GA–A27283

CONCEPTUAL DESIGN OF THE NGNP REACTOR SYSTEM

by

M. RICHARDS, A. BAXTER, C. ELLIS, O. GUTIERREZ, and J. CROZIER

MARCH 2012

GENERAL ATOMICS

GA–A27283

CONCEPTUAL DESIGN OF THE NGNP REACTOR SYSTEM

by

M. RICHARDS, A. BAXTER, C. ELLIS, O. GUTIERREZ, and J. CROZIER

This is a preprint of a paper to be presented at the 2012 20th International Conference on Nuclear Engineering, Anaheim, California, July 30, 2012 through August 3, 2012 and to be published in the *Proceedings.*

Work supported by
General Atomics Internal
Research and Development Funding

GENERAL ATOMICS PROJECT 04962
MARCH 2012

ABSTRACT

In May 2010, General Atomics (GA) was awarded a contract by the U.S. Department of Energy (DOE) under the Next Generation Nuclear Plant (NGNP) project to develop the conceptual design for a steam-cycle modular helium reactor (SC-MHR) demonstration plant. The SC-MHR is a graphite-moderated, helium-cooled reactor that is designed to produce steam for industrial applications and/or electricity production using a Rankine cycle. The SC-MHR operates with a thermal power level of 350 MW and a coolant outlet temperature of 725°C. This paper provides an overview of the conceptual design of the SC-MHR reactor system (RS), including assessments of core performance in the areas of reactor physics and power distributions, temperature/flow distributions, fuel integrity, and fission product release.

I. INTRODUCTION

The primary functions of the RS are to generate heat from fission energy, to transfer this heat to the helium coolant within the RS, and to control the neutron generation rate in the core [1]. The functions also include providing the first barrier to the release of radionuclides, providing sufficient reactivity control for shutdown assurance under licensing basis events, providing heat removal and storage during conduction cooldown events, and providing shielding of the reactor vessel (RV) from core radiation. Figure 1 shows the RS located within the vessel system. Figures 2 and 3 show isometric and plan views of the reactor core and internal components.

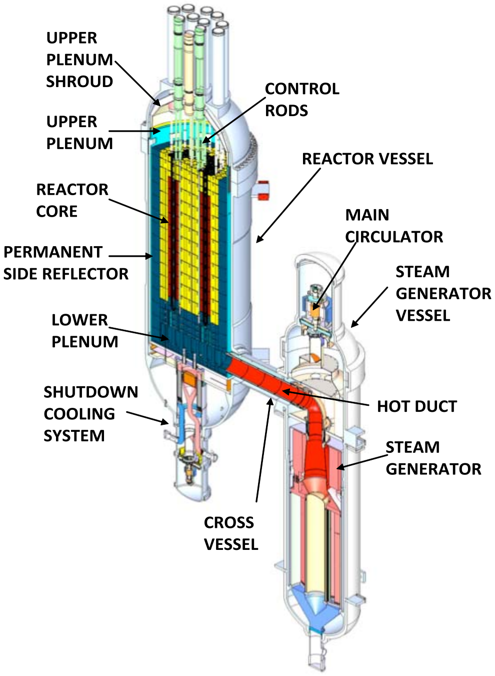


Fig. 1. Reactor system located within vessel system.

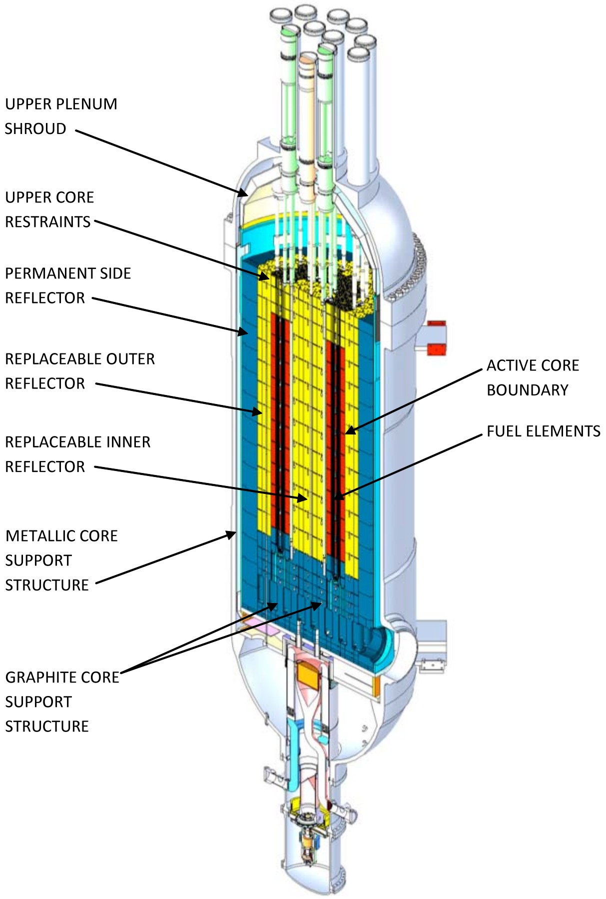


Fig. 2. Reactor core and internal components (isometric view).

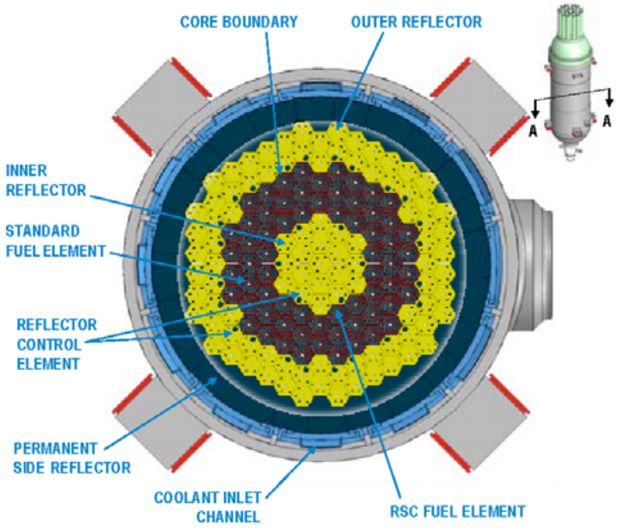


Fig. 3. Reactor core and internal components (plan view).

II. REACTOR SYSTEM COMPONENTS

The RS consists of reactor core components, reactor internals components, neutron control components, and reactor service equipment components. The reactor core components are located entirely within the RV and its top penetrations. The reactor internal components are located entirely in the RV. The neutron control components are located within the RV, within the top and bottom penetrations and within penetrations in the concrete cavity wall of the building structure.

The reactor core components include the hexagonal fuel elements, hexagonal graphite reflector elements, startup sources, and reactivity control material. The reactor core components, together with elements of the reactor internal components, constitute a graphite assembly which is supported on the graphite core support assembly and restrained by the metallic core support assembly, as shown in Figs. 2 and 3. The hexagonal fuel elements are stacked in columns that form an active core annulus with columns of hexagonal graphite reflector elements in the central region (inner or central reflector) and surrounding the active core (outer or side reflector), as shown in Fig. 3. The core produces a power of 350 MW(t) at a nominal power density of 5.9 MW/m^3.

Placed on top of the upper graphite reflector is the upper core restraint (UCR) assembly. The UCR elements are inter-keyed and keyed to the core barrel to provide lateral stability to the core array. These elements also channel the coolant flow and contain radiation shielding material.

Above and beneath the active core are replaceable hexagonal graphite reflector elements which contain coolant passages to channel the flow from the UCR to the active core and from the active core to the graphite core support assembly. These reflector elements also contain radiation shielding material.

The reactor core components contain both fixed and movable poison for reactivity control. The fixed poison is in the form of fixed burnable poison (FBP) rods. The movable poison is in the form of metal clad control rods. In the event that the control rods become inoperable, a backup, reserve shutdown control material in the form of boronated pellets may be released into the core as a redundant, backup option for reactivity control. The operational mechanisms for the control rods and for the reserve shutdown control are part of the neutron control components.

The reactor internal components consist of the UCR elements, permanent side reflector (PSR) elements, graphite core support assembly, metallic core support assembly, and the upper plenum shroud (UPS), as shown Fig. 2. The graphite core support assembly and the metallic core support assembly together comprise the core support structure. The metallic core support assembly includes the core barrel, core support floor, seismic keys/keyways, and coolant riser channels. The core support floor is located beneath the graphite core support assembly.

The primary coolant is circulated by the heat transport system (HTS) through the RS to the steam generator. The reactor fuel generates heat by nuclear fission and transfers this heat to the helium primary coolant. The reactor internal components and neutron control components work in conjunction with the reactor core components to control the neutron generation rate in the reactor core. The reactor core components provide the first barrier in containing the radionuclides produced by fission. The reactor internal components together with the reactor core components shield the RV from core radiation.

The neutron control components consist of outer neutron control assemblies (ONCAs), inner neutron control assemblies (INCAs), in-core flux mapping units (IFMUs), source range detector assemblies, and ex-vessel neutron detectors.

A. Fuel Elements

There are 660 fuel elements, with 540 standard elements and 120 elements that contain a channel for reserve shutdown control (RSC) material. All fuel elements are manufactured from nuclear-grade graphite and are right hexagonal prisms 793 mm (31.2 in.) high and 360 mm (14.2 in.) across the flats. Fuel and coolant holes are parallel through the length of the fuel element, in a regular pattern of two fuel holes per coolant hole (except near the boundaries). The standard fuel element, shown in Fig. 4, contains a continuous pattern of fuel and coolant holes except for a central fuel handling hole surrounded by smaller coolant holes, and the corner fuel holes in which the fuel is replaced with FBP. The reserve shutdown fuel elements differ in that they contain a 95.2 mm (3.75 in.) diameter channel for reserve shutdown control material. This channel replaces 22 fuel and 12 coolant holes. The pitch of the coolant and fuel hole array is 18.8 mm (0.74 in.). At each top-to-bottom, element-to-element interface in a column there is a dowel/socket connection system which provides alignment for refueling, alignment of coolant channels, and which transfers seismic loads on core elements. A handling hole, located at the center of the element, extends down about one-third of the height, with a ledge where the grapple of the fuel handling machine engages.

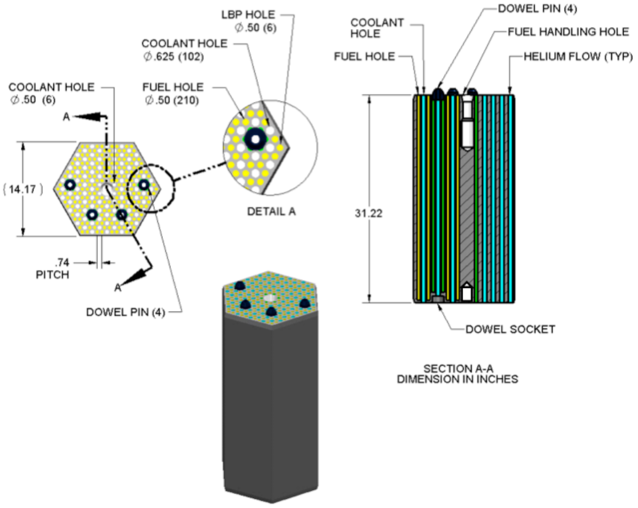


Fig. 4. Standard fuel element.

The fuel holes contain fuel compacts, which have dimensions of 12.45 mm (0.49 in.) in diameter 49.3 mm (1.94 in.) in length. The fuel compacts contain the fuel particles and are formed through a process that includes applying an overcoat of carbonaceous matrix material to the coated fuel particle. The fuel compacts are stacked in each of the fuel element fuel holes. Each stack contains 15 fuel compacts except for the six stacks under each of the four dowels which contain 14 fuel compacts. Figure 5 shows an isometric view of a fuel element containing fuel compacts. A nominal radial gap of 0.13 mm (0.005 in.) exists between the fuel compact and the fuel hole to allow for fuel element assembly and to preclude interference between the fuel compact and the graphite body during operation. Graphite plugs are cemented into the tops of the fuel compact holes to enclose the fuel compact stacks. A gap exists between the top of the fuel compact stack and the graphite plug, also to preclude interference during operation.

A key design feature of the SC-MHR that allows for high-temperature operation is the use of ceramic, coated particle fuel. A coated fuel particle consists of a microsphere ("kernel") of nuclear fuel (usually in the form of an oxide, carbide, or oxycarbide) that is coated with multiple layers of pyrolytic carbon and silicon carbide. The buffer, inner pyrolytic carbon (IPyC), silicon carbide (SiC), and outer pyrolytic carbon (OPyC) layers are referred to collectively as a TRISO coating. The coating system can be viewed as a miniature pressure vessel that provides containment of radionuclides and gases. Figure 6 shows a typical coated particle fuel design and describes the functions of the fuel kernel and the coating layers. Design parameters for NGNP Low-Enriched Uranium (LEU) coated particle fuel are given in Table I.

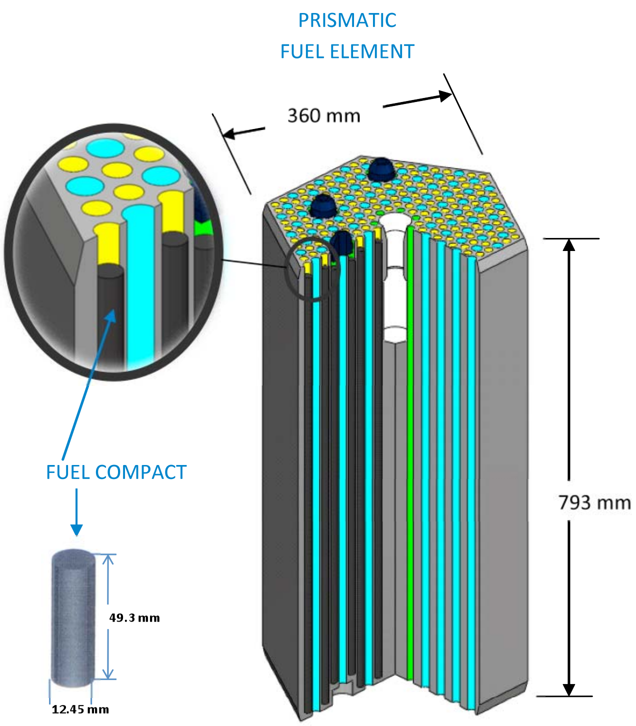
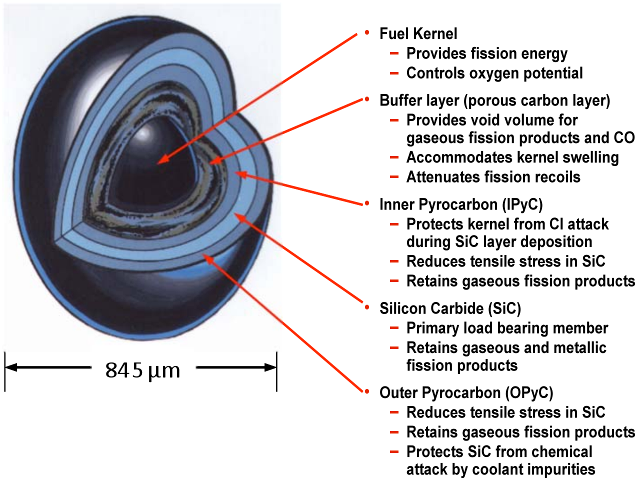


Fig. 5. Isometric view of fuel element containing fuel compacts.



- Fuel Kernel
  - Provides fission energy
  - Controls oxygen potential
- Buffer layer (porous carbon layer)
  - Provides void volume for gaseous fission products and CO
  - Accommodates kernel swelling
  - Attenuates fission recoils
- Inner Pyrocarbon (IPyC)
  - Protects kernel from Cl attack during SiC layer deposition
  - Reduces tensile stress in SiC
  - Retains gaseous fission products
- Silicon Carbide (SiC)
  - Primary load bearing member
  - Retains gaseous and metallic fission products
- Outer Pyrocarbon (OPyC)
  - Reduces tensile stress in SiC
  - Retains gaseous fission products
  - Protects SiC from chemical attack by coolant impurities

Fig. 6. Coated particle fuel design.

TABLE I. Coated particle fuel design parameters.

Composition	$UC_{0.5}O_{1.5}$
Uranium Enrichment (weight % U-235)	15.5
Kernel Diameter (μm)	425
Kernel Density (g/cm^3)	10.5
Coating Thickness (μm)	
Buffer	100
IPyC	35
SiC	35
OPyC	40
Particle Diameter (μm)	845
Density (g/cm^3)	
Buffer	1.0
IPyC	1.9
SiC	3.2
OPyC	1.9

B. Replaceable Reflector Elements

The replaceable reflector elements are also manufactured from nuclear-grade graphite and have a similar size, shape, and handling hole to the fuel elements (except that some are half-height or three-quarter height). These elements are located above, below, and outside the inner and outer boundaries of the active core.

The outer reflector includes two rows of hexagonal reflector columns, as shown in Fig. 3. The outer row elements are solid, with the exception of the fuel handling hole. In addition, 24 elements in the inner row also have a control rod channel as shown in Fig. 7. The inner reflector also includes six columns next to the active core that contain a control rod channel of the same size as in the outer reflector control columns.

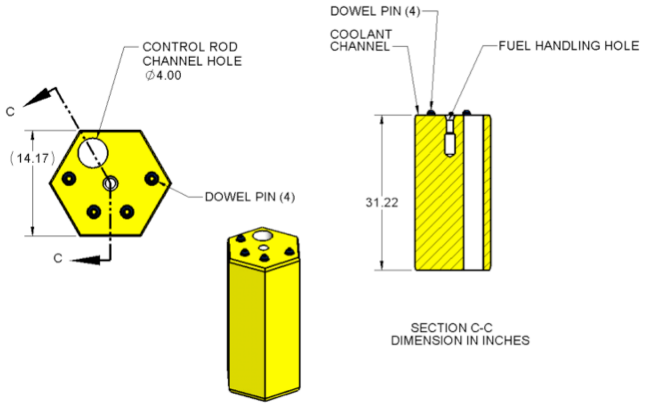


Fig. 7. Side reflector element with control rod channel.

C. Upper Core Restraint

The UCR is an assembly of hexagonal and irregularly-shaped metallic elements located on top of the core and PSR. There are five types of UCR elements, depending on location. The UCR elements are made of Alloy 800H. The UCR elements above fuel columns have vertical channels to direct coolant through the coolant channels in the top reflector. The UCR elements above RSC fuel columns also contain a channel for insertion of the RSC material, as shown in Fig. 8. Similar channels are also provided for the UCR elements above columns for control rod insertion and insertion of IFMUs.

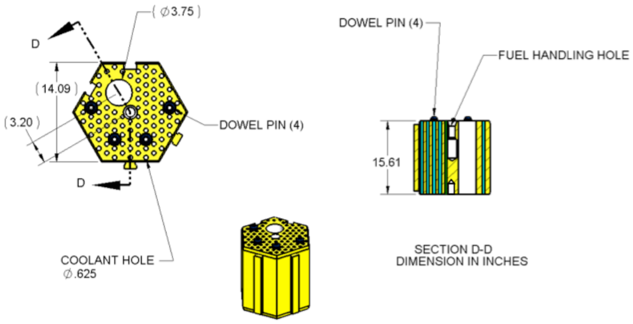


Fig. 8. UCR element above RSC fuel column.

All UCR elements are connected together with mating vertical T-slots and T-bars on their sides. The closely-fitted stems of the T-connections act as shear keys and keyways to limit relative motion of the UCR elements within the UCR assembly. Similar connections between the PSR UCR elements and the core barrel limit translation and rotation of the UCR assembly with respect to the metallic core support assembly. The heads of the T connections between UCR elements prevent the remaining elements from separating when elements are removed during refueling. The connecting T-bars and T-slots are vertically elongated to accommodate differential vertical movements between elements and the core barrel.

D. Permanent Side Reflector

The PSRs are bounded between two horizontal planes as shown in Fig. 2. The top plane is the bottom surface of the UCR elements and the bottom plane is the top of the ceramic insulating elements stacked on the metallic core support floor. The PSRs are bounded by two vertical concentric boundaries, as indicated in Fig. 3. The outer boundary is the inside surface of the core barrel and the inner boundary is the vertical projection of the outer surfaces of the outer (replaceable) reflector elements. The two vertical boundaries form a ring of PSR elements that encircle the core. There is one hole through the PSR at the location of the hot duct. This hole channels the flow from the core outlet plenum to the hot duct entrance (see Fig. 9).

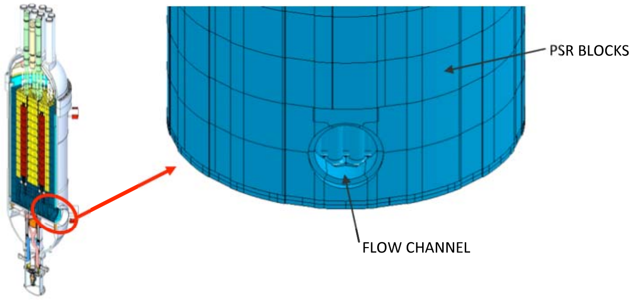


Fig. 9. PSR structure at hot duct entrance.

E. Graphite Core Support Assembly

The graphite core support assembly consists of (1) flow distribution elements, (2) core support elements, (3) inner reflector support elements, (4) outer reflector support elements, (5) core support pedestals, (6) lower plenum floor elements, and (7) ceramic insulation elements. Figure 10 shows an isometric view of the graphite core support assembly.

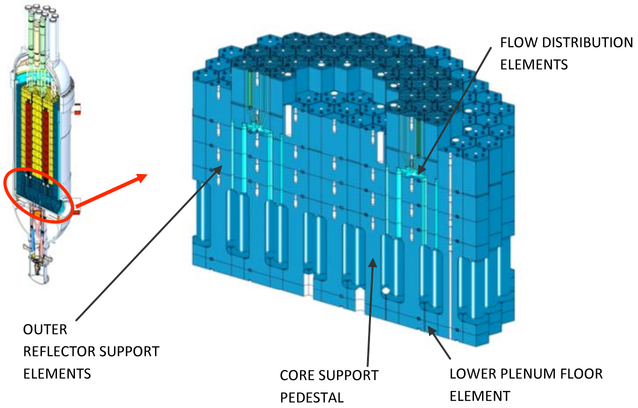


Fig. 10. Graphite core support assembly (isometric view).

F. Metallic Core Support Assembly

The metallic core support assembly is a prefabricated weldment made up of a floor section and a barrel section. The floor consists of a bottom plate and a top plate connected by radial webs, and, at the outer edge, by a cylindrical shell which is continuous with the core barrel above it. This construction maximizes the bending and shear strength and stiffness of the metallic core support assembly. The barrel is a cylindrical shell continuous with the floor. A cylindrical skirt welded to the bottom of the floor houses the top of the shutdown cooling heat exchanger. The metallic core support assembly design is illustrated in Figs. 11 and 12.

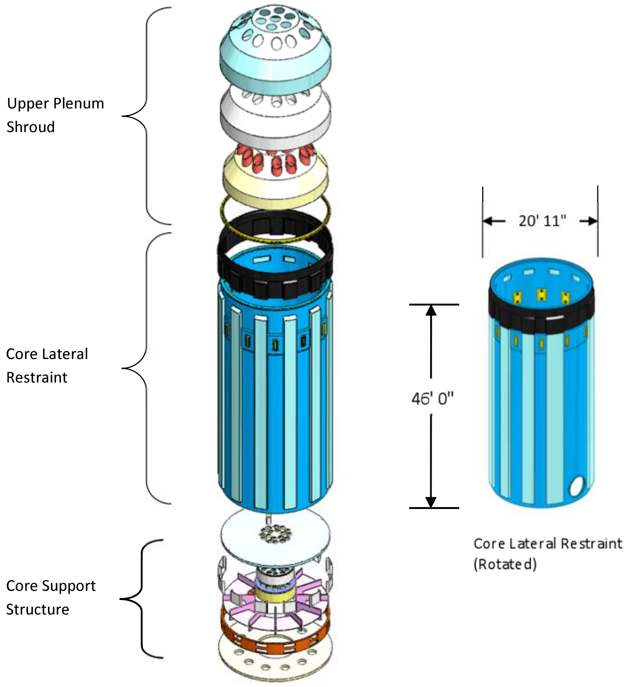


Fig. 11. Metallic core support assembly (isometric/exploded view).

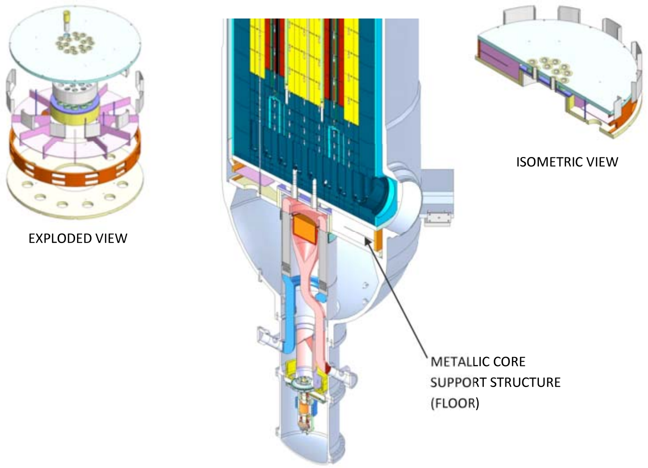


Fig. 12. Metallic core support floor.

G. Upper Plenum Shroud

The UPS (see Fig. 13) forms the top part of the core inlet plenum. It rests on, and is attached and sealed to, the top of the core barrel. The UPS is a continuous, dome-shaped, prefabricated weldment composed of an inner shell and an outer shell connected by radial and loop stiffeners. Nineteen vertical penetrations provide access and sealing surfaces for the removable control rod, reserve shutdown, and IFMU assemblies.

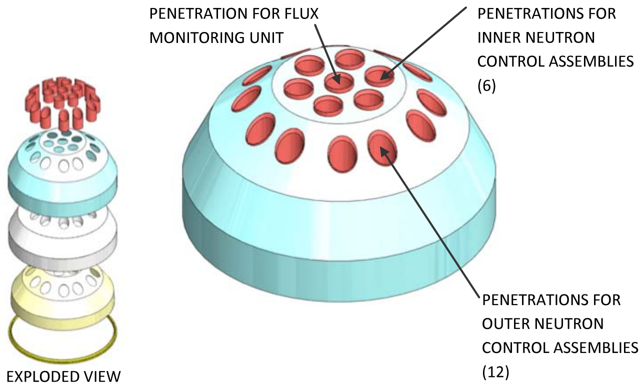


Fig. 13. Upper plenum shroud.

The cavities between the inner and outer shells of the UPS contain shielding material (boronated graphite) for neutron flux attenuation. These cavities are vented to accommodate primary coolant pressure changes without inducing differential pressure loads on the UPS cavities.

A thermal barrier, consisting of fibrous ceramic insulation blankets with metallic cover plates and retainers, cover the outside of the UPS to protect the head of the RV from excessive temperatures during conduction cooldown events.

H. Neutron Control Assemblies

Figure 14 shows the INCAs and ONCAs installed within the reactor vessel. Each ONCA is equipped with two independent control rod and drive units. These assemblies are interchangeable with each other in any of the assigned penetrations. Each INCA is equipped with one control rod and drive unit and two independent sets of reserve shutdown control equipment. These assemblies are also interchangeable with each other in any of the assigned penetrations. Figure 15 shows an overall view of an INCA. Two sets of reserve shutdown control equipment are mounted in each INCA package. Each set consists of a reserve shutdown hopper which contains the boronated graphite shutdown material, the fuse link mechanism, which opens the hopper gate by means of the actuation rod, and the reserve shutdown material gate. The reserve shutdown guide tube, provided with the INCA structural equipment, guides the reserve shutdown material into a special channel in the core. Figure 16 shows the arrangement of the reserve shutdown control equipment within the INCA package.

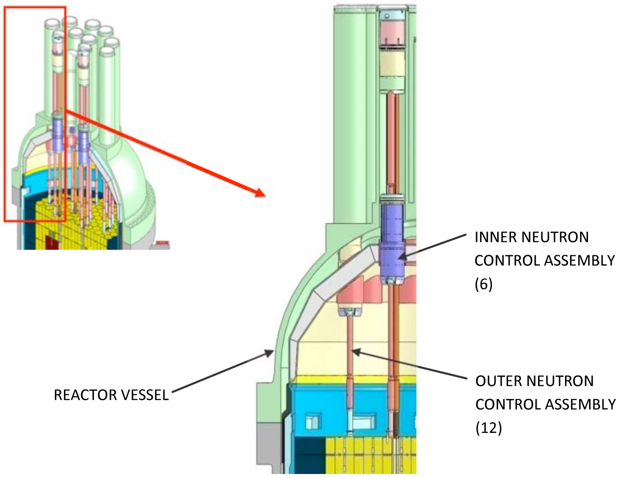


Fig. 14. ONCAs and INCAs installed within the RV.

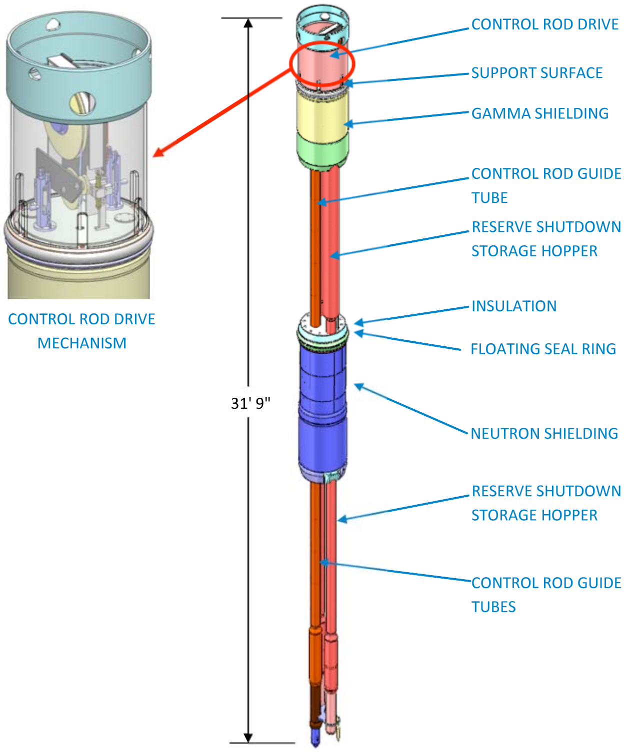


Fig. 15. Inner neutron control assembly.

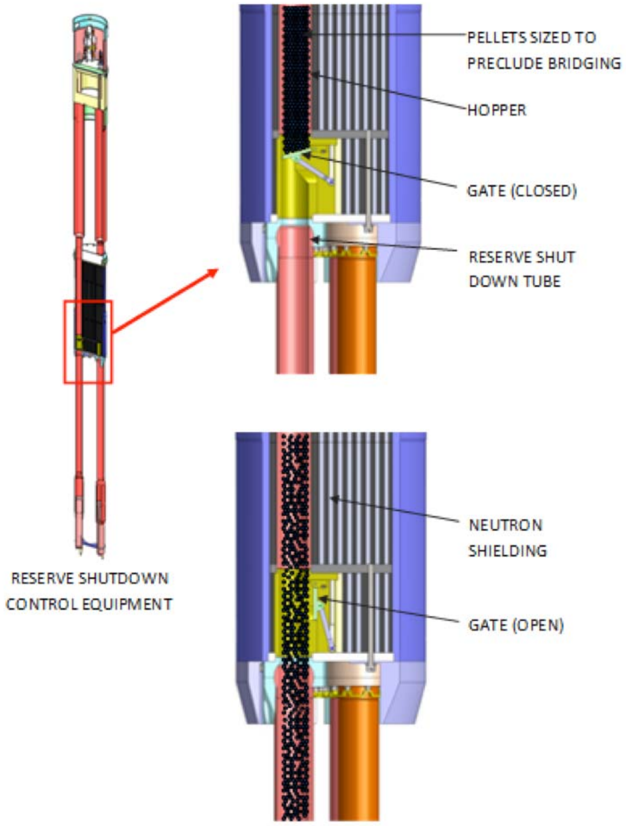


Fig. 16. Reserve shutdown control equipment in an INCA.

III. NUCLEAR FEATURES

The annular core configuration was selected, in combination with the power density of 5.9 MW/m^3, to achieve a maximum power rating and still permit passive core heat removal while maintaining the peak fuel temperature at less than 1600°C during a depressurized conduction cooldown event, which ensures acceptable fuel performance. The active core outer diameter of 3.51 m is sized to maintain a minimum reflector thickness of 1.0 m. An inner core effective diameter of 1.65 m was selected on the basis of studies conducted on the reactivity worth of control rods with annular cores and on passive heat removal studies to limit fuel element temperatures in heatup events. The reactor core height was established based on axial power stability studies. To meet the projected reactivity control requirements using reflector control rods, the annular width of the core can be no greater than 1.0 m. The core reactivity is controlled by a combination of movable poison (control rods), FBP, and a negative temperature coefficient.

The current zoning scheme consists of three radial and three axial zones. The three axial zones consist of 5, 3, and 2 fuel elements in the top, middle, and bottom zones, respectively. The three radial zones correspond to the three annular rings of fuel elements, i.e., 18, 24, and 24 fuel elements per ring, as seen in Fig 3. These zoning schemes result in power distributions that maintain the maximum time-averaged fuel temperature at less than 1,250°C, which ensures acceptable fuel performance.

The equilibrium fuel cycle length is 530 effective full power days (EFPDs). The fuel cycle uses two reload segments, so that 33 of the 66 core columns (half of the core) are replaced each reload. The control rod pattern (6 control rods in the inner reflector and 24 control rods in the outer reflector) allows refueling by 1/6 core sectors. The refueling operation is performed one sector at a time. Each refueling sector thus contains 11 core columns, with 10 fuel elements per column. In refueling each of the six refueling core sectors, the sector elements are removed one layer at a time until all of the core elements in the sector have been removed. Fresh reload elements and one-cycle-old elements are then reloaded into the sector. Thus, all 660 core elements are removed from the reactor each reload, and the 330 two-cycle-old elements are replaced with 330 fresh reload elements.

IV. THERMAL-FLUIDS FEATURES

The reactor core is designed to limit temperatures and temperature distributions in the core components to ensure their integrity, and to limit core exit coolant temperature streaks to ensure the integrity of downstream components (hot duct and steam generator). Specifically, in the core, fuel temperatures must be controlled for acceptable fuel performance and circulating activity levels, while fuel element temperatures and temperature distributions must be controlled for acceptable graphite stresses. Control rod metallic cladding temperatures must be limited to protect the control rods. In addition, the core must be designed to prevent unacceptable flow-induced vibration or thermal/flow-induced temperature fluctuations. To accomplish these objectives requires close coordination of the nuclear and mechanical design efforts, since the nuclear design dictates the core power distributions and the core mechanical design dictates the core flow distributions.

A. Flow Driven by Heat Transport System

The core inlet coolant temperature is 290°C and the average core exit coolant temperature is 725°C. The coolant enters the reactor internal components from the annular space between the outer, structural part of the hot duct and the cross vessel. From there it continues around the core support floor, flowing into a series of passages in the vertical wall at the outside edge of the core support floor. It then is directed radially inward and radially back outward by a baffle separator, to cool the core support floor. After cooling the core support floor, the coolant enters the rectangular channels attached to the outside of the core barrel. These channels direct the flow up to the core upper plenum.

From the core upper plenum, the coolant flow traverses the core in the coolant channels and the control rod channels. The graphite core support assembly channels these flows to the lower plenum. Coolant flows from adjacent fuel columns merge and are mixed in the corner flow passages in the post block prior to exiting into the lower plenum.

Most of the total core cavity flow takes this path. The remaining flow bypasses the core in the gaps between the fuel and reflector columns. These gaps have been kept as small as possible to minimize this flow, but large enough to permit refueling.

All of these flows merge in the lower plenum. They traverse the plenum, flowing around the core support pedestals, and then pass through a hole in the PSR and core barrel boss before entering the hot duct assembly.

B. Flow Driven by Shutdown Cooling System (SCS)

The coolant flow path is different when the SCS is in operation. The major differences are (1) the helium coolant enters the metallic core support floor via holes located in the bottom plate of the floor, and (2) the heated coolant in the core lower plenum exits to the SCS through holes in graphite core support assembly and the metallic core support floor.

C. Conduction Cooldown

During conduction cooldown events, neither the HTS nor the SCS is operational. In this condition, the heat is transferred from the reactor core to the Reactor Cavity Cooling System (RCCS) by a combination of conduction, natural convection, and radiation.

When the RV remains pressurized, natural convection plays a major role in removing heat from the active core. During this event the mixed mean temperature of the gas in the upper plenum reaches 760°C. The UPS and its penetrations are designed to limit the temperature of the RV to within its design temperature allowables. The UPS achieves this by insulating the RV from direct contact with the upper plenum helium. In addition, the UPS and the top of the core barrel have a thermal barrier over their outer surface to limit the amount of heat that can be transferred to the RV.

D. Core Flow Distribution

The reactor power is removed by the downward flow of pressurized helium. Flow enters the core fuel columns through the coolant channels in the metallic UCR elements. From the UCR elements, the coolant flow continues through the top reflector, the fuel elements in the active core, and the bottom reflector. Most of the coolant flows down the coolant channels in the fuel elements. In addition to the flow in the coolant channels in the fuel elements, some flow is directed to the control rod channels, and there is some flow in the vertical gaps between the columns in the active core and in the side reflector. Additionally, a small amount of flow exchange occurs between the flows in the gaps between columns and the coolant and control rod channel flows, through small horizontal gaps between stacked elements. This flow is called crossflow.

Flow enters the control rod channels via small holes in the control rod guide tubes in the core upper plenum, traverses the core in the control rod channels and exits to the core support element corner channels via small channels. The entrance holes and exit channels for the control rod channels are sized to produce the desired flow in the control rod channels and to produce about equal pressure drop at the entrance and exit to the control rod channels. This design minimizes the flow through unrodded control rod channels and minimizes the coolant flow velocities in the rodded channels. Further, by setting the entrance and exit flow resistances to produce about equal pressure drops, the pressure differences between this flow and the gap flows are minimized. This helps to minimize crossflow between the gap and the control rod channel flows.

All of the coolant flow that bypasses the coolant channels is called bypass flow. The reactor core is designed to achieve a nominal bypass flow fraction, averaged over the length of the active core, of $\leq 15\%$. This bypass flow includes approximately 1.5% of the total core flow that is allocated for control rod cooling. The total pressure drop from the upper plenum to the lower plenum is approximately 50 kPa. Most of this pressure drop results from the wall friction associated with the coolant channels.

E. Core Stability

The UCR elements are keyed together, which is similar to the design of the region constraint devices used in the Ft. St. Vrain reactor. This design maintains the core array during refueling, and provides a stabilizing effect against fluctuations and flow-induced vibrations during operation. Given the low pressure drop and low flow velocities in the SC-MHR reactor module, it is expected these measures will be adequate to prevent core fluctuations and flow-induced vibrations. More detailed analyses will be performed to confirm core stability and to determine if confirmation by testing is required.

V. STRUCTURAL FEATURES

A primary function of the metallic core support assembly is to restrain the PSR laterally and thus maintain the core geometry. This is achieved by the core barrel, which is connected to the vessel by seismic keys/keyways. These keys are designed to permit relative expansion between the core barrel and RV while still maintaining accurate location and lateral restraint of the barrel within the vessel.

The hot duct is attached to the core barrel. This attachment transfers some of the weight and seismic loads of the hot duct to the core barrel which is, in turn, supported by the metallic core support floor. The graphite core support assembly transfers the vertical loads from the reactor core and PSR to the metallic core support assembly. The metallic core support floor, in turn, transfers these loads together with the vertical loads of the core barrel and UPS to the RV ring forging. These vertical loads include deadweight, pressure drop of the primary coolant, and seismic loads.

Alignment of the elements within the columns is provided by dowels and sockets. These dowel/socket combinations are designed to withstand the highest shear forces generated during seismic conditions to maintain alignment for coolant flow and for control rod insertion.

Fast neutron fluence will cause the graphite elements to shrink slightly. Since the flux field is nonuniform, the elements will also distort. However, the distorted elements will not interfere with reactor operation or with refueling. The nonuniform flux fields in combination with the temperature gradients will also cause stresses in the graphite elements. These thermal/irradiation induced stresses are added to the stresses from other sources (of which seismic loads are the most important), and the total stresses are generally kept at levels below half the strength of the material to preclude functional failures and to avoid unsafe conditions.

VI. CORE PERFORMANCE ASSESSMENT

The core performance assessment includes: (1) core physics analyses to determine fuel cycle length, power distributions, and reactivity coefficients; (2) thermal/fluids analyses to determine fuel/graphite/coolant temperatures and flow distributions; and (3) fuel performance and fission product release analyses. The computer code sequence used for the core performance assessment is shown in Fig. 17.

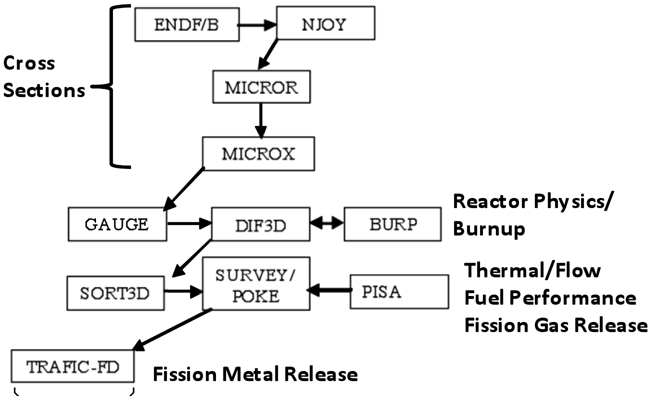


Fig. 17. Computer code sequence for core performance analysis.

Figure 18 shows the one-third symmetry layout used for core performance assessments. Figure 19 shows a typical equilibrium power distribution in the fuel columns averaged over their axial lengths (referred to as a radial power distribution) and Fig. 20 shows the corresponding axial power profile.

The calculation of fuel temperatures from the core power distribution and coolant flow distribution is of fundamental importance to predicting fuel performance and fission product release. The fuel temperatures at a given location in the core result from the following temperature rises: (1) the temperature rise in the coolant from the sensible heat added as the coolant flow transverses the core; (2) the film temperature rise from the flowing helium coolant to the surface of the coolant channels in the graphite fuel blocks (forced convection); (3) the temperature rise across the graphite web separating the coolant holes from the adjacent fuel holes (conduction); (4) the temperature rise across the small gap between the fuel hole and fuel compacts (conduction and radiation), and (5) the temperature rise from the fuel compact surface to the compact centerline (conduction).

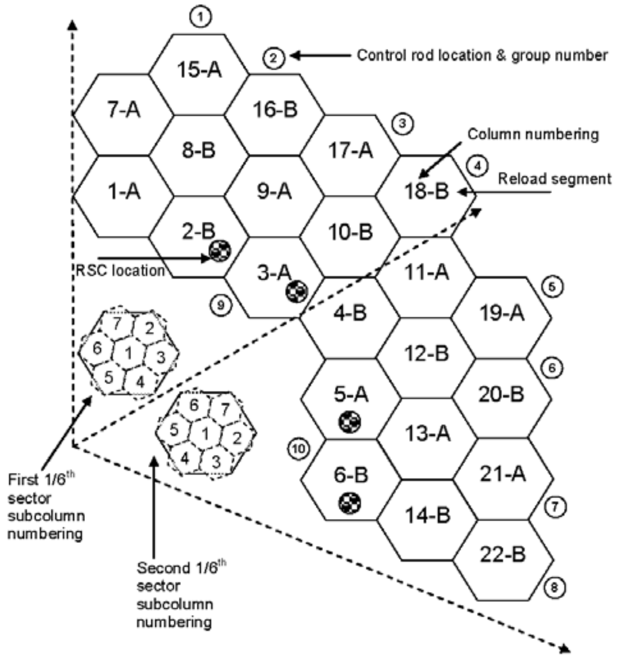


Fig. 18. One-third symmetry layout used for core performance assessment.

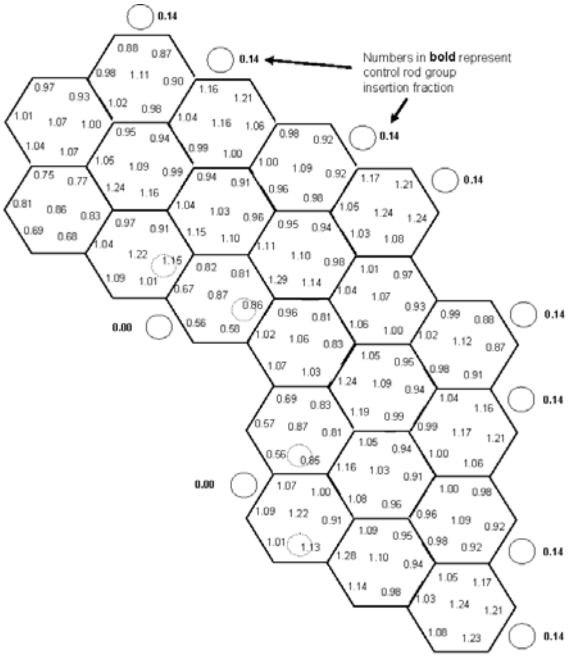


Fig. 19. Radial power distribution (equilibrium cycle).

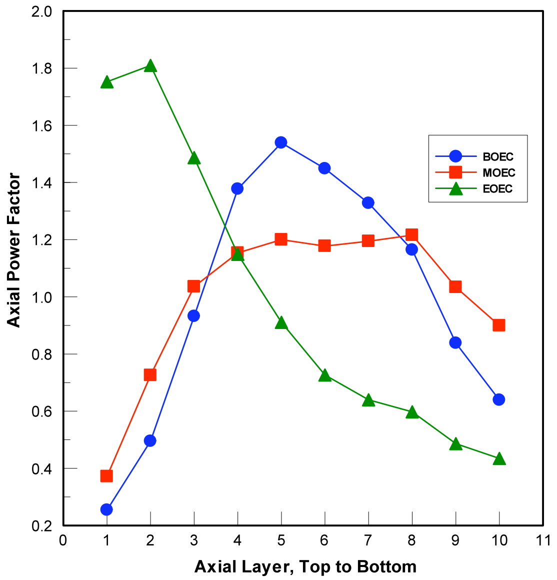


Fig. 20. Axial power distribution (equilibrium cycle).

Figure 21 shows a typical axial temperature distribution for an average power location. Figure 22 shows the volume distribution of fuel temperature averaged over the residence time for an equilibrium cycle. Only a small fraction of the core (<10%) experiences time-averaged temperatures above 1000°C.

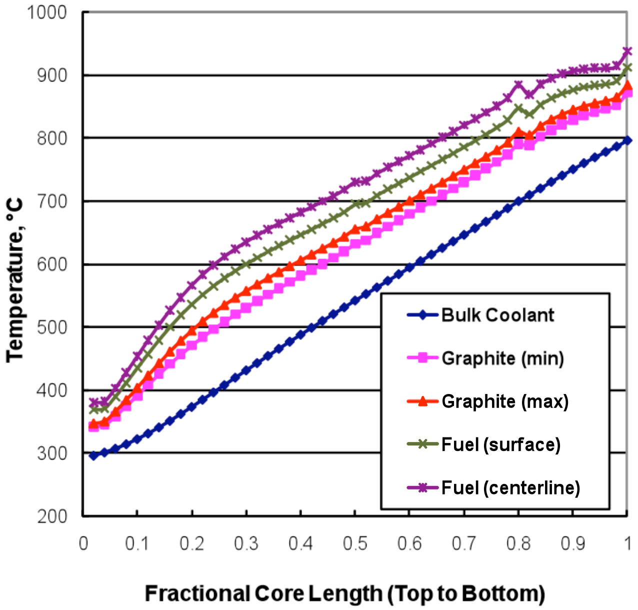


Fig. 21. Typical axial fuel temperature distribution (average power location).

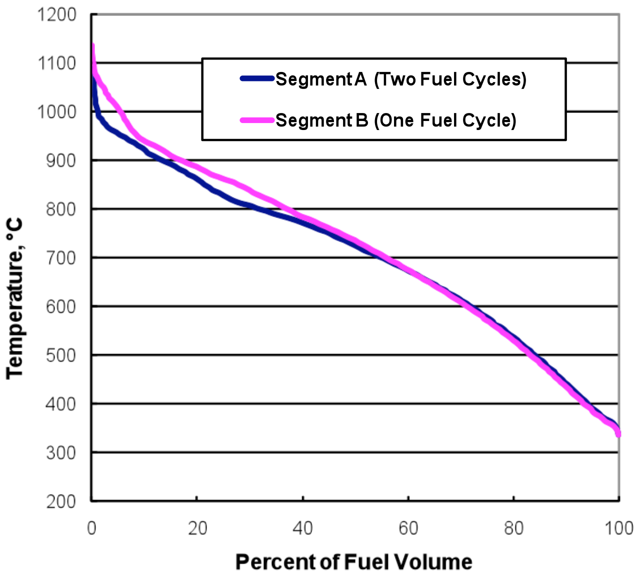


Fig. 22. Volume distribution of fuel temperature averaged over residence time.

As shown schematically in Fig. 23, the five principal release barriers in the radionuclide containment system are: (1) the fuel kernel, (2) the particle coatings (particularly the SiC coating), (3) the fuel element structural graphite, (4) the primary coolant pressure boundary, and (5) the vented, low-pressure containment building. The effectiveness of these individual barriers in containing radionuclides depends upon a number of fundamental factors including the chemistry and half-lives of the various radionuclides, the service conditions, and irradiation effects. The effectiveness of these release barriers is also event specific.

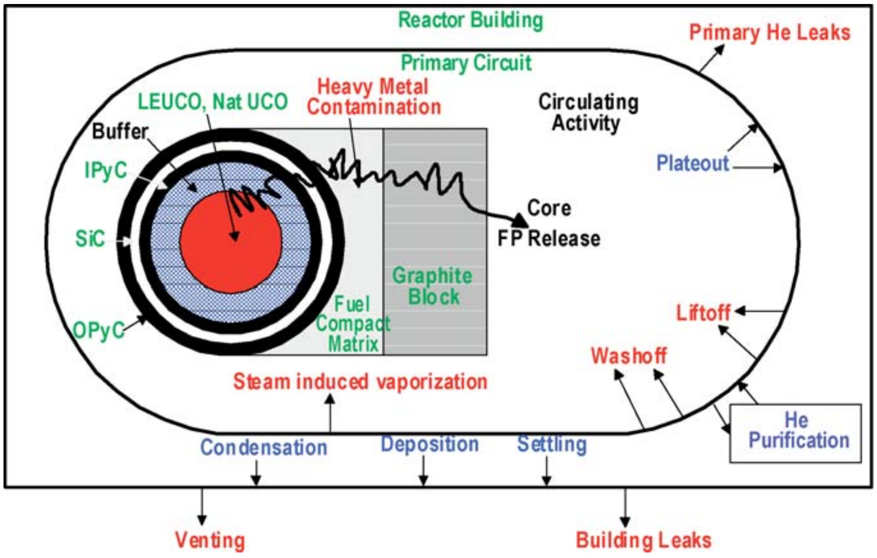


Fig. 23. Radionuclide containment system.

Standard GA design practice is to define a two-tier set of radionuclide design criteria, referred to as “Maximum Expected” and “Design” criteria. The “Design” criteria are derived from externally imposed requirements, such as site-boundary dose limits, occupational exposure limits, etc.; in principle, any of these radionuclide control requirements could be the most constraining for a given reactor design. Once the “Design” criteria have been derived from the radionuclide control requirements, the corresponding “Maximum Expected,” criteria are derived by dividing the “Design” criteria by an uncertainty factor, or design margin, to account for uncertainties in the design methods. This uncertainty factor is typically a factor of 4 for the release of fission gases from the core and a factor of 10 for the release of fission metals. The fuel and core are to be designed such that there is at least a 50% probability that the fission product release will be less than the “Maximum Expected” criteria and at least a 95% probability that the release will be less than the “Design” criteria.

As part of the design process, performance requirements must be derived for each of the five release barriers described above. Of these barriers, the particle coatings are the most important. Moreover, the in-reactor performance characteristics of coated-particle fuel are strongly influenced by its as-manufactured attributes. Consequently, the fuel performance requirements and fuel quality requirements must be systematically defined and controlled.

The core performance assessment is a set of best-estimate calculations for which the “maximum expected” requirements are applicable. Table II shows results for in-service fuel failure, fission-gas release, and fission-metal release at the beginning, middle, and end of an equilibrium fuel cycle (BOEC, MOEC, and EOEC). Additional design optimization, along with further assessments of the fuel requirements, is planned for the preliminary/final design phases.

TABLE II. Comparison of core performance results
with provisional NGNP requirements.

Parameter	**Maximum Expected Limit**	**BOEC**	**MOEC**	**EOEC**
In-service fuel failure (exposed kernel fraction)	≤ 5.0E-5	2.0E-6	3.3E-6	5.0E-6
Kr-88 fractional release	≤ 8.3E-7	3.5E-7	3.4E-7	3.9E-7
I-131 fractional release	≤ 2.0E-6	1.3E-6	1.3E-6	1.4E-6
Cs-137 fractional release	≤ 7.0E-6	4.9E-6	4.3E-6	4.0E-6
Ag-110m fractional release	≤ 2.0E-4	9.3E-5	5.7E-5	3.7E-5

VII. CONCLUSIONS

The conceptual design of the SC-MHR demonstration plant has been completed as part of the NGNP project. The RS design is based largely on previous high-temperature gas-cooled reactor design and operational experience. Additional optimization of the design is planned during the preliminary and final design phases.

The SC-MHR provides an outlet helium temperature of 725°C to generate steam at temperatures up to 585°C and pressures up to 16.5 MPa. Applications include electricity production with ~ 40% thermal efficiency and process steam for the petrochemical industry. Commercialization of the SC-MHR can (1) provide energy security and diversity, (2) effect a major reduction in carbon emissions, (3) create price stability for process heat, (4) preserve natural gas for use as a raw material, and (5) re-energize the nuclear industry and grow industry-related jobs.

ACKNOWLEDGMENT

This work was supported by internal research and development funding provided by General Atomics.

REFERENCES

[1] M. Richards, *et al.*, 2010, "Reactor System Design Description," Next Generation Nuclear Plant (NGNP) Prismatic HTGR Conceptual Design Project, General Atomics Report NGNP-S00218, Rev. 0, General Atomics, San Diego, CA, December 2010.