessel shield 105 has the following benefits. First, in-vessel shield 105 reduces the required distance between the pressure vessel 160 and the active nuclear reactor core 101 and thus the total size of the pressure vessel 160. For example, in-vessel shield 105 enables a thinner reflector 140 and smaller overall pressure vessel 160. Prior neutron reflectors unnecessarily increase diameter of the pressure vessel 160, leading to a larger diameter of the pressure vessel 160 that is more expensive to manufacture and difficult to transport and field. Second, in-vessel shield 105 extends the life of the pressure vessel 160 by reducing fast flux to the pressure vessel 160. Third, in-vessel shield 105 reduces activation outside of the nuclear reactor core 101 and therefore makes the nuclear reactor 107 easier to install. Fourth, in-vessel shield 105 can be incorporated (e.g., retrofitted) into an already built nuclear reactor 107 or the design of a new nuclear reactor 107.

[0010] To achieve breakthrough performance, in-vessel shield 105 includes a near black neutron absorbing material and a gray neutron absorbing material to survive the radiation environment and thermal environment that is found inside of the pressure vessel 160 of the nuclear reactor 107. For fabrication of the in-vessel shield 105, advanced 3D printing and spark plasma sintering methods can be employed. In-vessel shield 105 is shaped as a plurality of in-vessel shield tiles 131A-N that shape the near black neutron absorbing material and the gray neutron absorbing material into an interlocking geometry pattern well-suited for placement on the interior wall 133 inside of the pressure vessel 160 and removes neutron streaming paths. In-vessel shield 105 solves the problem of limiting fast fluence to the pressure vessel 160 in a volume-constrained nuclear reactor 107 and decreases fast flux to the pressure vessel 160 with the outside. In-vessel shield 105 reduces the radiation field around the nuclear reactor 107, which in turn decreases the activation in the area around the nuclear reactor 107 and facilitates installation.

[0011] An example nuclear reactor system 100 includes a pressure vessel 160 including an interior wall 133 and a nuclear reactor core 101 located within the interior wall 133 of the pressure vessel 160. Nuclear reactor core 101 includes a plurality of fuel elements 104A-N and at least one moderator element 103. Nuclear reactor system 100 includes a reflector 140 located inside the pressure vessel that includes a plurality of reflector blocks 141A-N laterally surrounding the plurality of fuel elements 104A-N and the at least one moderator element 103. Nuclear reactor system 100 includes an in-vessel shield

105 located on the interior wall 133 of the pressure vessel 160 to surround the reflector blocks 141A-N. In-vessel shield 105 is formed of two or more neutron absorbing materials. The two more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material.

[0012] An example method includes selecting two or more neutron absorbing materials to form an in-vessel shield 105 (block 610). The two or more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material. The method further includes eutectic sintering the near black neutron absorbing material to fabricate a ceramic absorbing powder (block 615). The method further include kinetically mixing the gray neutron absorbing material and the ceramic absorbing powder to create an in-vessel shield mixture (block 620). The method further includes cold press sintering the in-vessel shield mixture into an in-vessel shield 105 (block 625).

[0013] 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

[0014] The drawing figures depict one or more implementations, by way of example only, not by way of limitations. In the figures, like reference numerals refer to the same or similar elements.

[0015] FIG. 1A is an isometric view showing a cutaway of a pressure vessel that includes an in-vessel shield to enhance performance of a nuclear reactor system.

[0016] FIG. 1B is a cross-sectional view of a nuclear reactor core of the nuclear reactor system.

[0017] FIG. 1C illustrates a nuclear reactor system that implements a pressure vessel with an in-vessel shield liner on an interior wall and including a plurality of control drums.

[0018] FIG. 1D illustrates a nuclear reactor system that implements the pressure vessel with the in-vessel shield liner on the interior wall and including a plurality of control rods.

[0019] FIG. 2 is an overall shielding thickness graph for a reflector and in-vessel shield in eleven different implementations of the nuclear reactor.

[0020] FIG. 3 is an example depletion graph of two different types of nuclear reactor cores.

[0021] FIG. 4 is an in-vessel shield material optimization chart for the nuclear reactor system.

[0022] FIG. 5 is a flowchart depicting an in-vessel shield material selection method.

[0023] FIG. 6 is a flowchart depicting an in-vessel shield method that includes in-vessel shield manufacture, as well techniques for in-vessel shield design and installation.

[0024] **Parts Listing**

100	Nuclear Reactor System
101	Nuclear Reactor Core
101A-B	Nuclear Reactor Cores
102A-N	Insulator Elements
103A-N	Moderator Elements
104A-N	Fuel Elements
105	Reflector
105	In-Vessel Shield
107	Nuclear Reactor
107A-K	Nuclear Reactor Cores
112	Insulator Element Array
113	Moderator Element Array
114	Nuclear Fuel Tile Array
115A-N	Control Drums
116	Reflector Material
117	Absorber Material
118A-N	Control Rods
131A-N	In-Vessel Shield Tiles
132	In-Vessel Shield Liner
133	Interior Wall
140	Reflector
141A-N	Reflector Blocks
160	Pressure Vessel
200	Overall Shielding Thickness Graph
205A-K	Percentage of Overall Shielding as In-Vessel Shield
206A-K	Total Thickness
210A-K	In-vessel Shield Thickness
211A-K	Reflector Thickness
300	Depletion Graph of Nuclear Reactor Cores
301A-B	Nuclear Reactor Core Architectures
305	Lifetime
305A-B	Lifetimes
310	K-Effective

400	In-Vessel Shield Material Optimization Chart
401	Short Lifetime
402	Medium Lifetime
403	Long Lifetime
405	Cost
406	Reduction in Pressure Vessel Fast Neutron Fluence
411	Short Pareto Front
412	Medium Pareto Front
413	Long Pareto Front
500	In-Vessel Shield Material Selection Method
600	In-Vessel Shield Method
601	In-Vessel Shield Manufacture
602	In-Vessel Shield Design and Installation

Detailed Description

[0025] 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.

[0026] The term “coupled” as used herein refers to any logical or physical connection. Unless described otherwise, coupled elements or devices are not necessarily directly connected to one another and may be separated by intermediate components, elements, etc.

[0027] Unless otherwise stated, any and all measurements, values, ratings, positions, magnitudes, sizes, angles, 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 ± 5% or as much as ± 10% from the stated amount. The term “approximately” or “substantially” means that the parameter value or the like varies up to ± 10% from the stated amount.

[0028] The orientations of the nuclear reactor core 101, nuclear reactor 107, associated components, and/or any nuclear reactor system 100 incorporating the in-vessel shield 105, such as shown in any of the drawings, are given by way of example only, for illustration and discussion purposes. In operation for a particular nuclear reactor system

100, the nuclear reactor 107 may be oriented in any other direction suitable to the particular application of the nuclear reactor 107, for example upright, sideways, or any other orientation. Also, to the extent used herein, any directional term, such as lateral, longitudinal, up, down, upper, lower, top, bottom, and side, are used by way of example only, and are not limiting as to direction or orientation of any nuclear reactor 107 or component of the nuclear reactor 107 constructed as otherwise described herein. Reference now is made in detail to the examples illustrated in the accompanying drawings and discussed below.

[0029] FIG. 1A is an isometric view showing a cutaway of a pressure vessel 160 that includes an in-vessel shield 105 to enhance performance of a nuclear reactor system 100. As shown in FIG. 1A a nuclear reactor system 100 includes a nuclear reactor 107. Nuclear reactor 107 includes a pressure vessel 160 that includes an interior wall 133.

[0030] FIG. 1B is a cross-sectional view of a nuclear reactor core 101 of the nuclear reactor system 100. Nuclear reactor system 100 includes the nuclear reactor core 101, and the nuclear reactor core 101 is located within the interior wall 133 of the pressure vessel 160. Generally, the nuclear reactor core 101 includes a plurality of fuel elements 104A-N and at least one moderator element 103. In the implementation of FIG. 1B, the plurality of fuel elements 104A-N are arranged as a nuclear fuel tile array 114 of nuclear fuel tiles 104A-N and the nuclear reactor core 101 includes a plurality of moderator elements 103A-N. In a second example, the nuclear reactor core 101 can be implemented like the nuclear reactor core 110 described in FIGS. 3-4 and the associated text of U.S. Patent No. 10,643,754 to Ultra Safe Nuclear Corporation of Seattle, Washington, issued May 5, 2020, titled “Passive Reactivity Control of Nuclear Thermal Propulsion Reactors” the entirety of which is incorporated by reference herein. In the second example, the fuel elements 104A-N can be implemented like the fuel elements 310A-N and the moderator elements 103A-N can be implemented like the tie tubes 320A-N described in FIGS. 3-4 and the associated text of U.S. Patent No. 10,643,754.

[0031] In a third example, the nuclear reactor core 101 can be implemented like the nuclear reactor core 101 described in FIG. 2C and the associated text of U.S. Patent Pub. No. 2020/0027587 to Ultra Safe Nuclear Corporation of Seattle, Washington, published Jan. 23, 2020, titled “Composite Moderator for Nuclear Reactor Systems,” the entirety of which is incorporated by reference herein. In the third example, the fuel elements 104A-N can be implemented like the fuel elements 102A-N and the moderator elements 103A-N can be implemented like the composite moderator blocks described in FIG. 2C and the associated text of U.S. Patent Pub. No. 2020/0027587.

[0032] As further shown in FIG. 1B, nuclear reactor 107 includes a reflector 140 (e.g., an outer reflector region) located inside the pressure vessel 160. Reflector 140 includes a plurality of reflector blocks 141A-N laterally surrounding the plurality of fuel elements 104A-N and the at least one moderator element 103. As shown in FIG. 1A, nuclear reactor 107 includes an in-vessel shield 105 located on the interior wall 133 of the pressure vessel 160 to surround the reflector blocks 141A-N.

[0033] In-vessel shield 105 is formed of two or more neutron absorbing materials. The two more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material. Near black neutron absorbing material includes a composited ceramic material. Gray neutron absorbing material includes a heavy metal material. The composited ceramic material and the heavy metal material form the in-vessel shield. Composited ceramic material includes boron carbide (B_4C), hafnium carbide (HfC), or gadolinium oxide (Gd_2O_3). Composited ceramic material can further include aluminum oxide (Al_2O_3) or silicon carbide (SiC). Heavy metal material includes tungsten (W), iron (Fe), Nickel (Ni), or copper (Cu).

[0034] More specifically, composited ceramic material can include: boron-10 carbide ($^{10}B_4C$), a boron-10 carbide and aluminum oxide composite ($^{10}B_4C$-Al_2O_3) of 50% boron-10 carbide by weight percent, a boron-10 carbide and silicon carbide composite ($^{10}B_4C$-SiC) of 50% boron-10 carbide by weight percent, or a borated stainless steel alloy of 5% boron-10 by weight percent. The heavy metal material includes atomized tungsten heavy metal with a tungsten content greater than or equal to 90% by weight percent. Composited ceramic material can include a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns. The plurality of composited ceramic particles can be embedded inside a heavy metal matrix of the heavy metal material to form the in-vessel shield 105.

[0035] In-vessel shield 105 can be formed integrally (e.g., as one component, part, or piece) as an in-vessel shield liner 132 that is a suitable shape to line the interior wall 133, e.g., a tube or pipe shape. Alternatively, the in-vessel shield 105 is formed as several components formed separately and then connected together. Hence, as depicted in FIG. 1B, in-vessel shield 107 is formed as a plurality of in-vessel shield tiles 131A-N, that are joined together to form the in-vessel shield liner 132. The plurality of in-vessel shield tiles 131A-N are disposed on (e.g., attached to) the interior wall 133 to form the in-vessel shield 105.

[0036] In-vessel shield tiles 131A-N are tolerant to swelling and reduce neutron streaming paths. In-vessel shield tiles 131A-N have a composition and geometry that can be designed particularly for the nuclear reactor system 100, for example, the plurality of in-vessel shield tiles 131A-N include a base shape with an interlocking geometry pattern. In FIG. 1A, the in-vessel shield 105 is made up of the in-vessel shield tiles 131A-N with the base shape. All or a subset of the plurality of in-vessel shield tiles 131A-N can be a curved polyhedron shape or a truncated portion thereof to match the curvature of the interior wall 133. In-vessel shield tiles 131A-N are also shaped to conform to match contour(s) (e.g., curvature) of the surface(s) of the interior wall 133. In FIG. 1B, the in-vessel shield tiles 131A-N are shaped as a curved hexagonal prism in three-dimensional space or a curved hexagon in two-dimensional space. In-vessel shield tiles 131A-N can be another polyhedron shape (e.g., a triangular prism or a cuboid) in three-dimensional space or circular, oval, square, rectangular, triangular, or another polygon shape in two-dimensional space.

[0037] Interior wall 133 can be formed of a continuous surface, e.g., rounded, aspherical, or spherical surfaces to form a cylinder or other conical surfaces to form a quadric surface, such as a hyperboloid, cone, ellipsoid, paraboloid, etc. Alternatively or additionally, the interior wall 133 can be formed of a plurality of discontinuous surfaces (e.g., to form a cuboid or other polyhedron). As used herein, “discontinuous” means that the surfaces in aggregate do not form a continuous round (e.g., circular or oval) perimeter of the interior wall 133. In FIG. 1A, the portion of the interior wall 133 shown is a rounded continuous surface. The interlocking geometry pattern of the plurality of in-vessel shield tiles 131A-N are joined to cover and collectively form an in-vessel shield liner 132 on the continuous or discontinuous surfaces of the interior wall 133.

[0038] Returning now to FIG. 1B, nuclear reactor 107 includes the nuclear reactor core 101, in which a controlled nuclear chain reactions occurs, and energy is released. 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.

By sustaining controlled nuclear fission, the nuclear reactor system 100 produces heat energy. In an example implementation, the nuclear reactor system 100 is implemented as a gas-cooled high temperature nuclear reactor 107. However, the nuclear reactor system 100 with the in-vessel shield 105 can enable breakthrough performance by reducing mass and size and improving power per mass in a large utility scale nuclear reactor, heat pipe nuclear reactor, molten-salt-cooled nuclear reactor, fuel-in-salt nuclear reactor, or a sodium-cooled fast nuclear reactor. For example, in-vessel shield 105 can be included in a nuclear reactor system 100, such as a gas-cooled graphite-moderated nuclear reactor, a fluoride salt-cooled high-temperature nuclear reactor with a higher thermal neutron flux than the gas-cooled graphite-moderated nuclear reactor, or a sodium fast nuclear reactor with a faster neutron flux than the gas-cooled graphite-moderated nuclear reactor.

[0039] In the depicted example, the nuclear reactor system 100 with the nuclear reactor core 101 is utilized in a space environment, such as in a nuclear thermal propulsion (NTP) system. An example NTP system that the integrated in-vessel shield 105 can be implemented in is described in FIGS. 1-2 and the associated text of U.S. Patent No. 10,643,754 to Ultra Safe Nuclear Corporation of Seattle, Washington, issued May 5, 2020, titled "Passive Reactivity Control of Nuclear Thermal Propulsion Reactors" the entirety of which is incorporated by reference herein. For example, the nuclear reactor system 100 that includes the in-vessel shield 105 can be a nuclear thermal rocket reactor, nuclear electric propulsion reactor, Martian surface reactor, or lunar surface reactor.

[0040] In such an NTP system (e.g., compact space nuclear reactor), 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. Typically, this is done by heating a propellant, typically low molecular weight hydrogen, to over 2,600° Kelvin by harnessing thermal energy from the nuclear reactor core 101. In addition, the NTP nuclear reactor system 100 can be used in the propulsion of submarines or ships.

[0041] As noted above, the nuclear reactor system 100 can also be a nuclear power plant in a terrestrial land application, e.g., for providing nuclear power (e.g., thermal and/or electrical power) for remote region applications, including outer space, celestial bodies, planetary bodies, and remotes regions on Earth. An example terrestrial land nuclear reactor system that the in-vessel shield 105 can be implemented in is described in FIG. 1 and the associated text of U.S. Patent Pub. No. 2020/0027587 to Ultra Safe Nuclear Corporation of Seattle, Washington, published Jan. 23, 2020, titled “Composite Moderator for Nuclear Reactor Systems,” the entirety of which is incorporated by reference herein. For example, the nuclear reactor system 100 can be a small commercial fission power system for near term space operations, lunar landers, or a commercial fission power system for high-power spacecraft and large-scale surface operations, such as in-situ resource utilization. In another example, the nuclear reactor system 100 with the in-vessel shield 105 is utilized in a space reactor for electrical power production on a planetary surface.

[0042] Nuclear reactor system 100 can also be a terrestrial power system, such as a nuclear electric propulsion (NEP) system for fission surface power (FSP) system. NEP powers electric thrusters such as a Hall-effect thruster for robotic and human spacecraft. FSP provides power for planetary bodies such as the moon and Mars. In the NEP and FSP power applications, the nuclear reactor system 100 heats a working fluid (e.g., He, HeXe, Ne, CO_2) through a power conversion system (e.g., Brayton) to produce electricity. Moreover, in the NEP and FSP power applications, the nuclear reactor system 100 does not include a propellant, but rather includes a working fluid that passes through a reactor inlet when producing power. In the NEP and FSP power applications, the moderator elements 103A-N can be cooled via the reactor inlet working fluid (e.g., the flow coming out of a recuperator) before the working fluid passes through the fuel elements 104A-N.

[0043] Each of the fuel elements 104A-N, shown as nuclear fuel tiles 104A-N, includes a nuclear fuel. The nuclear fuel includes a fuel compact comprised of coated fuel particles, such as tristructural-isotropic (TRISO) fuel particles embedded inside a high-temperature matrix. In some implementations, the nuclear fuel includes a fuel compact comprised of bistructural-isotropic (BISO) fuel particles embedded inside the high-temperature matrix. The high-temperature matrix includes silicon carbide, zirconium carbide, titanium carbide, niobium carbide, tungsten, molybdenum, or a combination thereof. Each of the TRISO fuel particles can include a fuel kernel surrounded by a porous carbon buffer layer, an inner pyrolytic carbon layer, a binary carbide layer (e.g., ceramic layer of SiC or a refractory metal carbide layer), and an outer pyrolytic carbon layer. The refractory metal carbide layer of the TRISO fuel particles can include at least one of titanium carbide (TiC), zirconium carbide (ZrC), niobium carbide (NbC), tantalum carbide, hafnium carbide, ZrC-ZrB_2 composite, ZrC-ZrB_2-SiC composite, or a combination thereof.

The high-temperature matrix can be formed of the same material as the binary carbide layer of the TRISO fuel particles.

[0044] A description of TRISO fuel particles dispersed in a silicon carbide matrix to form a cylindrical shaped nuclear fuel compact is provided in the following patents and publications of Ultra Safe Nuclear Corporation of Seattle, Washington: U.S. Patent No. 9,299,464, issued March 29, 2016, titled "Fully Ceramic Nuclear fuel and Related Methods"; U.S. Patent No. 10,032,528, issued July 24, 2018, titled "Fully Ceramic Micro-encapsulated (FCM) fuel for CANDUs and Other Reactors"; U.S. Patent No. 10,109,378, issued Oct. 23, 2018, titled "Method for Fabrication of Fully Ceramic Microencapsulation Nuclear Fuel"; U.S. Patent Nos. US 9,620,248, issued April 11, 2017 and 10,475,543, issued Nov. 12, 2019, titled "Dispersion Ceramic Micro-encapsulated (DCM) Nuclear Fuel and Related Methods"; U.S. Patent Pub. No. 2020/0027587, published Jan. 23, 2020, titled "Composite Moderator for Nuclear Reactor Systems"; and U.S. Patent No. 10,573,416, issued Feb. 25, 2020, titled "Nuclear Fuel Particle Having a Pressure Vessel Comprising Layers of Pyrolytic Graphite and Silicon Carbide," the entireties of which are incorporated by reference herein. As described in those Ultra Safe Nuclear Corporation patents, the nuclear fuel can include a cylindrical fuel compact or pellet comprised of TRISO fuel particles embedded inside a silicon carbide matrix to create a cylindrical shaped nuclear fuel compact.

[0045] As shown, nuclear reactor core 101 includes an insulator element array 112 of insulator elements 102A-N and a moderator element array 113 of moderator elements 103A-N. Insulator elements 102A-N are formed of a high-temperature thermal insulator material with low thermal conductivity. The high-temperature thermal insulator material can include low density carbides, metal-carbides, metal-oxides, or a combination thereof. More specifically, the high-temperature thermal insulator material includes low density SiC, stabilized zirconium oxide, aluminum oxide, low density ZrC, low density carbon, or a combination thereof. Moderator elements 103A-N are formed of a low-temperature solid-phase moderator. The low-temperature solid-phase moderator includes MgH_x, YH_x, ZrH_x, CaH_x, ZrO_x, CaO_x, BeO_x, BeC_x, Be, enriched boron carbide, $^{11}B_4C$, CeH_x, LiH_x, or a combination thereof.

[0046] In an NTP, NEP, or FSP nuclear reactor system 100, the nuclear reactor 107 can include a plurality of control drums 115A-N and a reflector 140. The control drums

115A-N may laterally surround the insulator element array 112 of insulator elements 102A-N, the moderator element array 113 of moderator elements 103A-N, and nuclear fuel tile array 114 of nuclear fuel tiles 104A-N to change reactivity of the nuclear reactor core 101 by rotating the control drums 115A-N. As depicted, the control drums 115A-N reside on the perimeter or periphery of a pressure vessel 160 and are positioned circumferentially around the insulator elements 102A-N, moderator elements 103A-N, and nuclear fuel tiles 104A-N of the nuclear reactor core 101. Control drums 115A-N may be located in an area of the reflector 140, e.g., an outer reflector region formed of reflector blocks 141A-N immediately surrounding the nuclear reactor core 101, to selectively regulate the neutron population and nuclear reactor power level during operation. For example, the control drums 115A-N can be a cylindrical shape and formed of both a reflector material 116 (e.g., beryllium (Be), beryllium oxide (BeO), BeSiC, BeMgO, Al_2O_3, etc.) on a first outer surface and an absorber material 117 on a second outer surface.

[0047] The reflector material 116 and the absorber material 117 can be on opposing sides of the cylindrical shape, e.g., portions of an outer circumference, of the control drums 115A-N. The reflector material 116 can include a reflector substrate shaped as a cylinder or a truncated portion thereof. The absorber material 117 can include an absorber plate or an absorber coating. The absorber plate or the absorber coating are disposed on the reflector substrate to form the cylindrical shape of each of the control drums 115A-N. For example, the absorber plate or the absorber coating covers the reflector substrate formed of the reflector material to form the control drums 115A-N.

[0048] Rotating the depicted cylindrical-shaped control drums 115A-N changes proximity of the absorber material 117 (e.g., boron carbide, B_4C) of the control drums 115A-N to the nuclear reactor core 101 to alter the amount of neutron reflection. When the reflector material 116 is inwards facing towards the nuclear reactor core 101 and the absorber material 117 is outwards facing, neutrons are scattered back (reflected) into the nuclear reactor core 101 to cause more fissions and increase reactivity of the nuclear reactor core 101. When the absorber material 117 is inwards facing towards the nuclear reactor core 101 and the reflector material 116 is outwards facing, neutrons are absorbed and further fissions are stopped to decrease reactivity of the nuclear reactor core 101. In a terrestrial land application, the nuclear reactor core 101 may include control rods 118A-N (see FIG. 1D) composed of chemical elements such as boron, silver, indium, and cadmium that are capable of absorbing many neutrons without themselves fissioning.

[0049] Neutron reflector 140, e.g., shown as the outer reflector region, can be filler elements disposed between outermost nuclear fuel tiles 104A-N and the control drums 115A-N as well as around the control drums 115A-N. Reflector 140 can be formed of a moderator that is disposed between the outermost nuclear fuel tiles 104A-N and an optional barrel (e.g., formed of beryllium). The reflector 140 can include hexagonal or partially hexagonal shaped filler elements and can be formed of a neutron moderator (e.g., beryllium oxide, BeO). Although not required, nuclear reactor 107 can include the optional barrel (not shown) to surround the bundled collection that includes the insulator element array 112, moderator element array 113, nuclear fuel tile array 114 of the nuclear reactor core 101, as well as the reflector 140. As depicted, the control drums 115A-N reside on the perimeter of the pressure vessel 160 and can be interspersed or disposed within the reflector 140, e.g., surround a subset of the filler elements (e.g., reflector blocks 141A-N) forming the reflector 140.

[0050] Pressure vessel 160 can be formed of aluminum alloy, carbon-composite, titanium alloy, a radiation resilient SiC composite, nickel based alloys (e.g., Inconel™ or Haynes™), or a combination thereof. Pressure vessel 160 and nuclear reactor system 100 can be comprised of other components, including cylinders, piping, and storage tanks that transfer a moderator coolant that flows through moderator coolant passages 121A-N; and a separate nuclear fuel coolant, such as a propellant (e.g., hydrogen gas or liquid) that flows through the fuel coolant passages 141A-N. The moderator coolant and the nuclear fuel coolant can be a gas or a liquid, e.g., that transitions from a liquid to a gas state during a burn cycle of the nuclear reactor core 101 for thrust generation in an NTP nuclear reactor system 100. Hydrogen is for an NTP nuclear reactor system 100. In NEP or FSP applications, the nuclear reactor system 100 circulates a working fluid, such as He, neon, HeXe, CO_2, instead.

[0051] In the example of FIG. 1B, nuclear reactor system 100 advantageously enables the moderator coolant to flow through the moderator coolant passages 121A-N and a separate nuclear fuel coolant (e.g., a propellant, such as hydrogen gas) to flow through the fuel coolant passages 141A-N. The moderator coolant passages 121A-N are flattened ring shaped (e.g., O-shape) openings, such as a channels or holes to allow the moderator coolant to pass through in the nuclear reactor core 101 and into a heat sink (not shown) via a dedicated moderator coolant loop, for example. The fuel coolant passages 141A-N are channels or holes to allow the nuclear fuel coolant to pass through in the nuclear reactor core 101 and into a thrust chamber (not shown) for propulsion in a separate nuclear fuel coolant loop, for example.

[0052] In an alternative implementation, a coolant that is shared between the moderator elements 103A-N and the nuclear fuel tiles 104A-N may be flowed through both the moderator coolant passages 121A-N and the fuel coolant passages 141A-N. In the alternative implementation, the coolant that flows through the plurality of fuel elements 104A-N can include helium, FLiBe molten salt formed of lithium fluoride (LiF), beryllium fluoride (BeF_2), sodium, He, HeXe, CO_2, neon, or HeN. The shared coolant flows through the moderator coolant passages 121A-N before the shared coolant is heated in the nuclear fuel tiles 104A-N. This keeps the moderator elements 103A-N cool.

[0053] FIG. 1C illustrates a nuclear reactor system 100 that implements a pressure vessel 160 with the in-vessel shield liner 132 on the interior wall 133 and including control drums 115A-N. FIG. 1D illustrates a nuclear reactor system 100 that implements the pressure vessel 160 with the in-vessel shield liner 132 on the interior wall 133 and including control rods 118A-N. Control rods 118A-N may be positioned in an area of the nuclear reactor core 101 to regulate the neutron population and nuclear reactor power level during operation by changing reactivity of the nuclear reactor core 101. Control rods 118A-N protrude from the top of the pressure vessel 160, can insert the length of the nuclear reactor 107, but also have the ability to be withdrawn from the nuclear reactor 107. Control drums 115A-N enable horizontal, axially symmetrical configurations for the nuclear reactor 107, which can be more easily transported and deployed. Control drums 115A-N also maximize the usable volume in the nuclear reactor core 101, as they can be integrated into the reflector 140, while control rods 118A-N leave an un-fueled void in the nuclear reactor core 101 when withdrawn.

[0054] Reflector 140 may be an integrally formed body (e.g. a tube or pipe), to surround the nuclear reactor core 101, or it may be several components or parts, such as a reflector region made of reflector blocks 141A-N, which surround the nuclear reactor core 101. Reflector 140 elastically scatters neutrons attempting to escape the nuclear reactor core 101, and redirects these neutrons back toward the nuclear reactor core 101 in order to create more fission events and therefore more energy. Reflector 140 also has a secondary purpose: by redirecting neutrons back toward the nuclear reactor core 101, the reflector 140 necessarily reduces the amount of neutrons impacting the pressure vessel 160 and areas beyond. Reflector 140 reduces the fast neutron fluence or fast neutron flux, the total length travelled by all free fast neutrons per time and volume outside the nuclear reactor core 101. Efficacy of the reflector 140 in both the electron scattering and the fast neutron fluence reduction is logarithmically improved by increasing the thickness of the reflector 140.

[0055] In a conventional nuclear reactor, the thickness of the reflector 140 is increased until the fast neutron fluence outside the reflector 140 is acceptably low: this thickness is often materially beyond the thickness required to efficiently reflect neutrons back toward the nuclear reactor core 101. In contrast, the nuclear reactor system 100 described herein additionally includes in-vessel shield 105, which can behave as a neutron poison, and is designed to absorb free neutrons, rather than reflect free neutrons back toward the nuclear reactor core 101. In-vessel shield 105 can absorb free neutrons to stop fast neutron fluence through the neutron poison. In-vessel shield 105, while being significantly thinner than the reflector 140, can nevertheless reduce the fast neutron fluence in an area by the same amount as the reflector 140. Although the in-vessel shield 105 can have the effect of neutron poisoning, the in-vessel shield 105 does not behave as a pure neutron poison because the neutron poison is composited within a moderator to form the in-vessel shield 105. By using a combination of materials with large neutron absorption cross-sections as in-vessel shield 105 between the pressure vessel 160 and the reflector 140 of the nuclear reactor 107, the fast neutron fluence can be reduced to acceptable levels.

[0056] Pressure vessel 160 includes the reflector 140 disposed therein, for the primary purpose of redirecting neutrons back to the nuclear reactor core 101, and secondarily for the purpose of reducing fast neutron fluence to an acceptable level at and beyond the pressure vessel 160. Pressure vessel 160 is then lined with in-vessel shield 105, with the primary purpose of reducing fast neutron fluence to an acceptable level at and beyond the pressure vessel 160. The in-vessel shield 105 is disposed between the pressure vessel 160 and the reflector 140. Together, the in-vessel shield 105 and the reflector 140 are able to reduce fast neutron fluence to an acceptable level at and beyond the pressure vessel 160, while the combined thickness of the in-vessel shield 105 and the reflector 140 is overall thinner than an equivalent reflector 140 without a paired in-vessel shield 105 reducing fast neutron fluence to the same acceptable or equivalent level. Overall, this allows for a combined thinner reflector 140 and in-vessel shield 105 than a reflector 140 alone, and a smaller pressure vessel 160 and nuclear reactor 107 with a reduced volume and mass.

[0057] Because of fast neutron fluence concerns, any size nuclear reactor system 100 (large or small) can benefit from the in-vessel shield 105. In a large nuclear reactor system 100, the fast neutron fluence to the pressure vessel 160 is limited by the distance of the active nuclear reactor core 101 from the pressure vessel 160. For a volume-constrained small nuclear reactor system 100, the use of in-vessel shield 105 can be very advantageous. In-vessel shield 105 reduces the radiation field around the nuclear reactor system 100, which in turn decreases the activation in the area outside (e.g., around) the nuclear reactor 107, decreases the footprint, and aids in ease of install.

[0058] During operation of the nuclear reactor system 100, the materials disposed within the pressure vessel 160 become activation products, made radioactive by the free neutrons. Nuclear reactor system 100 includes an inner density of activation product within the nuclear reactor core 101 and an outer density of activation product outside the nuclear reactor core 101. The outer density of activation product is lower than the inner density of activation product during operation of the nuclear reactor core 101. This is because the purpose of the reflector 140 is to redirect free neutrons back into the nuclear reactor core 101, increasing core activation. The in-vessel shield 105 is made of materials that are selected because the materials absorb free neutrons while becoming minimally radioactive: this further reduces activation outside the nuclear reactor core 101, in comparison to the activation within the nuclear reactor core 101.

[0059] In-vessel shield 105 is formed out of two or more neutron absorbing materials, at least one of which is a near black neutron absorbing material, and another is a gray neutron absorbing material. Near black neutron absorbing material is more efficient at absorbing free neutrons and reducing neutron fluence than a gray neutron absorbing material in similar quantities. Near black neutron absorbing material is a composited ceramic material, that is, a composited high-temperature neutron absorbing ceramic. For example, the composited ceramic material includes mixtures of neutron absorbing ceramics, such as B_4C, HfC, and Gd_2O_3 with radiation tolerant materials, such as Al_2O_3 and SiC, to minimize radiation induced swelling. Adding radiation tolerant materials, such as Al_2O_3

and SiC, to form the composted ceramic material increases temperature and radiation endurance of the near black neutron absorbing material.

[0060] The gray neutron absorbing material includes a heavy metal. Together, the near black neutron material and the gray neutron absorbing material form an in-vessel shield 105. In-vessel shield 105 is fabricated through advanced manufacturing of the composited materials specifically engineered for shielding. This contrasts with the current paradigm of shield selection based on selection of the generic class of engineering alloys and structural ceramics, perhaps with some degree of isotopic enrichment. In-vessel shield 105 may be primarily made of composited ceramics, and therefore could also be described in such examples as a composited ceramic shield.

[0061] FIG. 2 is an overall shielding thickness graph 200 for a reflector 140 and in-vessel shield 105 in eleven different implementations of the nuclear reactor 107A-K. Overall shielding graph 200 shows that a reduction of the total thickness 206A-K of the nuclear reactor 107 is achieved by: (a) increasing the in-vessel shield thickness 210A-K; and (b) reducing the reflector thickness 211A-K. In other words, increasing the percentage of overall shielding as in-vessel shield 205A-K directly reduces the total thickness 206A-K of the nuclear reactor 107. Accordingly, the in-vessel shield 105 technology can improve performance of the nuclear reactor system 100 by enabling the nuclear to be lighter and more compact.

[0062] In the overall shielding thickness graph 200, white bars represent the in-vessel shield thickness 210A-K added by the in-vessel shield 105, while the black bars represent the reflector thickness 211A-K added by the reflector 140. Total thickness 206A-K is the sum of the in-vessel shield thickness 210A-K and the reflector thickness 211A-K. As shown, in a first implementation of nuclear reactor 107A, nuclear reactor 107A includes a total thickness 206A of 30 centimeters (cm), where the reflector thickness 211A is 30 cm and the in-vessel shield thickness 210K is 0 cm. In the eleventh implementation of nuclear reactor 107K, nuclear reactor 107K includes a total thickness 206K of 10 cm, where the reflector thickness 211K is 0 cm and the in-vessel shield thickness 210K is 10 cm.

[0063] Based on the total thicknesses 206A-K of the overall shielding thickness graph 200, in-vessel shield 105 has a shield diffusion length three times shorter than a reflector diffusion length of the reflector 140. Diffusion length is in part a factor of the diffusion coefficient of the material, as well as the macroscopic absorption cross section of that material. This diffusion length difference of three between the reflector 140 and in-vessel shield 105 is based on differences in the materials forming the in-vessel shield 105 and the reflector 140. Diffusion length difference may vary, e.g., based on materials selected, differences in the in-vessel shield thickness 210A-K, reflector thickness 211A-K, and overall geometry of the nuclear reactor core 101.

[0064] In FIG. 2, the total thickness 206A-K represents the same overall amount of neutron fluence reduction. As shown, a 10 cm in-vessel shield thickness 210K for the in-vessel shield 105 results in the same neutron fluence reduction as a 30 cm reflector thickness 211A for the reflector 140. Advantageously, the 10 cm in-vessel shield thickness 210K for the in-vessel shield 105 achieves a pressure vessel 160 with a 20 cm smaller radius, which is a 66% reduction in total thickness 206K. Moreover, the nuclear reactor 107K can be lighter, as the 10 cm of in-vessel shield thickness 210K may weigh less than the equivalent 30 cm of reflector thickness 211A, and in-vessel shield 105 equivalently reduces fast neutron fluence at and beyond the pressure vessel 160.

[0065] A second implementation of nuclear reactor 107B has 1 cm of in-vessel shield thickness 210B, which reduces total thickness 206B by 2 cm compared to the first implementation of the nuclear reactor 107A where no in-vessel shield 105 is present. Every 1 cm of in-vessel shield thickness 210 reduces the need for 3 cm of reflector thickness 211. Based on the overall shielding thickness graph 200, an optimal balance can be met: if only 15 cm reflector thickness 211F is needed to efficiently reflect free neutrons back toward the nuclear reactor core 101, then the sixth implementation of nuclear reactor 107F shows that only 5 cm in-vessel shield thickness 210F is needed to efficiently reduce fast neutron fluence to an acceptable level at and beyond the pressure vessel 160.

[0066] In a conventional nuclear reactor without the in-vessel shield 105, the pressure vessel 160 needs 30 cm of additional radius to properly reflect free neutrons back toward the nuclear reactor core 101 and to efficiently reduce fast neutron fluence to an acceptable level at and beyond the pressure vessel 160. Advantageously, the nuclear reactor system 100 with the in-vessel shield 105 only requires 20 cm of additional radius to properly reflect free neutrons back toward the nuclear reactor core 101 and to efficiently reduce fast neutron fluence to an acceptable level at and beyond the pressure vessel 160. An even further improvement in reduction in size and mass can be achieved if the nuclear reactor system 100 includes control rods 118A-N in a drum form factor to alternatively face reflector blocks 141A-N of the reflector 140 and moderator elements 103A-N of nuclear reactor core 101. In such a nuclear reactor system 100 with control rods 118A-N, the pressure vessel 160 does not necessarily require a fully efficient reflector 140 to reflect free neutrons back toward the nuclear reactor core 101.

[0067] Accordingly, overall shielding thickness graph 200 of FIG. 2 depicts how to design a nuclear reactor system 100, where a reflector block 141 of the plurality of reflector blocks 141A-N has a reflector thickness 211, a reflector diffusion coefficient based on a reflector material that forms the reflector block 141, and a reflector macroscopic absorption cross section based on the reflector material. Additionally, the in-vessel shield 105 has an in-vessel shield thickness 210, a shield diffusion coefficient based on an in-vessel shield material that forms the in-vessel shield 105, and an in-vessel shield macroscopic absorption cross section based on the in-vessel shield material. The reflector block 140 in combination with the in-vessel shield 105 have a combined diffusion length, based at least on: the reflector thickness 211, the reflector diffusion coefficient based on the reflector material, the reflector macroscopic absorption cross section based on the reflector material, the in-vessel shield thickness 210, the shield diffusion coefficient based on the in-vessel shield material, and the shield macroscopic absorption cross section based on the in-vessel shield material. In an example efficient nuclear reactor system 100, the reflector thickness 211 and the in-vessel shield thickness 210 when added together are less (e.g., not thicker) than double the combined diffusion length. This is because the performance of a reflector 140 does not materially improve when the reflector 140 is thicker than double the diffusion length of the reflector material. Therefore, the nuclear reactor system 100 including both the reflector 140 and in-vessel shield 105 can provide the same amount of diffusion as a conventional nuclear reactor system that includes the reflector 140 (and no in-vessel shield 105), but provide the equivalent diffusion in a much smaller amount of space.

[0068] FIG. 3 is an example depletion graph 300 of two different types of nuclear reactor cores 101A-B. As shown, the coefficient k-effective (k-eff) 310 over the lifetime 305 as measured in years of the nuclear reactor cores 101A-B is improved. K-eff 310, also known as the neutron multiplication factor, characterizes the criticality state of the fissile material in the nuclear reactor core 101. Generally K-eff = number of neutrons produced / number of neutrons lost (through leakage or absorption). If K-eff 310 is greater than or equal to 1, only then can the nuclear fission chain reaction can be sustained. As shown, a first reactor nuclear core architecture 301A enables a nuclear reactor core 101A with a lifetime 305A of approximately 10 years. A second nuclear reactor core architecture 301B enables a nuclear reactor core 101B with a lifetime 305B of approximately 15 years. Based on the expected lifetimes 305A-B of the nuclear reactor cores 101A-B, an appropriate amount and type of in-vessel shield 105 and reflector 140 are selected. Even if the in-vessel shield 105 or reflector 140 is a non-burnable neutron poison, ultimately all neutron poisons degrade in efficacy as they experience neutron fluence for an extended period of time. Thus, the first nuclear reactor core 301A potentially may require a thicker in-vessel shield 105 than the second nuclear reactor core 301B, e.g., assuming that any effect of geometrical buckling is similar in both nuclear reactor core architectures 301A-B.

[0069] FIG. 4 is an in-vessel shield material optimization chart 400 for the nuclear reactor system 100. In-vessel shield material optimization chart 400 shows three Pareto fronts 411-413, including a short Pareto front 411, a medium Pareto front 412, and a long Pareto front 413. Each Pareto front 411-413 is divided between different in-vessel shield materials with a short lifetime 401, in-vessel shield materials with a medium lifetime 402, and in-vessel shield materials with a long lifetime 403.

[0070] As explained in FIG. 3, lifetime 305 (which correlates to K-effective 305) is how long the nuclear reactor system 100 can operate. In FIG. 4, the three lifetimes 401-403 are assumed to be the most common initial selection criteria of the in-vessel shield material optimization chart 400. A short lifetime 401, a medium lifetime 402, and a long lifetime 403 determine the number of Pareto fronts 411-413. Pareto fronts 411-413 are utilized to evaluate design options of the nuclear reactor system 100 and the design options are on a Pareto front 411-413 if the design option scores the highest on all figures of merit possible without decreasing other figures of merit. Therefore, an in-vessel shield material on the short Pareto front 411 at a given cost 405 will have the greatest reduction in pressure vessel fast neutron fluence 406 for the nuclear reactor system 100 with a short lifetime 401.

[0071] Three Pareto fronts 411-413 of FIG. 4 demonstrate how different neutron poisons can fit in the design of the in-vessel shield 105. First, a simplistic B_4C material effectively reduces fast neutron fluence at a minimal cost. Boron-10 within B_4C (also known as $^{10}B_4C$) has a large absorption cross section of fast spectrum neutrons. This means that an in-vessel shield 105 formed of B_4C does not need to be as thick as another in-vessel shield 105 formed of another material with a smaller absorption cross section of fast spectrum neutrons. Pure B_4C has a short lifetime because of helium-induced swelling that B_4C undergoes in a radiation environment. Therefore, pure B_4C material is on the line of the short Pareto front 411.

[0072] Second, B_4C-Al_2O_3 composite or B_4C-SiC composites can be on the order of 50% B_4C by weight and better accommodate the swelling of B_4C. B_4C-Al_2O_3 composite or B_4C-SiC composites are less effective in shielding fast neutrons than pure B_4C. Thus, B_4C-Al_2O_3 composite or B_4C-SiC composites are on the line of the medium Pareto front 412.

[0073] Third, borated stainless steel is less than 5% boron by weight, but is very dimensionally-stable under irradiation. Borated stainless steel is on the line of the long Pareto front 413. Based on the in-vessel shield material optimization chart 400, the two or more neutron absorbing materials to form the in-vessel shield 105 are selected, including a near black neutron absorbing material and a gray neutron absorbing material.

[0074] FIG. 5 is a flowchart depicting an in-vessel shield material selection method 500. This example assumes that the dependent variable is material cost, and the independent variables are lifetime 305 of the nuclear reactor system 100, reflector thickness 211, an amount of fast neutron fluence reduced by the reflector 140, and a maximum radius of the pressure vessel 160. Any of these independent variables could be switched for the dependent variable: for example, if a nuclear reactor system 100 already has a pressure vessel 160 and has been retrofitted with in-vessel shield 105, then the material selection method 500 can determine the reflector thickness 211 of the reflector 140 needed for the nuclear reactor system 100 to run efficiently. In-vessel shield material selection method 500 assumes that the radius of the nuclear reactor core 101 and the fast neutron fluence produced by the nuclear reactor core 101 are known values.

[0075] In block 510, the in-vessel shield material selection method 500 includes determining the lifetime 305 of the nuclear reactor system 100. Moving to block 520, the in-vessel shield material selection method 500 further includes determining a reflector thickness 211 that is efficient based on: (i) the nuclear reactor core 101; (ii) a reflector material utilized to form the reflector 140; (iii) geometry of the pressure vessel 160; or (iv) a combination thereof. Ultimately, a reflector thickness 211 is determined that most efficiently redirects neutrons back to the nuclear reactor core 101.

[0076] Continuing to block 530, the in-vessel shield material selection method 500 includes determining the reduction of in vessel fast neutron fluence at and beyond the pressure vessel 160 caused by the reflector 140. Any amount of fast neutron fluence reduction by the reflector 140 reduces the amount of fast neutron fluence reduction required by the in-vessel shield 105 to preserve the targeted reduction in pressure vessel 160 fast neutron fluence for the nuclear reactor system 100. In block 540, the in-vessel shield material selection method 500 includes determining a maximum pressure vessel radius of the pressure vessel 160 based on: (i) installation; (ii) transportation; and/or (iii) footprint requirements of the nuclear reactor system 100, e.g., a nuclear reactor core radius of the nuclear reactor core 101.

[0077] In block 550, the in-vessel shield material selection method 500 includes selecting the in-vessel shield material of the in-vessel shield. Referring to the in-vessel shield material optimization chart 400 of FIG. 4, the proper Pareto front 411-413 can be selected based on the short lifetime 401, medium lifetime 402, and long lifetime 403 (see block 510). The reduction in pressure vessel fast neutron fluence is determined by two factors: (i) a reduction in fast neutron fluence requirement of the pressure vessel 160; and (b) a maximum thickness of the in-vessel shield 105. The reduction in fast neutron fluence requirement of the pressure vessel 160 is the difference between the fast neutron fluence created by the nuclear reactor core 101 and the fast neutron fluence reduction due to the reflector 140 from block 530. The maximum thickness of the in-vessel shield 105 is the radius of the pressure vessel 160 from block 540 minus the radius of the nuclear reactor core 101 and the reflector thickness 211 of the reflector 140 from block 520.

[0078] With the reduction in the fast neutron fluence requirement of the pressure vessel 160 and the maximum thickness of the in-vessel shield 105, an in-vessel shield material on the proper Pareto front 411-413 is found with a diffusion length able to reduce fast neutron fluence sufficiently to meet the reduction in fast neutron fluence requirement of the pressure vessel 160 while no more than the maximum thickness of the in-vessel shield 105. The list of factors considered in the selection of the in-vessel shield material can further include: average and peak temperature of the nuclear reactor core 101; material and volume of the coolant; geometry of the nuclear reactor system 100; and the nuclear, thermal, and mechanical limits of the various in-vessel shield material to form the in-vessel shield 105.

[0079] Selecting the in-vessel shield material can include determining the effectiveness of in-vessel shield 105 based on a computational multi-physics computational model that combines thermal, mechanical, and time-dependent irradiation effects. The computational multi-physics model may be a tool with time-dependent neutron fluxes. The computational multi-physics model may also be used to inform the lifetime 305 and the geometric form of the in-vessel shield 105. Finishing in block 560, the in-vessel shield material selection method 500 includes accepting the in-vessel shield material cost. Once an in-vessel shield material from the in-vessel shield material optimization chart 400 is selected that satisfies the reduction in fast neutron fluence of the pressure vessel 160 on the appropriate Pareto fronts 411-413, an in-vessel shield material with the lowest cost 405 on the appropriate Pareto front 411-413 with sufficient reduction in fast neutron fluence 406 of the pressure vessel 160 is accepted for the nuclear reactor system 100.

[0080] FIG. 6 is a flowchart depicting an in-vessel shield method 600 that includes in-vessel shield manufacture 601, as well as techniques for in-vessel shield design and installation 602. As shown in FIG. 6, a manufacturing flow 601 includes blocks 610, 615, 620, and 625, which are a subset of a design and installation flow 602. Manufacturing flow 601 is focused on the ceramic and metallurgical processes to manufacture the in-vessel shield 105. Design and installation flow 602 includes the manufacturing flow 601 and additionally includes blocks 610 and 630. Design and installation flow 602 includes designing the in-vessel shield 105 for a particular nuclear reactor system 100, which includes forming the in-vessel shield 105 of an appropriate size and shape (e.g., in-vessel shield tiles 131A-N) for the nuclear reactor system 100. Design and installation flow 602 also includes installing the in-vessel shield 105, e.g., by mounting in-vessel shield tiles 131A-N on the interior wall 133 of the pressure vessel 160.

[0081] Beginning in block 605, the in-vessel shield method 600 includes selecting a nuclear reactor system 100, which includes unique parameters of the nuclear reactor system 100, such as a fast neutron fluence to the pressure vessel 160, radius of the pressure vessel 160, reflector thickness 211, and other parameters relevant to design of the in-vessel shield 105. Nuclear reactor system 100 may include a small nuclear reactor 107, which is designed to produce up to approximately 500 thermal megawatts. A smaller nuclear reactor 107 may benefit more greatly from the space savings achieved by the in-vessel shield 105.

[0082] Moving to block 610, the in-vessel shield method 600 includes selecting two or more neutron absorbing materials. In addition to the material selection method 500 of FIG. 5, in a first example, the selecting the two or more neutron absorbing materials can be based on: (i) an in-vessel shield thickness 210 of the in-vessel shield 105; (ii) a geometric configuration of the in-vessel shield 105; (iii) an estimated reduction in vessel fast fluence of the pressure vessel 160, in which the pressure vessel 160 encapsulates the in-vessel shield 105, and the in-vessel shield 105 encapsulates a nuclear reactor core 101; (iv) an anticipated lifetime of the in-vessel shield 105; or (v) a combination thereof. In a second example, the selecting the two or more neutron absorbing materials can based on: (i) an interior diameter of a pressure vessel 160; (ii) an exterior diameter of a nuclear reactor core 101; (iii) a reflector thickness 211 of a reflector 140; or (iv) a combination thereof. In a third example, the selecting the two or more neutron absorbing materials to form the in-vessel shield 105 can be based on: (i) a pressure vessel diameter of a pressure vessel 160 of the nuclear reactor system 100; (ii) a fast neutron fluence to the pressure vessel 160 of the nuclear reactor system 100; and (iii) neutron fluence outside of the nuclear reactor system 100.

[0083] The two or more neutron absorbing materials include at least one near black neutron absorbing material and at least one gray neutron absorbing material. The selection process of block 610 is like that described in blocks 510, 520, 530, 540, 550, and 560 of the in-vessel shield material selection method 500 of FIG. 5. The near black neutron absorbing material includes a composited ceramic material, while the gray neutron absorbing material includes a heavy metal material. Selecting the near black neutron absorbing material can include selecting an isotopically-tailored near black neutron absorbing material.

[0084] Composited ceramic material can include boron carbide (B_4C), hafnium carbide (HfC), or gadolinium oxide (Gd_2O_3); each acting as strong neutron poisons, providing the near black neutron absorbing material potency in reducing neutron fluence. Composited ceramic material can further include aluminum oxide (Al_2O_3) or silicon carbide (SiC); these structural materials are hardy, inert, and have small neutron capture-cross sections: they physically support the strong neutron poisons while not being strong neutron poisons themselves, and may be categorized as neutron moderators. The near black neutron absorbing material selected may be an isotopically-tailored near black neutron absorbing material. Particular combinations of strong neutron poisons and structural materials for forming the composited ceramic material include: boron-10 carbide ($^{10}B_4C$), a boron-10 carbide and aluminum oxide composite ($^{10}B_4C$-Al_2O_3) of 50% boron-10 carbide by weight percent, a boron-10 carbide and silicon carbide composite ($^{10}B_4C$-SiC) of 50% boron-10 carbide by weight percent, or a borated stainless steel alloy of 5% boron-10 by weight percent. Boron-10 carbide and borated stainless steel are not composited ceramics alone, but must be paired with other ceramics in order to form a composited ceramic.

[0085] Heavy metal material of the gray neutron absorbing material includes tungsten (W), iron (Fe), Nickel (Ni), or copper (Cu), for example, in the form of atomized tungsten heavy metal. The heavy metal material, when in the form of atomized tungsten heavy metal can include atomized tungsten heavy metal with a tungsten content greater than or equal to 90% by weight percent.

[0086] Continuing to block 615, the in-vessel shield method 600 includes eutectic sintering base powders of the near black neutron absorbing material to fabricate a ceramic absorbing powder. Eutectic sintering the near black neutron absorbing material to fabricate a ceramic absorbing powder is performed with spark plasma sintering. For example, base powders to form the ceramic absorbing powder include an average particle size in the range of approximately 80 nm to 50 microns. The base powders are spark plasma sintered to form the ceramic absorbing powder. The ceramic absorbing powder includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns. Sintering in blocks 615 and 625 can include high-vacuum direct-current sintering. Sintering operations in some examples may be substituted for or include an advanced 3D printing process.

[0087] In block 620, the in-vessel shield method 600 further includes kinetically mixing the gray neutron absorbing material with the ceramic absorbing powder fabricated in block 615, creating an in-vessel shield mixture. This ensures thorough, impurity free dispersion.

[0088] Moving to block 625, the in-vessel shield method 600 includes cold press sintering the in-vessel shield mixture into an in-vessel shield 105. The cold press sintering the in-vessel shield mixture into the in-vessel shield 105 includes embedding the plurality of composited ceramic particles inside a heavy metal matrix of the heavy metal material to form the in-vessel shield 105. As noted above, composited ceramic particles include an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

[0089] The cold press sintering the in-vessel shield mixture into the in-vessel shield 105 further includes forming the in-vessel shield 105 as a plurality of in-vessel shield tiles

131A-N. All or a subset of the plurality of in-vessel shield tiles 131A-N are a curved polyhedron shape or a truncated portion thereof. The plurality of in-vessel shield tiles 131A-N can include a base shape with an interlocking geometry pattern. The base shape of the in-vessel shield tiles 131A-N can be selected based in part on: (i) tolerance towards swelling; (ii) reduction of neutron streaming paths; or (iii) a combination thereof.

[0090] Finishing in block 630, the in-vessel shield method 600 includes mounting the in-vessel shield 105 on an interior wall 133 of a pressure vessel 160, for example, by disposing the plurality of in-vessel shield tiles 131A-N on the interior wall 133 of the pressure vessel 160. Interior wall 133 is formed of a continuous surface or a plurality of discontinuous surfaces. Hence, block 630 can include covering the continuous or discontinuous surfaces of the interior wall 133 with the plurality of in-vessel shield tiles 131A-N by joining the interlocking geometry pattern.

[0091] The scope of protection is limited solely by the claims that now follow. That scope is intended and should be interpreted to be as broad as is consistent with the ordinary meaning of the language that is used in the claims when interpreted in light of this specification and the prosecution history that follows and to encompass all structural and functional equivalents. Notwithstanding, none of the claims are intended to embrace subject matter that fails to satisfy the requirement of Sections 101, 102, or 103 of the Patent Act, nor should they be interpreted in such a way. Any unintended embracement of such subject matter is hereby disclaimed.

[0092] 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,” “has,” “having,” “with,” “formed of,” 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.

[0093] 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.

[0094] 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 pressure vessel including an interior wall;

a nuclear reactor core located within the interior wall of the pressure vessel, wherein the nuclear reactor core includes a fuel element array of a plurality of fuel elements and at least one moderator element;

a reflector located inside the pressure vessel that includes a plurality of reflector blocks laterally surrounding the plurality of fuel elements and the at least one moderator element; and

an in-vessel shield located on the interior wall of the pressure vessel to surround the plurality of reflector blocks, wherein:

the in-vessel shield is formed of two or more neutron absorbing materials, and

the two more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material.

2. The nuclear reactor system of claim 1, wherein:

the near black neutron absorbing material includes a composited ceramic material;

the gray neutron absorbing material includes a heavy metal material; and

the composited ceramic material and the heavy metal material form the in-vessel shield.

3. The nuclear reactor system of claim 2, wherein:

the composited ceramic material includes boron carbide (B_4C), hafnium carbide (HfC), or gadolinium oxide (Gd_2O_3); and

the heavy metal material includes tungsten (W), iron (Fe), Nickel (Ni), or copper (Cu).

4. The nuclear reactor system of claim 3, wherein:

the composited ceramic material further includes aluminum oxide (Al_2O_3) or silicon carbide (SiC).

5. The nuclear reactor system of claim 2, wherein:

the composited ceramic material includes:

boron-10 carbide ($^{10}B_4C$),

a boron-10 carbide and aluminum oxide composite ($^{10}B_4C$-Al_2O_3) of 50% boron-10 carbide by weight percent,

a boron-10 carbide and silicon carbide composite ($^{10}B_4C$-SiC) of 50% boron-10 carbide by weight percent, or

a borated stainless steel alloy of 5% boron-10 by weight percent; and

the heavy metal material includes atomized tungsten heavy metal with a tungsten content greater than or equal to 90% by weight percent.

6. The nuclear reactor system of claim 2, wherein:

the composited ceramic material includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

7. The nuclear reactor system of claim 6, wherein:

the plurality of composited ceramic particles are embedded inside a heavy metal matrix of the heavy metal material to form the in-vessel shield.

8. The nuclear reactor system of claim 7, wherein:

the in-vessel shield is formed as a plurality of in-vessel shield tiles; and

the plurality of in-vessel shield tiles are disposed on the interior wall.

9. The nuclear reactor system of claim 8, wherein all or a subset of the plurality of in-vessel shield tiles are a curved polyhedron shape or a truncated portion thereof.

10. The nuclear reactor system of claim 8, wherein:

the plurality of in-vessel shield tiles include a base shape with an interlocking geometry pattern.

11. The nuclear reactor system of claim 10, wherein:

the interior wall is formed of a continuous surface or a plurality of discontinuous surfaces; and

the interlocking geometry pattern of the plurality of in-vessel shield tiles are joined to cover and collectively form an in-vessel shield liner on the continuous or discontinuous surfaces of the interior wall.

12. The nuclear reactor system of claim 11, further comprising:

an inner density of activation product within the nuclear reactor core; and

an outer density of activation product outside the nuclear reactor core;

wherein the outer density of activation product is lower than the inner density of activation product during operation of the nuclear reactor core.

13. The nuclear reactor system of claim 1, wherein:

a reflector block of the plurality of reflector blocks has a reflector thickness, a reflector diffusion coefficient, and a reflector macroscopic absorption cross section;

the in-vessel shield has an in-vessel shield thickness, an in-vessel shield diffusion coefficient, and an in-vessel shield macroscopic absorption cross section;

the reflector block and the in-vessel shield have a combined diffusion length, based at least on: the reflector thickness, the reflector diffusion coefficient, the reflector macroscopic absorption cross section, the in-vessel shield thickness, the in-vessel shield diffusion coefficient, and the in-vessel shield macroscopic absorption cross section; and

the reflector thickness and the in-vessel shield thickness added together are less than double the combined diffusion length.

14. The nuclear reactor system of claim 1, wherein:

each of the fuel elements includes a nuclear fuel;

the nuclear fuel includes a fuel compact comprised of coated fuel particles embedded inside a high-temperature matrix; and

the high-temperature matrix includes silicon carbide, zirconium carbide, titanium carbide, niobium carbide, tungsten, molybdenum, or a combination thereof.

15. The nuclear reactor system of claim 14, wherein:

the coated fuel particles includes tristructural-isotropic (TRISO) fuel particles or bistructural-isotropic (BISO) fuel particles.

16. The nuclear reactor system of claim 15, further comprising a plurality of control drums, wherein:

the control drums are interspersed or disposed within the reflector.

17. The nuclear reactor system of claim 1, wherein the nuclear reactor system includes a gas-cooled high-temperature nuclear reactor, a molten salt cooled nuclear reactor, fuel-in-salt nuclear reactor, or a sodium-cooled fast nuclear reactor.

18. The nuclear reactor system of claim 1, further comprising a coolant that flows through the plurality of fuel elements, wherein the coolant includes helium, FLiBe molten salt formed of lithium fluoride (LiF) and beryllium fluoride (BeF_2), sodium, He, HeXe, CO_2, neon, or HeN.

19. A method comprising:

selecting two or more neutron absorbing materials to form an in-vessel shield, wherein the two or more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material;

eutectic sintering the near black neutron absorbing material to fabricate a ceramic absorbing powder;

kinetically mixing the gray neutron absorbing material and the ceramic absorbing powder to create an in-vessel shield mixture; and

cold press sintering the in-vessel shield mixture into an in-vessel shield.

20. The method of claim 19, wherein:

the near black neutron absorbing material includes a composited ceramic material; and the gray neutron absorbing material includes a heavy metal material.

21. The method of claim 20, wherein:

the composited ceramic material includes boron carbide (B_4C), hafnium carbide (HfC), or gadolinium oxide (Gd_2O_3); and

the heavy metal material includes tungsten (W), iron (Fe), Nickel (Ni), or copper (Cu).

22. The method of claim 21, wherein:

the composited ceramic material further includes aluminum oxide (Al_2O_3) or silicon carbide (SiC).

23. The method of claim 20, wherein:

the composited ceramic material includes at least one of:

boron-10 carbide ($^{10}B_4C$),

a boron-10 carbide and aluminum oxide composite ($^{10}B_4C$-Al_2O_3) of 50% boron-10 carbide by weight percent,

a boron-10 carbide and silicon carbide composite ($^{10}B_4C$-SiC) of 50% boron-10 carbide by weight percent, or

a borated stainless steel alloy of 5% boron-10 by weight percent; and

the heavy metal material includes atomized tungsten heavy metal with a tungsten content greater than or equal to 90% by weight percent.

24. The method of claim 20, wherein:

the composited ceramic material includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

25. The method of claim 19, wherein

the ceramic absorbing powder includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

26. The method of claim 24, wherein:

the cold press sintering the in-vessel shield mixture into the in-vessel shield includes embedding the plurality of composed ceramic particles inside a heavy metal matrix of the heavy metal material to form the in-vessel shield.

27. The method of claim 26, wherein the cold press sintering the in-vessel shield mixture into the in-vessel shield further includes forming the in-vessel shield as a plurality of in-vessel shield tiles.

28. The method of claim 27, further comprising:

disposing the plurality of in-vessel shield tiles on an interior wall of a pressure vessel.

29. The nuclear reactor system of claim 28, wherein all or a subset of the plurality of in-vessel shield tiles are a curved polyhedron shape or a truncated portion thereof.

30. The method of claim 28, wherein:

the plurality of in-vessel shield tiles include a base shape with an interlocking geometry pattern.

31. The method of claim 30, wherein:

the interior wall is formed of a continuous surface or a plurality of discontinuous surfaces; and

the method further comprises covering the continuous or discontinuous surfaces of the interior wall with the plurality of in-vessel shield tiles by joining the interlocking geometry pattern.

32. The method of claim 0, further comprising selecting the base shape of the in-vessel shield tiles, based in part on:

(i) tolerance towards swelling;

(ii) reduction of neutron streaming paths; or

(iii) a combination thereof.

33. The method of claim 32, wherein the base shape is a curved polyhedron shape or a truncated portion thereof.

34. The method of claim 19, wherein the selecting the two or more neutron absorbing materials is based on:

(i) an in-vessel shield thickness of the in-vessel shield;

(ii) a geometric configuration of the in-vessel shield;

(iii) an estimated reduction in vessel fast fluence of a pressure vessel, in which the pressure vessel encapsulates the in-vessel shield, and the in-vessel shield encapsulates a nuclear reactor core;

(iv) an anticipated lifetime of the in-vessel shield; or

(v) a combination thereof.

35. The method of claim 19, wherein the selecting the two or more neutron absorbing materials is based on:

(i) an interior diameter of a pressure vessel;

(ii) an exterior diameter of a nuclear reactor core;

(iii) a reflector thickness of a reflector; or

(iv) a combination thereof.

36. The method of claim 19, wherein eutectic sintering the near black neutron absorbing material to fabricate the ceramic absorbing powder is performed with spark plasma sintering.

37. The method of claim 19, further comprising:

selecting a reflector block to pair with the in-vessel shield; and
selecting an in-vessel shield thickness of the in-vessel shield;
wherein:

the reflector block has a reflector thickness, a reflector diffusion coefficient, and a reflector macroscopic absorption cross section;

the in-vessel shield has an in-vessel shield diffusion coefficient and an in-vessel shield macroscopic absorption cross section;

the reflector block and the in-vessel shield have a combined diffusion length, based at least on: the reflector thickness, the reflector diffusion coefficient, the reflector macroscopic absorption cross section, the selected in-vessel shield thickness, the in-vessel shield diffusion coefficient, and the in-vessel shield macroscopic absorption cross section; and

the reflector thickness and the selected in-vessel shield thickness added together are less than double the combined diffusion length.

38. The method of claim 19, wherein selecting the near black neutron absorbing material includes selecting an isotopically-tailored near black neutron absorbing material.

39. The method of claim 19, further comprising:
selecting a nuclear reactor system; and
wherein:

selecting the two or more neutron absorbing materials to form the in-vessel shield is based on:

(i) a pressure vessel diameter of a pressure vessel of the nuclear reactor system;
(ii) a fast neutron fluence to the pressure vessel; and
(iii) neutron fluence outside of the nuclear reactor system.

40. The method of claim 19, further comprising:
mounting the in-vessel shield on an interior wall of a pressure vessel.

AMENDED CLAIMS
received by the International Bureau on 02 March 2021 (02.03.2021)

What is claimed is:

1. A nuclear reactor system comprising:

a pressure vessel including an interior wall;

a nuclear reactor core located within the interior wall of the pressure vessel, wherein the nuclear reactor core includes a fuel element array of a plurality of fuel elements and at least one moderator element;

a reflector located inside the pressure vessel that includes a plurality of reflector blocks laterally surrounding the plurality of fuel elements and the at least one moderator element; and

an in-vessel shield located on the interior wall of the pressure vessel to surround the plurality of reflector blocks, wherein:

the in-vessel shield is formed of two or more neutron absorbing materials, and

the two more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material.

2. The nuclear reactor system of claim 1, wherein:

the near black neutron absorbing material includes a composited ceramic material;

the gray neutron absorbing material includes a heavy metal material; and

the composited ceramic material and the heavy metal material form the in-vessel shield.

3. The nuclear reactor system of claim 2, wherein:

the composited ceramic material includes boron carbide (B_4C), hafnium carbide (HfC), or gadolinium oxide (Gd_2O_3); and

the heavy metal material includes tungsten (W), iron (Fe), Nickel (Ni), or copper (Cu).

4. The nuclear reactor system of claim 3, wherein:

the composited ceramic material further includes aluminum oxide (Al_2O_3) or silicon carbide (SiC).

5. The nuclear reactor system of claim 2, wherein:
the composited ceramic material includes:
boron-10 carbide ($^{10}B_4C$),
a boron-10 carbide and aluminum oxide composite ($^{10}B_4C$-Al_2O_3) of 50% boron-10 carbide by weight percent,
a boron-10 carbide and silicon carbide composite ($^{10}B_4C$-SiC) of 50% boron-10 carbide by weight percent, or
a borated stainless steel alloy of 5% boron-10 by weight percent; and
the heavy metal material includes atomized tungsten heavy metal with a tungsten content greater than or equal to 90% by weight percent.

6. The nuclear reactor system of claim 2, wherein:
the composited ceramic material includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

7. The nuclear reactor system of claim 6, wherein:
the plurality of composited ceramic particles are embedded inside a heavy metal matrix of the heavy metal material to form the in-vessel shield.

8. The nuclear reactor system of claim 7, wherein:
the in-vessel shield is formed as a plurality of in-vessel shield tiles; and
the plurality of in-vessel shield tiles are disposed on the interior wall.

9. The nuclear reactor system of claim 8, wherein all or a subset of the plurality of in-vessel shield tiles are a curved polyhedron shape or a truncated portion thereof.

10. The nuclear reactor system of claim 8, wherein:
the plurality of in-vessel shield tiles include a base shape with an interlocking geometry pattern.

11. The nuclear reactor system of claim 10, wherein:

the interior wall is formed of a continuous surface or a plurality of discontinuous surfaces; and

the interlocking geometry pattern of the plurality of in-vessel shield tiles are joined to cover and collectively form an in-vessel shield liner on the continuous or discontinuous surfaces of the interior wall.

12. The nuclear reactor system of claim 11, further comprising:

an inner density of activation product within the nuclear reactor core; and

an outer density of activation product outside the nuclear reactor core;

wherein the outer density of activation product is lower than the inner density of activation product during operation of the nuclear reactor core.

13. The nuclear reactor system of claim 1, wherein:

a reflector block of the plurality of reflector blocks has a reflector thickness, a reflector diffusion coefficient, and a reflector macroscopic absorption cross section;

the in-vessel shield has an in-vessel shield thickness, an in-vessel shield diffusion coefficient, and an in-vessel shield macroscopic absorption cross section;

the reflector block and the in-vessel shield have a combined diffusion length, based at least on: the reflector thickness, the reflector diffusion coefficient, the reflector macroscopic absorption cross section, the in-vessel shield thickness, the in-vessel shield diffusion coefficient, and the in-vessel shield macroscopic absorption cross section; and

the reflector thickness and the in-vessel shield thickness added together are less than double the combined diffusion length.

14. The nuclear reactor system of claim 1, wherein:

each of the fuel elements includes a nuclear fuel;

the nuclear fuel includes a fuel compact comprised of coated fuel particles embedded inside a high-temperature matrix; and

the high-temperature matrix includes silicon carbide, zirconium carbide, titanium carbide, niobium carbide, tungsten, molybdenum, or a combination thereof.

15. The nuclear reactor system of claim 14, wherein:

the coated fuel particles includes tristructural-isotropic (TRISO) fuel particles or bistructural-isotropic (BISO) fuel particles.

16. The nuclear reactor system of claim 15, further comprising a plurality of control drums, wherein:

the control drums are interspersed or disposed within the reflector.

17. The nuclear reactor system of claim 1, wherein the nuclear reactor system includes a gas-cooled high-temperature nuclear reactor, a molten salt cooled nuclear reactor, fuel-in-salt nuclear reactor, or a sodium-cooled fast nuclear reactor.

18. The nuclear reactor system of claim 1, further comprising a coolant that flows through the plurality of fuel elements, wherein the coolant includes helium, FLiBe molten salt formed of lithium fluoride (LiF) and beryllium fluoride (BeF_2), sodium, He, HeXe, CO_2, neon, or HeN.

19. A method comprising:

selecting two or more neutron absorbing materials to form an in-vessel shield, wherein the two or more neutron absorbing materials include a near black neutron absorbing material and a gray neutron absorbing material;

eutectic sintering the near black neutron absorbing material to fabricate a ceramic absorbing powder;

kinetically mixing the gray neutron absorbing material and the ceramic absorbing powder to create an in-vessel shield mixture; and

cold press sintering the in-vessel shield mixture into an in-vessel shield.

20. The method of claim 19, wherein:

the near black neutron absorbing material includes a composited ceramic material; and the gray neutron absorbing material includes a heavy metal material.

21. The method of claim 20, wherein:

the composited ceramic material includes boron carbide (B_4C), hafnium carbide (HfC), or gadolinium oxide (Gd_2O_3); and

the heavy metal material includes tungsten (W), iron (Fe), Nickel (Ni), or copper (Cu).

22. The method of claim 21, wherein:

the composited ceramic material further includes aluminum oxide (Al_2O_3) or silicon carbide (SiC).

23. The method of claim 20, wherein:

the composited ceramic material includes at least one of:

boron-10 carbide ($^{10}B_4C$),

a boron-10 carbide and aluminum oxide composite ($^{10}B_4C$-Al_2O_3) of 50% boron-10 carbide by weight percent,

a boron-10 carbide and silicon carbide composite ($^{10}B_4C$-SiC) of 50% boron-10 carbide by weight percent, or

a borated stainless steel alloy of 5% boron-10 by weight percent; and

the heavy metal material includes atomized tungsten heavy metal with a tungsten content greater than or equal to 90% by weight percent.

24. The method of claim 20, wherein:

the composited ceramic material includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

25. The method of claim 19, wherein:

the ceramic absorbing powder includes a plurality of composited ceramic particles with an average particle diameter greater than or equal to approximately 80 nanometers and less than or equal to approximately 100 microns.

26. The method of claim 24, wherein:

the cold press sintering the in-vessel shield mixture into the in-vessel shield includes embedding the plurality of composed ceramic particles inside a heavy metal matrix of the heavy metal material to form the in-vessel shield.

27. The method of claim 26, wherein the cold press sintering the in-vessel shield mixture into the in-vessel shield further includes forming the in-vessel shield as a plurality of in-vessel shield tiles.

28. The method of claim 27, further comprising:

disposing the plurality of in-vessel shield tiles on an interior wall of a pressure vessel.

29. The nuclear reactor system of claim 28, wherein all or a subset of the plurality of in-vessel shield tiles are a curved polyhedron shape or a truncated portion thereof.

30. The method of claim 28, wherein:

the plurality of in-vessel shield tiles include a base shape with an interlocking geometry pattern.

31. The method of claim 30, wherein:

the interior wall is formed of a continuous surface or a plurality of discontinuous surfaces; and

the method further comprises covering the continuous or discontinuous surfaces of the interior wall with the plurality of in-vessel shield tiles by joining the interlocking geometry pattern.

32. The method of claim 30, further comprising selecting the base shape of the in-vessel shield tiles, based in part on:

(i) tolerance towards swelling;

(ii) reduction of neutron streaming paths; or

(iii) a combination thereof.

33. The method of claim 32, wherein the base shape is a curved polyhedron shape or a truncated portion thereof.

34. The method of claim 19, wherein the selecting the two or more neutron absorbing materials is based on:

(i) an in-vessel shield thickness of the in-vessel shield;

(ii) a geometric configuration of the in-vessel shield;

(iii) an estimated reduction in vessel fast fluence of a pressure vessel, in which the pressure vessel encapsulates the in-vessel shield, and the in-vessel shield encapsulates a nuclear reactor core;

(iv) an anticipated lifetime of the in-vessel shield; or

(v) a combination thereof.

35. The method of claim 19, wherein the selecting the two or more neutron absorbing materials is based on:

(i) an interior diameter of a pressure vessel;

(ii) an exterior diameter of a nuclear reactor core;

(iii) a reflector thickness of a reflector; or

(iv) a combination thereof.

36. The method of claim 19, wherein eutectic sintering the near black neutron absorbing material to fabricate the ceramic absorbing powder is performed with spark plasma sintering.

37. The method of claim 19, further comprising:

selecting a reflector block to pair with the in-vessel shield; and
selecting an in-vessel shield thickness of the in-vessel shield;
wherein:

the reflector block has a reflector thickness, a reflector diffusion coefficient, and a reflector macroscopic absorption cross section;

the in-vessel shield has an in-vessel shield diffusion coefficient and an in-vessel shield macroscopic absorption cross section;

the reflector block and the in-vessel shield have a combined diffusion length, based at least on: the reflector thickness, the reflector diffusion coefficient, the reflector macroscopic absorption cross section, the selected in-vessel shield thickness, the in-vessel shield diffusion coefficient, and the in-vessel shield macroscopic absorption cross section; and

the reflector thickness and the selected in-vessel shield thickness added together are less than double the combined diffusion length.

38. The method of claim 19, wherein selecting the near black neutron absorbing material includes selecting an isotopically-tailored near black neutron absorbing material.

39. The method of claim 19, further comprising:
selecting a nuclear reactor system; and
wherein:

selecting the two or more neutron absorbing materials to form the in-vessel shield is based on:

(i) a pressure vessel diameter of a pressure vessel of the nuclear reactor system;
(ii) a fast neutron fluence to the pressure vessel; and
(iii) neutron fluence outside of the nuclear reactor system.

40. The method of claim 19, further comprising:
mounting the in-vessel shield on an interior wall of a pressure vessel.

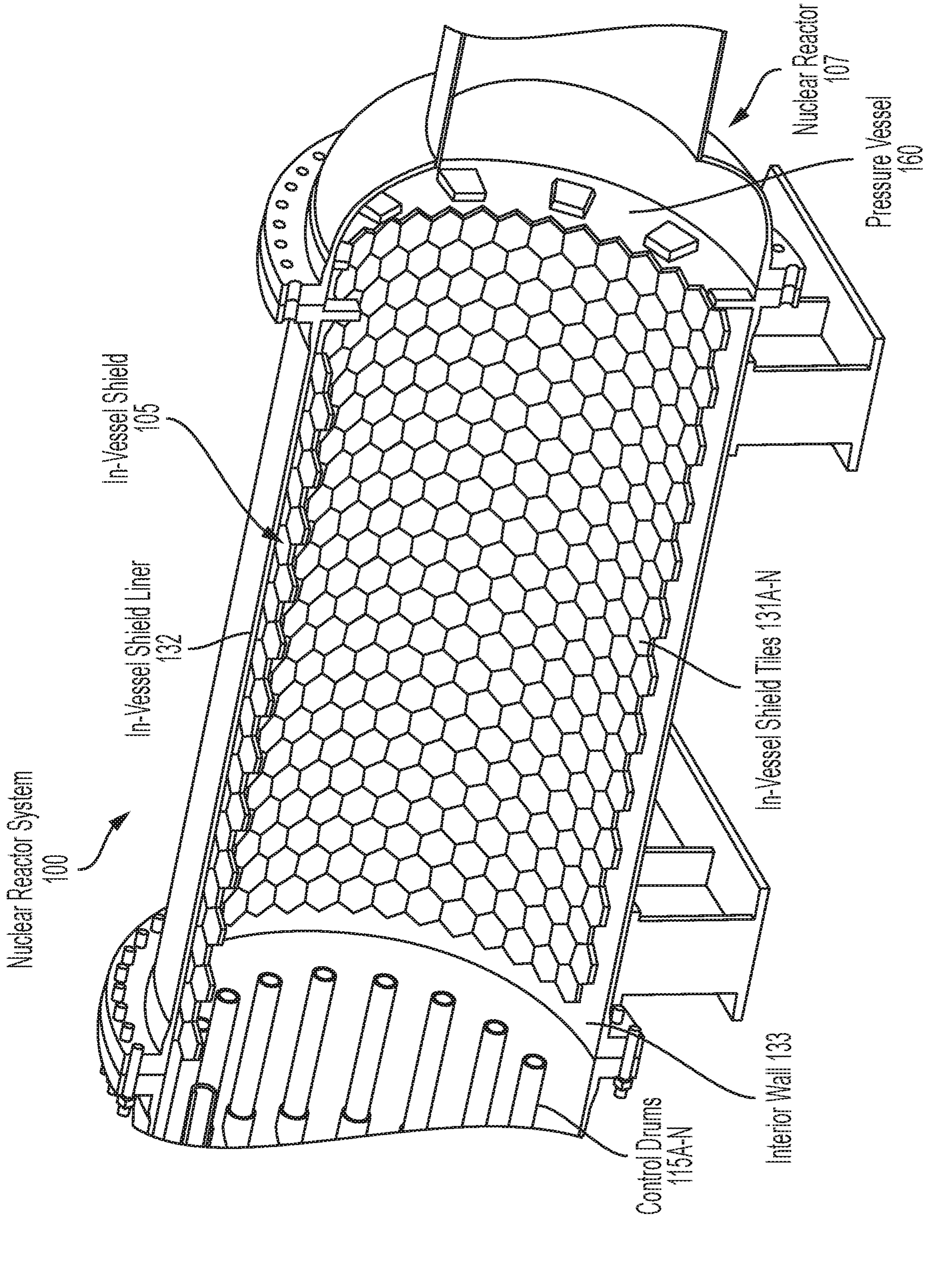

Nuclear Reactor 107
Pressure Vessel 160
In-Vessel Shield 105
In-Vessel Shield Liner 132
Nuclear Reactor System 100
In-Vessel Shield Tiles 131A-N
Interior Wall 133
Control Drums 115A-N

FIG. 1A

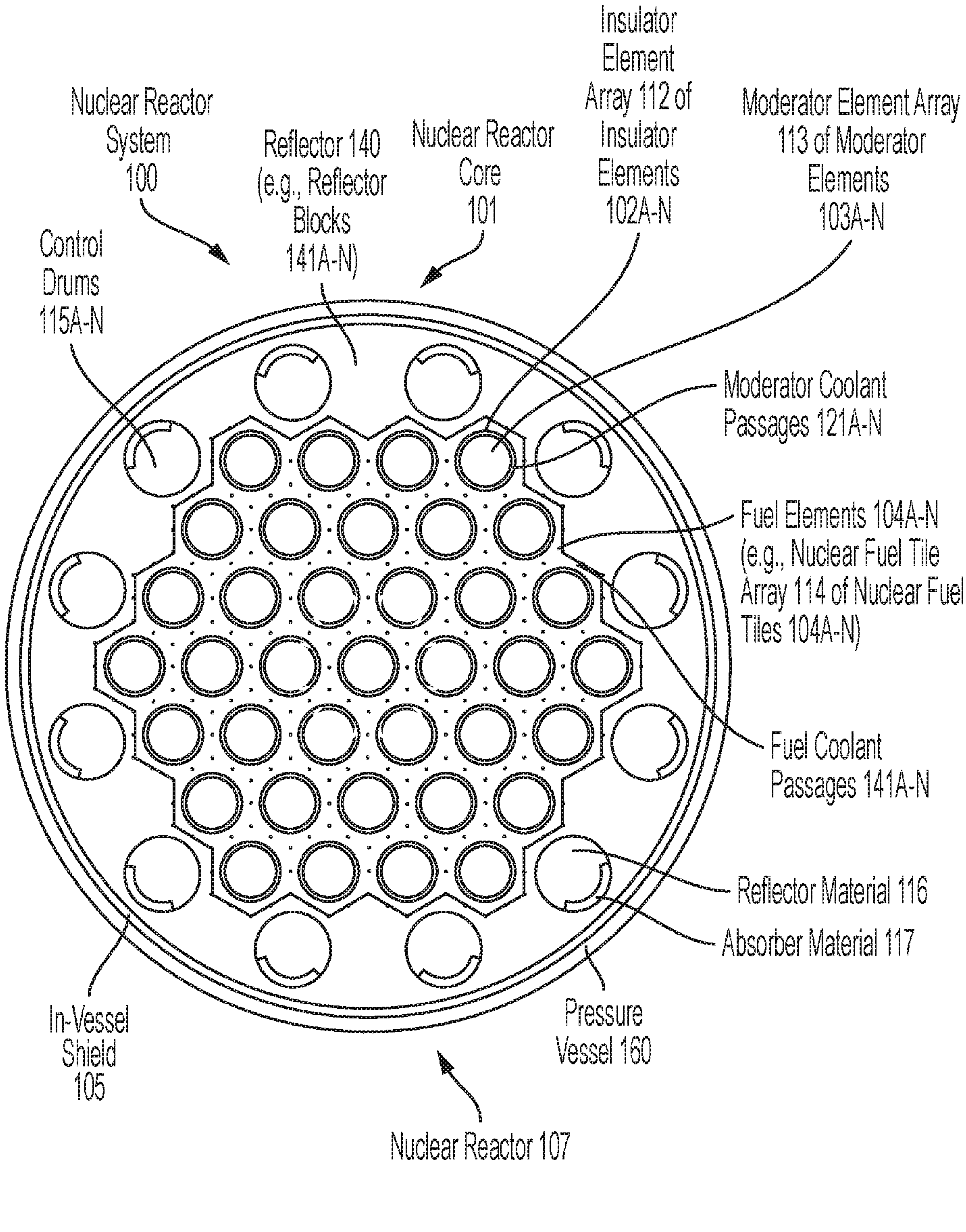

Nuclear Reactor System 100
Reflector 140 (e.g., Reflector Blocks 141A-N)
Nuclear Reactor Core 101
Insulator Element Array 112 of Insulator Elements 102A-N
Moderator Element Array 113 of Moderator Elements 103A-N
Control Drums 115A-N
Moderator Coolant Passages 121A-N
Fuel Elements 104A-N (e.g., Nuclear Fuel Tile Array 114 of Nuclear Fuel Tiles 104A-N)
Fuel Coolant Passages 141A-N
Reflector Material 116
Absorber Material 117
In-Vessel Shield 105
Pressure Vessel 160
Nuclear Reactor 107

FIG. 1B

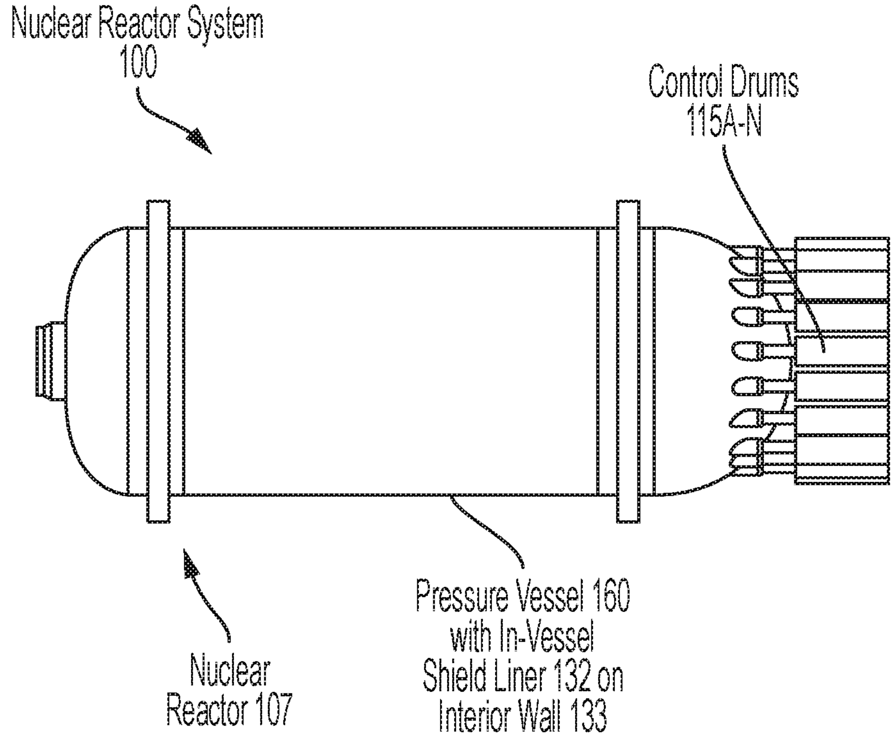

Nuclear Reactor System
100
Control Drums
115A-N
Nuclear
Reactor 107
Pressure Vessel 160
with In-Vessel
Shield Liner 132 on
Interior Wall 133

FIG. 1C

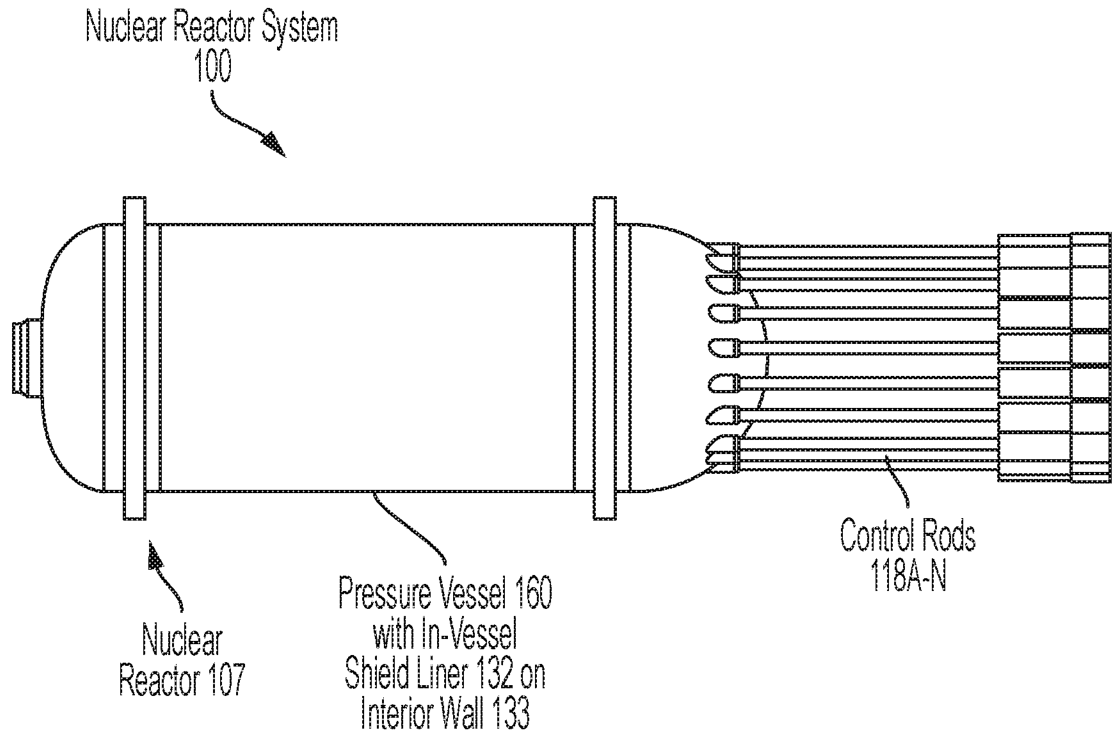
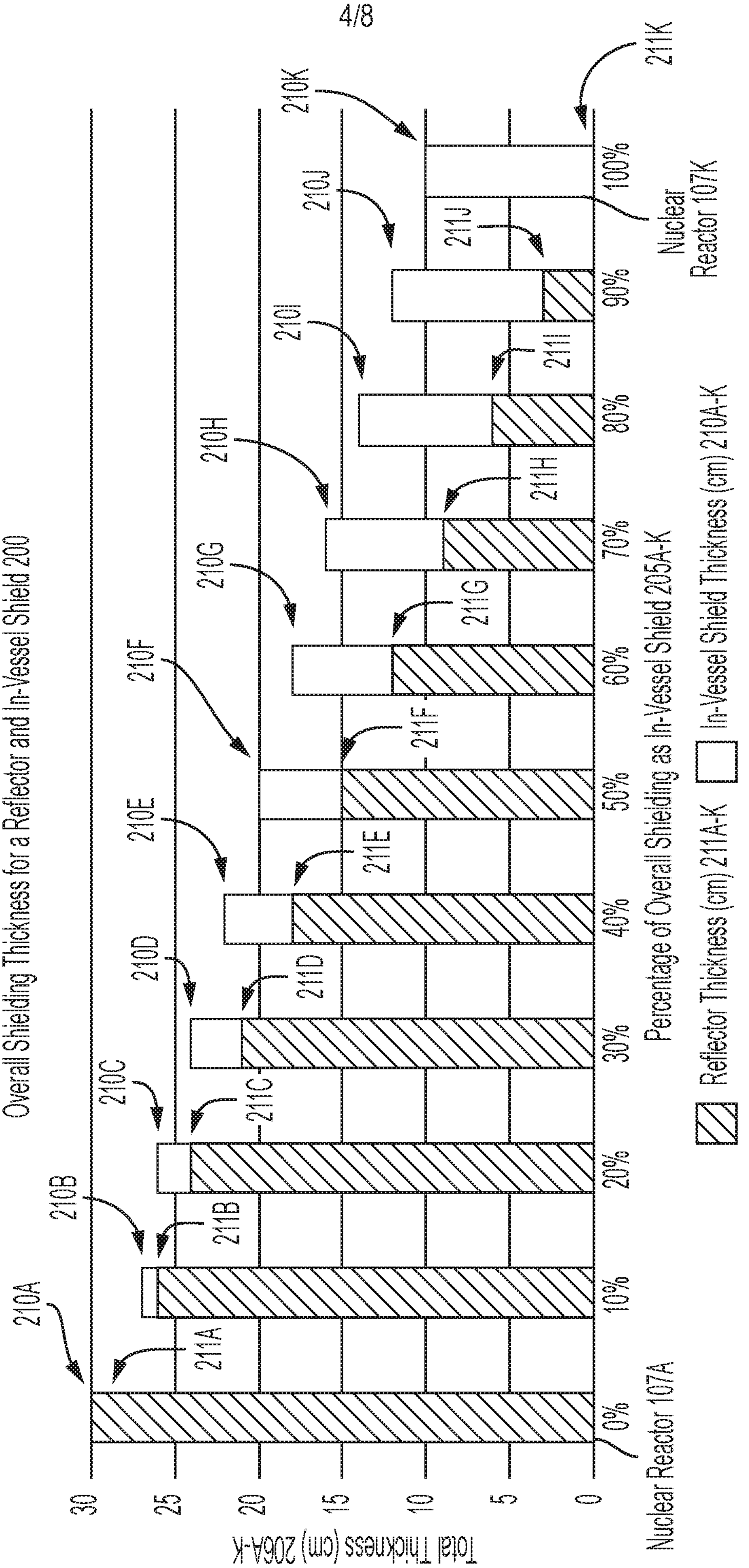
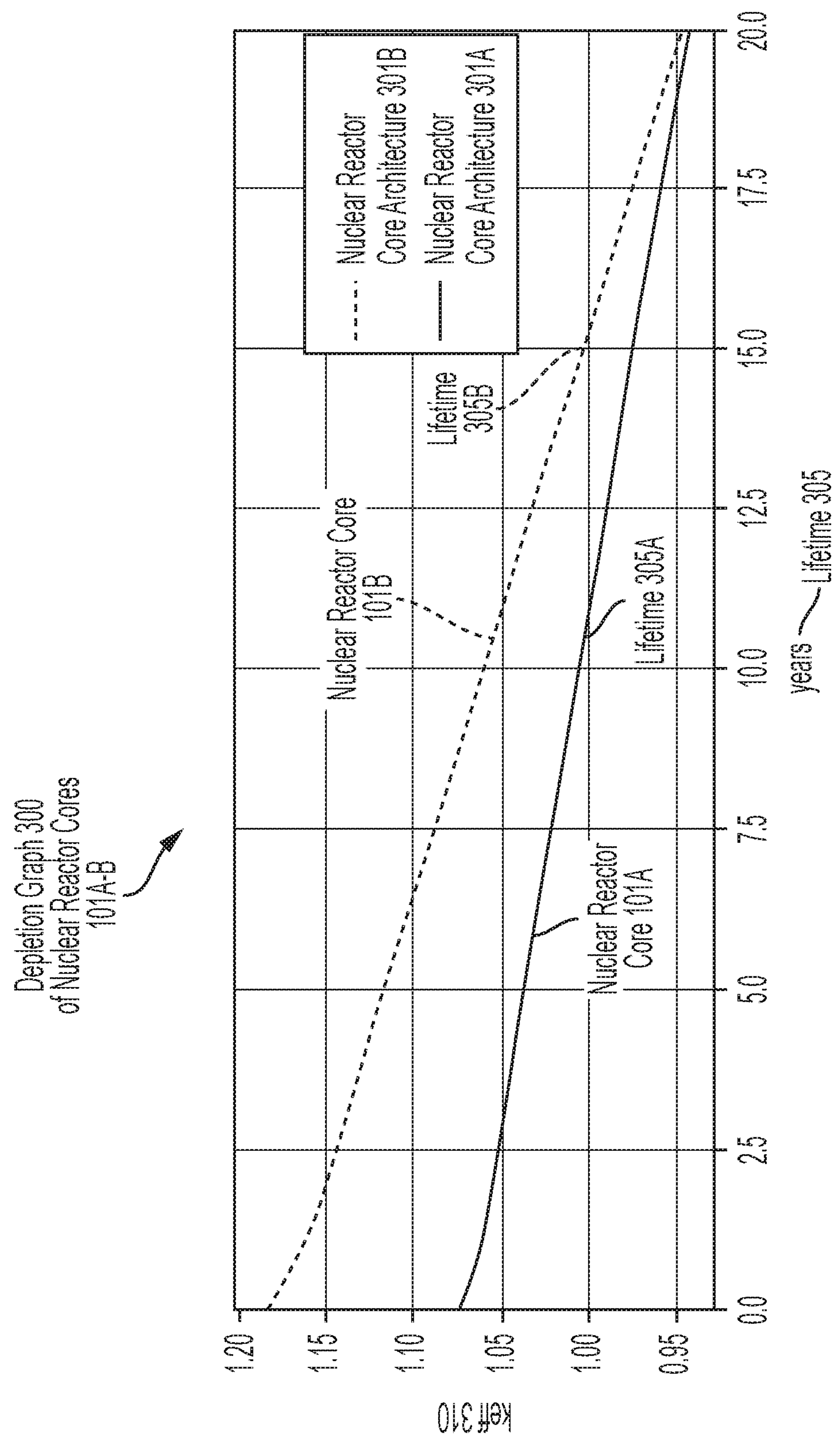

Nuclear Reactor System
100
Nuclear
Reactor 107
Pressure Vessel 160
with In-Vessel
Shield Liner 132 on
Interior Wall 133
Control Rods
118A-N

FIG. 1D



Overall Shielding Thickness for a Reflector and In-Vessel Shield 200
Total Thickness (cm) 206A-K
30
25
20
15
10
5
0
Percentage of Overall Shielding as In-Vessel Shield 205A-K
0%
10%
20%
30%
40%
50%
60%
70%
80%
90%
100%
Nuclear Reactor 107A
Nuclear Reactor 107K
210A
211A
210B
211B
210C
211C
210D
211D
210E
211E
210F
211F
210G
211G
210H
211H
210I
211I
210J
211J
210K
211K
Reflector Thickness (cm) 211A-K
In-Vessel Shield Thickness (cm) 210A-K

FIG. 2



Depletion Graph 300 of Nuclear Reactor Cores 101A-B
Nuclear Reactor Core Architecture 301B
Nuclear Reactor Core Architecture 301A
Nuclear Reactor Core 101B
Lifetime 305B
Lifetime 305A
Nuclear Reactor Core 101A
keff 310
1.20
1.15
1.10
1.05
1.00
0.95
0.0
2.5
5.0
7.5
10.0
12.5
15.0
17.5
20.0
years
Lifetime 305

FIG. 3

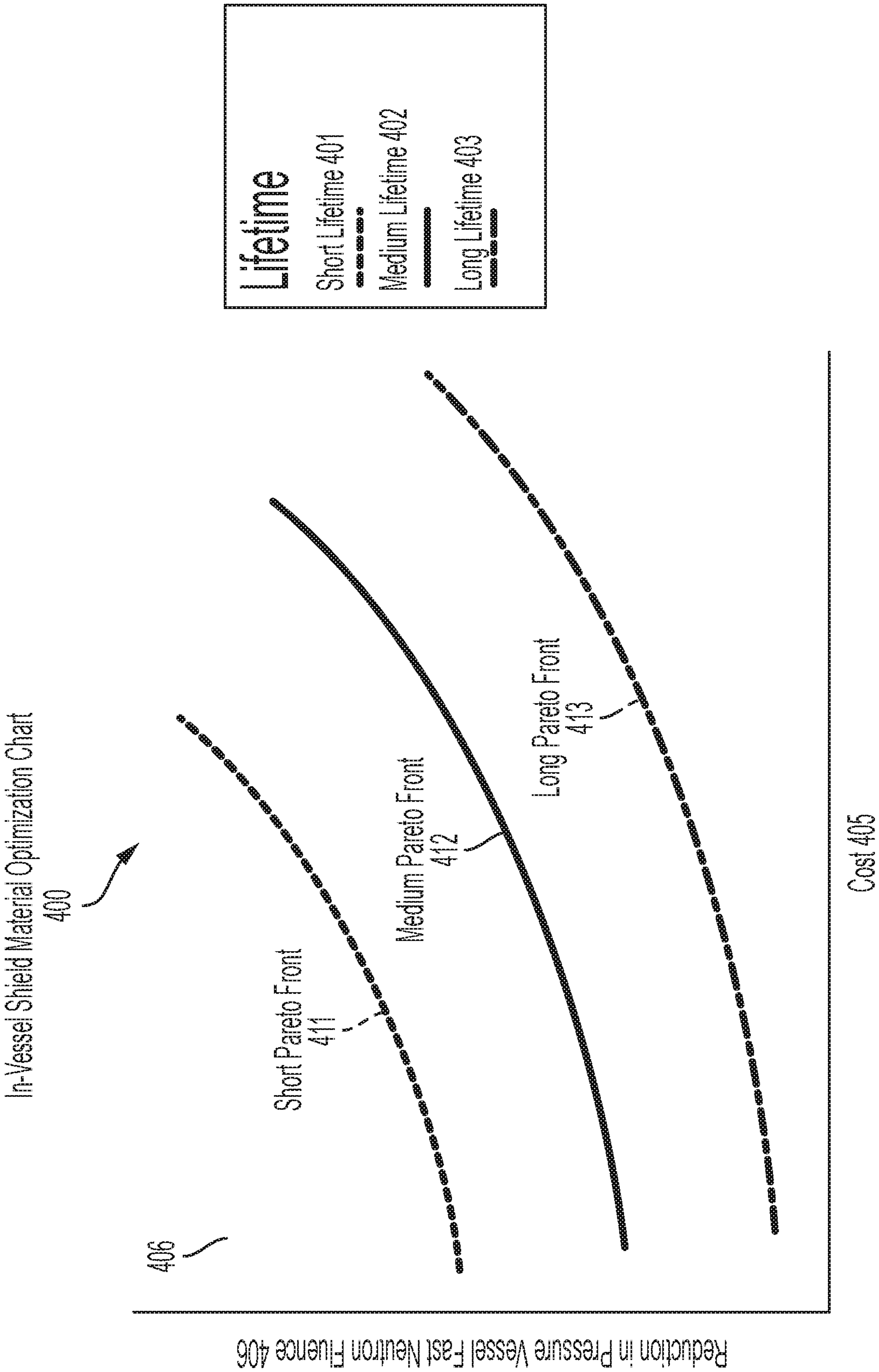

In-Vessel Shield Material Optimization Chart
400
Lifetime
Short Lifetime 401
Medium Lifetime 402
Long Lifetime 403
406
Short Pareto Front
411
Medium Pareto Front
412
Long Pareto Front
413
Cost 405
Reduction in Pressure Vessel Fast Neutron Fluence 406

FIG. 4

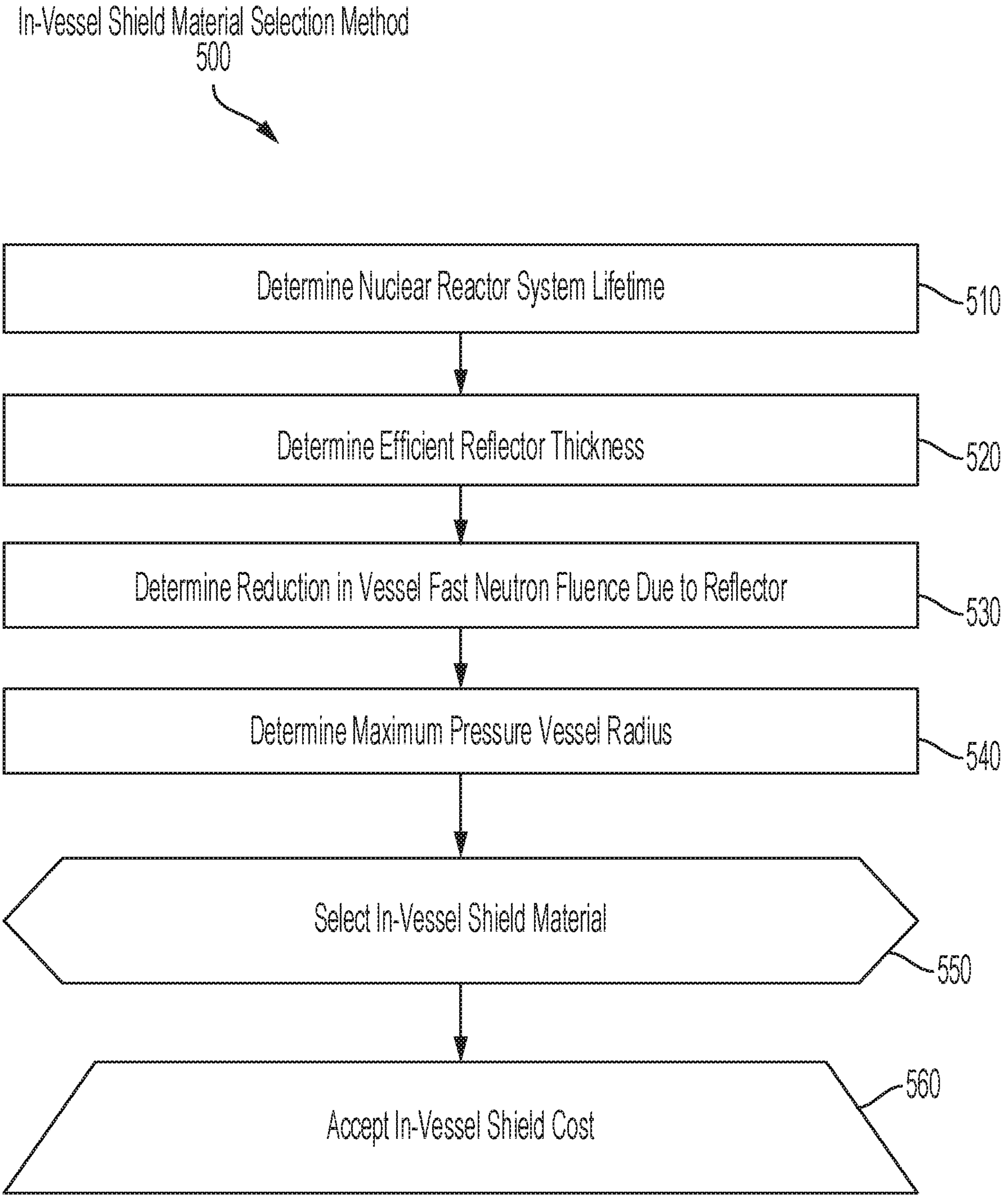

In-Vessel Shield Material Selection Method
500
Determine Nuclear Reactor System Lifetime
510
Determine Efficient Reflector Thickness
520
Determine Reduction in Vessel Fast Neutron Fluence Due to Reflector
530
Determine Maximum Pressure Vessel Radius
540
Select In-Vessel Shield Material
550
Accept In-Vessel Shield Cost
560

FIG. 5

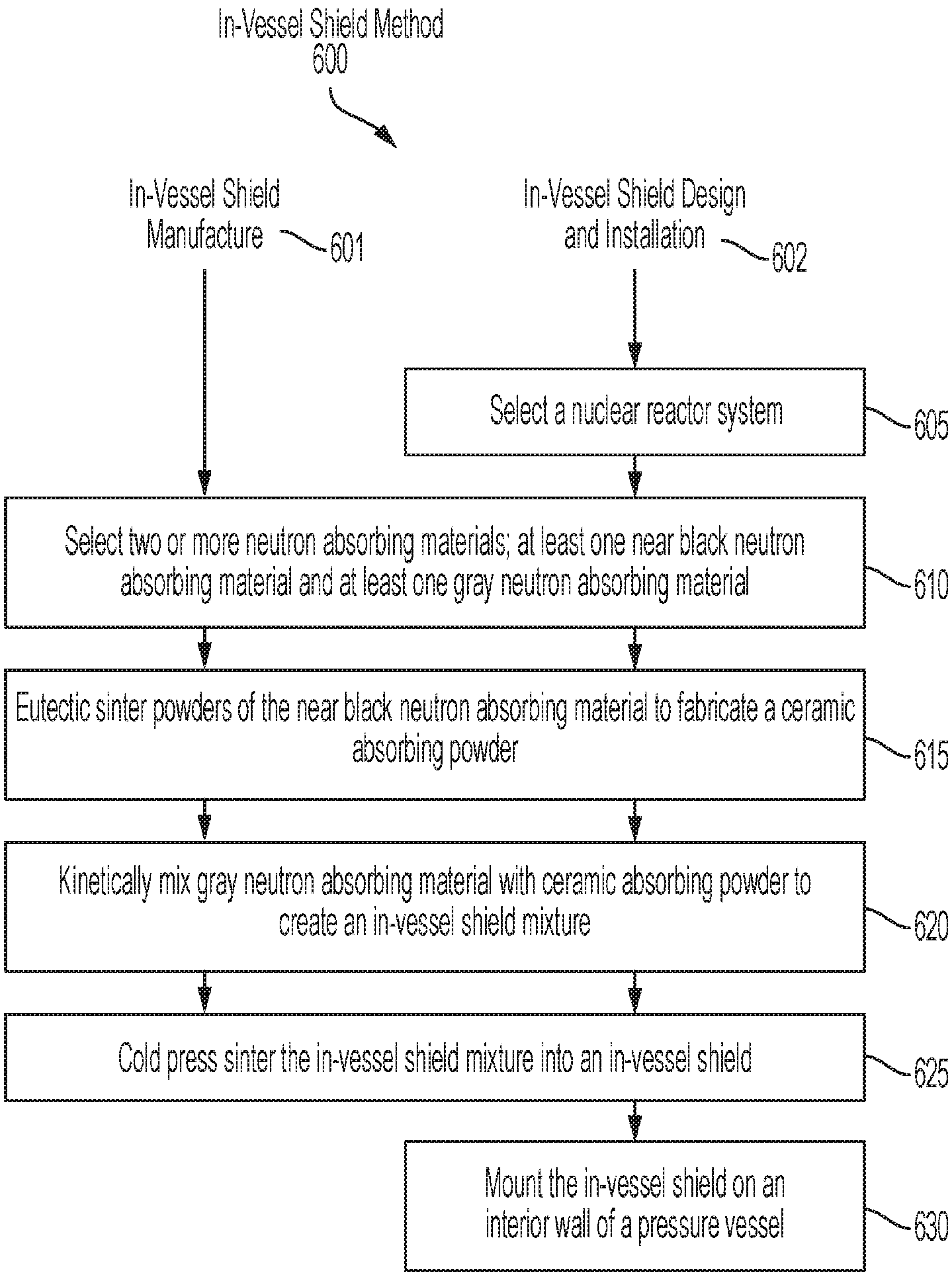

In-Vessel Shield Method
600
In-Vessel Shield
Manufacture
601
In-Vessel Shield Design
and Installation
602
Select a nuclear reactor system
605
Select two or more neutron absorbing materials; at least one near black neutron absorbing material and at least one gray neutron absorbing material
610
Eutectic sinter powders of the near black neutron absorbing material to fabricate a ceramic absorbing powder
615
Kinetically mix gray neutron absorbing material with ceramic absorbing powder to create an in-vessel shield mixture
620
Cold press sinter the in-vessel shield mixture into an in-vessel shield
625
Mount the in-vessel shield on an interior wall of a pressure vessel
630

FIG. 6

INTERNATIONAL SEARCH REPORT

International application No.
PCT/US 20/54188

A. CLASSIFICATION OF SUBJECT MATTER
IPC - G21C 11/06 (2021.01)
CPC - G21C 11/06, G21C 7/06, G21C 11/08, Y02E 30/30

According to International Patent Classification (IPC) or to both national classification and IPC

B. FIELDS SEARCHED

Minimum documentation searched (classification system followed by classification symbols)
See Search History document

Documentation searched other than minimum documentation to the extent that such documents are included in the fields searched
See Search History document

Electronic data base consulted during the international search (name of data base and, where practicable, search terms used)
See Search History document

C. DOCUMENTS CONSIDERED TO BE RELEVANT

Category*	Citation of document, with indication, where appropriate, of the relevant passages	Relevant to claim No.
Y --- A	US 3,296,083 A (HAAKE et al.) 03 January 1967 (03.01.1967), entire document especially Fig 2; col 1, lns 32-37; col 3, lns 29-32; col 10, lns 37-45	1, 14-18 --- 2-13, 19-40
Y --- A	US 2002/0185614 A1 (JOSEPH) 12 December 2002 (12.12.2002), entire document especially Abstract	1, 14-18 --- 2-13, 19-40
Y	US 2017/0040069 A1 (VENNERI) 09 February 2017 (09.02.2017), entire document especially paras [0050], [0053]-[0054]	14-18
A	US 4,320,204 A (WEAVER) 16 March 1982 (16.03.1982), entire document especially col 2, lns 22-27	19-40
A	YUN, "A Study on In-Vessel Shielding Design Concept for the Sodium-cooled Fast Burner Reactor." Proceedings of the Reactor Physics Asia 2015 (RPHA15) Conference, 18 September 2015 (18.09.2015) [onlin] < https://www.kns.org/files/int_paper/paper/RPHA_2015_5/RPHA15YunSH1.pdf >	1-40
A	GB 976,802 A (LICENTIA GMBH) 02 December 1964 (02.12.1964), entire document	1-40

☐ Further documents are listed in the continuation of Box C. ☐ See patent family annex.

* Special categories of cited documents:
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"L" document which may throw doubts on priority claim(s) or which is cited to establish the publication date of another citation or other special reason (as specified)
"O" document referring to an oral disclosure, use, exhibition or other means
"P" document published prior to the international filing date but later than the priority date claimed
"T" later document published after the international filing date or priority date and not in conflict with the application but cited to understand the principle or theory underlying the invention
"X" document of particular relevance; the claimed invention cannot be considered novel or cannot be considered to involve an inventive step when the document is taken alone
"Y" document of particular relevance; the claimed invention cannot be considered to involve an inventive step when the document is combined with one or more other such documents, such combination being obvious to a person skilled in the art
"&" document member of the same patent family

Date of the actual completion of the international search	Date of mailing of the international search report
19 January 2021	17 FEB 2021
Name and mailing address of the ISA/US Mail Stop PCT, Attn: ISA/US, Commissioner for Patents P.O. Box 1450, Alexandria, Virginia 22313-1450 Facsimile No. 571-273-8300	Authorized officer Lee Young Telephone No. PCT Helpdesk: 571-272-4300