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# DESIGN CHARACTERISTICS OF THE IDAHO NATIONAL LABORATORY HIGH-TEMPERATURE GAS-COOLED TEST REACTOR

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# DESIGN CHARACTERISTICS OF THE IDAHO NATIONAL LABORATORY HIGH-TEMPERATURE GAS-COOLED TEST REACTOR

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**Abstract.** *A point design for a graphite-moderated, high-temperature, gas-cooled test reactor (HTG-TR) has been developed by Idaho National Laboratory as part of a U.S. Department of Energy initiative to explore and potentially expand existing U.S. test reactor capabilities. This paper provides a summary of the design and its main attributes. The 200-MW HTG-TR is a thermal-neutron spectrum reactor composed of hexagonal prismatic fuel and graphite reflector blocks. The HTG-TR is designed to operate at 7 MPa with a coolant inlet/outlet temperature of 325/650°C and utilizes tristructural isotropic particle fuel with an enrichment of 15.5-wt% $^{235}$U. The primary mission of the HTG-TR is material irradiation, and therefore the core has been specifically designed and optimized to provide the highest possible thermal and fast-neutron fluxes. The highest thermal-neutron flux (3.90E+14 n/cm$^2$/s) occurs in the outer reflector, and the maximum fast-flux levels (1.17E+14 n/cm$^2$s) are produced in the central reflector column, where most of the graphite has been removed. The core features a large number of irradiation positions with large test volumes and long test lengths, providing an ideal environment for thermal-neutron irradiation of large test articles. The total available test volume is more than 1,100 L. Up to four test loop facilities can be accommodated with pressure tube boundaries to isolate test articles and test fluids (e.g., liquid metal, liquid salt, and light water) from the helium primary coolant system.*

## I. INTRODUCTION

A point design for a graphite-moderated, high-temperature, gas-cooled test reactor (HTG-TR) has been developed by Idaho National Laboratory (INL) as part of a U.S. Department of Energy (DOE) initiative to explore and potentially expand the existing U.S. test reactor capability. Although no high-temperature gas-cooled reactors (HTGRs) are operating today in the U.S., the design of the HTG-TR has leveraged design information and experience from both previously constructed and operated commercial U.S. HTGRs and more modern HTGR designs with annular cores.

In addition, the HTG-TR has drawn heavily on recent advancements in tristructural isotropic (TRISO) particle fuel, graphite, and in-core HTGR materials from the successful DOE Advanced Gas Reactor (AGR) Program and associated U.S. Nuclear Regulatory Commission interactions. These advancements, along with recent and past HTGR technology, have been incorporated into the design of the HTG-TR.

This paper provides an overview of the HTG-TR design objectives, its core characteristics, and the initial steady-state and transient results obtained with the Monte Carlo N-Particle (MCNP) and Reactor Excursion and Leak Analysis Program (RELAP) 5-3D simulation codes. A discussion on possible test loop design options is included here as well.

## II. HIGH-TEMPERATURE, GAS-COOLED TEST REACTOR OBJECTIVES

The primary objective of the HTG-TR design was to provide a versatile, multi-purpose, high-flux facility for irradiation of advanced reactor fuels and materials. Currently, such capability in the U.S. is provided mainly by the High-Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory and the Advanced Test Reactor (ATR) at INL. HFIR and ATR are light-water reactors that have provided more than 40 years of safe and reliable operations and irradiation services.

Table I provides a comparison of pertinent test positions and reactor data for HFIR, ATR, and the HTG-TR design. Flux levels in the HTG-TR are below those of HFIR and ATR but not substantially lower despite the large differences in core power density. The product of the flux and irradiation time and relatively large number of test positions and large test volumes available in the HTG-TR increase the usefulness of the HTG-TR relative to HFIR and ATR in terms of irradiation sample throughput. The main irradiation spaces are large enough to accommodate (in loops) full-length partial fuel assemblies from a light-water reactor, fast reactor, or fluoride salt-cooled reactor.

TABLE I. Comparison of irradiation characteristics of HFIR, ATR, and HTG-TR.

| Reactor | Test Position | Test Position Diameter (cm) | Test Position Length (cm) | Peak Thermal Flux (n/cm$^2$/s) | Peak Fast Flux (n/cm$^2$/s) | Core Power (MW) | Core Power Density (W/cm$^3$) | Cycle Length (days) |
|---|---|---|---|---|---|---|---|---|
| HFIR | Permanent beryllium reflector | 3.8–7.6 | 51 | 2-10E+14 | ≤1.5E+14 ($E_n$>0.111 MeV) | 85 | 1,251 | 23 |
| ATR | Flux trap | 13.3 | 122 | 4.4E+14 | 2.2E+14 ($E_n$>0.1 MeV) | 110 | 116 | 30–60 |
| HTGR-TR | Graphite reflector | ≤16.0 | 640 | 3.9E+14 | 1.2E+14 ($E_n$>0.18 MeV) | 200 | 23 | 110 |

Another very important and useful feature of the HTG-TR is the chemical compatibility with a wide variety of loop and target materials, including fuel, structural materials, and loop coolant fluids. The center loop can be filled with liquid salt (e.g., FLiBe), liquid metal (sodium), high-pressure and high-temperature light water or steam, or other coolant gases and is estimated to have small or minimal reactivity impact on the relatively large core.

Other useful features of the HTG-TR include the ability to generate electricity and produce isotopes. The electricity could be sold to a local utility for revenue, and any surplus could be supplied to the national laboratory reactor site. The production of commercial isotopes could generate substantial revenue by employing the huge “drop-in” test volume space available in the reflector regions. Other secondary missions, such as hydrogen production and process heat testing, may be the most important, especially for U.S. energy security research and development. Secondary heat-transfer loops could be connected via state-of-the-art heat exchangers to provide prototypical conditions for liquid salt and light water secondary loop coolants.

## III. HIGH-TEMPERATURE, GAS-COOLED TEST REACTOR POINT DESIGN DESCRIPTION

The point design effort has been focused on the core and reactor vessel behavior. Results of the reactor physics and core thermal-hydraulic evaluations are provided in this section.

### III.A. Reactor Fuel and Core Configuration

The HTG-TR point design uses TRISO particle fuel in the form of fuel compacts loaded into prismatic fuel blocks with both fuel and coolant channels. The prismatic fuel blocks are based on a General Atomics design[1] that was used in the Fort St. Vrain (FSV) reactor. This block design offers great flexibility in enrichment zoning, particle packing fraction (PF) zoning, burnable poison rod placement, and cooling. Fig. 1 shows a detailed model of the FSV fuel block used in the HTG-TR physics analysis.

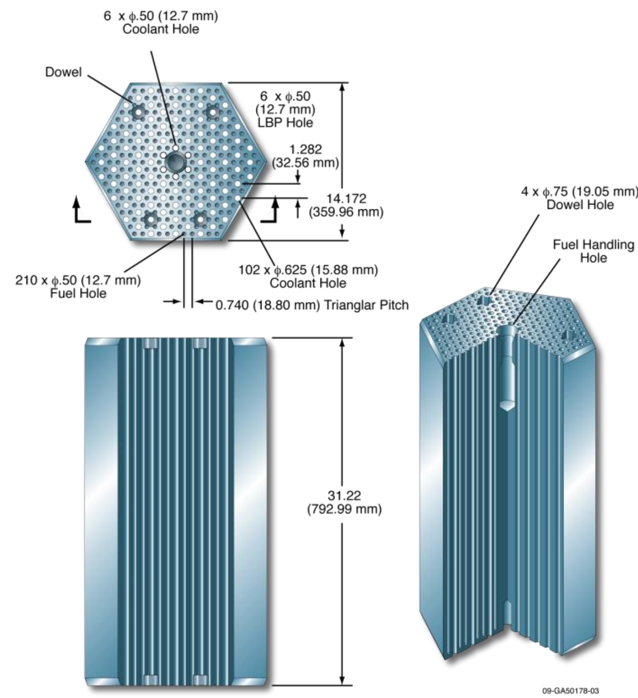


Fig. 1. FSV fuel block.

The TRISO particles matrixed in cylindrical fuel compacts form an integral high-temperature ceramic system specifically designed for the Next-Generation Nuclear Plant HTGR commercial reactors. The same TRISO fuel is used for the HTG-TR. Recent irradiation testing of the TRISO fuel in the DOE AGR Program has demonstrated the robustness and high performance of the fuel under high temperature (1,300°C), burnup (20% fissions of initial heavy metal atoms [FIMA]), and fast-fluence (5.5E+21 n/cm$^2$) conditions.

The specific TRISO particle design adopted for the HTG-TR will be based on the AGR-5/6/7 qualification test particle design that features a large 425-μm-diameter

UCO kernel, 15.5-wt% enrichment. The fuel compacts for the HTG-TR, however, have a much lower particle PF (PF=15%) to boost the irradiation fluxes, higher UCO density (11.04 versus 10.40 g/cm$^3$), higher graphite binder density (1.70 versus 1.2 g/cm$^3$), and a higher bulk graphite density (1.83 versus 1.74 g/cm$^3$).

The 200-MW core configuration (baseline) is shown in Fig. 2 and is similar in many respects to modern commercial HTGRs. The HTG-TR core configuration, however, diverges from the much larger commercial reactor in the number of fuel blocks and power as the mission changes to include the material irradiation. To boost irradiation flux in the outer reflectors where the irradiation test facilities are located, the HTG-TR core size is reduced to increase core power density (20–25 W/cm$^3$). Commercial HTGRs typically operate at much lower core power densities (6 to 8 W/cm$^3$).

Three of the six graphite block columns in Ring 3 contain control rods; the other three are irradiation test positions. These three test positions have the highest thermal flux in the core (3.90E+14 n/cm$^2$/s). The 18 columns of Ring 4 are all graphite block columns, 12 with control rods and the other six with additional irradiation test positions. Beyond Ring 5 is the permanent side reflector graphite blocks to form-fitted the core barrel.

Control rod and loop penetrations through the top head of the reactor pressure vessel may compete for the limited room available in the head region. An engineering assessment of the number, location, and diameters of tube penetrations will need to be part of the conceptual design phase. The current design, with its compact core configuration, specifically located the control rods in the outer reflector to address this potential problem. There are total of 15 control rods in the outer reflector with enough negative reactivity to shut the core down under both hot and cold conditions, even if two or three rods are stuck out. The key reactor parameters are summarized in Table II.

TABLE II. Key reactor parameters.

| | |
|---|---|
| Reactor thermal power | 200 MW |
| Primary coolant | Helium gas |
| Primary coolant system pressure | 7.0 MPa |
| Core pressure drop for normal operation | 192 kPa |
| Primary coolant flow rate | 117.3 kg/s |
| Core inlet temperature | 325°C |
| Core outlet temperature | 650°C |
| Average core power density | 23.4 W/cm$^3$ |
| Power cycle length | 110 days |
| Reactor vessel internals material | Alloy 800H (control rod sheath); stainless-steel 316L (irradiation loop pressure tube); molybdenum, zirconium, titanium (irradiation tubes in outer reflector) |
| Control rod material | $B_4C$ in graphite; $B_{10}$ enrichment 30 to 50% |
| Vessel material | Steel |
| Core fueled/total height | 6.4 m/9.2 m |
| Core outer diameter | 3.4 m |

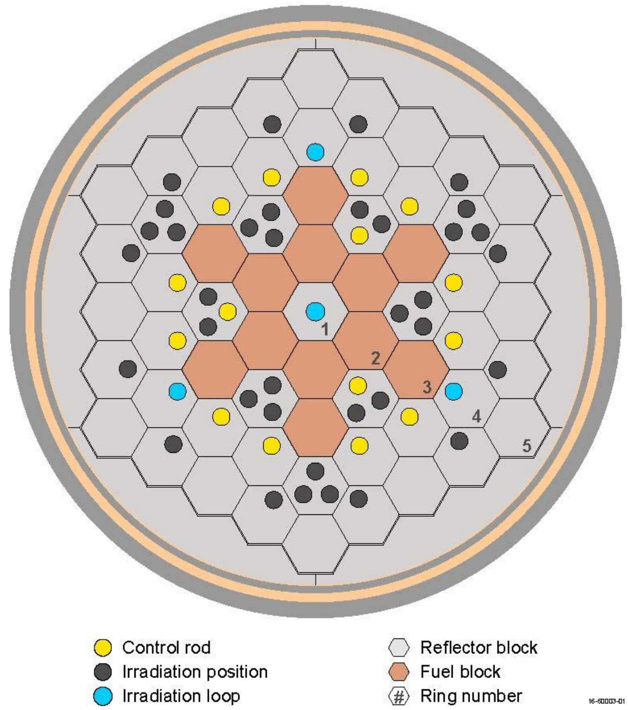


Fig. 2. Baseline test reactor core configuration.

### III.B. Testing Facilities

Test articles to be irradiated are mounted inside capsules held in test trains suspended directly in reflector holes (black circles in Fig. 2) or in independently cooled test loops (blue circles). Test trains in the reflector holes are cooled by the primary helium coolant, although some temperature control can be achieved by carefully designing the capsule's insulating layers to control the removal of gamma or fission heat generated within. In the configuration proposed for this study, there are 30 such test spaces, but there is sufficient reflector volume to add more.

The test loops remain outside the primary pressure boundary and have their own cooling systems; thus, the loops can provide prototypic conditions for testing fuels and materials for essentially any reactor concept. Test wells suspended from the top vessel head extend the primary pressure boundary downward into the reactor core,[2] as shown in Fig. 3.

The loops are inserted into the test wells with the test coolant inlet and outlet as well as the instrumentation leads emerging from the top of the wells. Piping and associated components (e.g., pumps and heaters) are outside the pressure vessel. Helium flow would be provided between the experiment loop and the well to provide cooling to the well wall. Loop coolant conditions are specified by the experimenter. It is expected that one of these loop locations in the outer reflector would contain a pneumatically driven rabbit system.

Because helium is chemically inert, leakage between the primary and loop coolants would not lead to chemical interactions between them. Nonetheless, such leakage is all but ruled out by the test well walls, which are designed to meet ASME Section III Class 8 pressure vessel criteria.

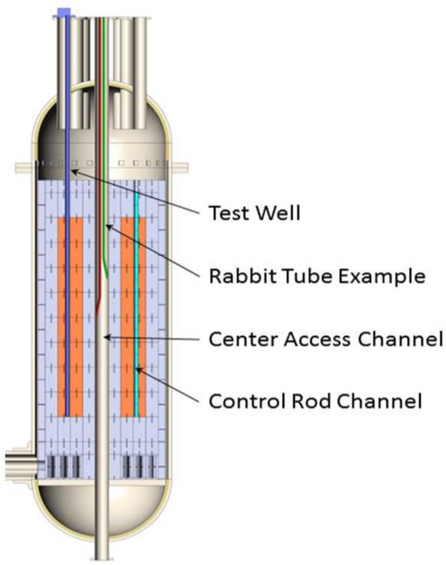


Fig. 3. Side view of vessel showing vertical test and control structures.[2]

The independent cooling loop can keep the test article at temperatures desired by the experimenter, but the primary coolant surrounding the test well will be within a temperature range (325 to 650$^{o}$C) that can assist in maintaining prototypical and bounding conditions suitable for testing fuels and materials under a wide range of temperatures. In the configuration proposed for this study, there are four such test loops, but more can be added in the reflector region.

These loops can accommodate relatively large test specimens cooled by various fluids, including high-pressure light water, low-pressure liquid salt, liquid metal, or different gases (e.g., helium). The loops each vary in test volumes between 14 and 30 L, resulting in a total available test volume of 1,136 L.

## IV. REACTOR PHYSICS

The proposed HTG-TR design shown in Fig. 2 represents an initial optimization and an evolved design derived from coupled-physics and thermal-hydraulic evaluations and based on results from five different core configurations. The five core configurations considered annular core configurations of 6, 7, 12, or 18 fuel columns in three rings for compactness. Allowing fuel columns in Ring 4 would have required an additional outer reflector ring of 30 graphite columns and increased the pressure vessel diameter by 0.72 m. Because the top priority for the physics evaluations was the maximization of the thermal flux in the inner and outer reflector block test positions, keeping the annular core as small as possible to boost core power density was the main focus. Higher power density translates into higher fluxes, and a smaller core with fewer fuel blocks meant fewer fuel blocks to reload each cycle.

To achieve the goal of the highest possible thermal-neutron irradiation flux, several variables needed to be maximized or minimized. These included maximization of the total core power, the minimization of the particle PF, and the reduction of the number of fuel blocks in the core through a reduced number of fuel columns and/or by a reduced height of the fuel columns (number of stacked fuel blocks). Arrangement of the maximum number of fuel blocks around a reflector block with an irradiation position enhanced the local thermal flux. All of these factors helped increase the core-power density and thermal-neutron irradiation fluxes.

Core-power density had a limit, however. Excessive power densities stress the TRISO particle fuel through excessive power output (>400 mW per particle) and time at temperature (>1,250°C). High power density also leads to excessive $^{235}$U fuel burnup rates and shorter power cycle lengths. The HTG-TR design attempted to maximize the power density while observing the fuel and the temperature limitations and cycle length goals.

The neutronic calculations used the MCNP 5 Version 1.60 computer code[3] and INL-developed depletion methods and software. Detailed MCNP core models were developed based on the General Atomics FSV fuel block design and the baseline core configuration depicted in Fig. 2. The calculated results are specifically for the core configuration at 200 MW, particle PF=15%, and eight-block-high fuel columns with no burnable poisons, enrichment grading, PF grading, or control rod insertion (except in the section on control rod worth). Fully explicit, 3D core models were constructed for the core physics calculations, which included a 1/12-core model, as shown in Fig. 4.

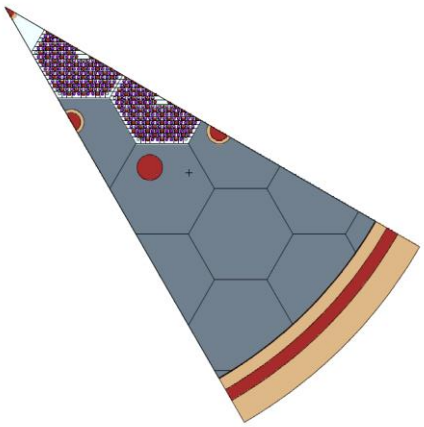

Fig. 4. One-twelfth-core MCNP model.

The maximum-thermal and fast-neutron fluxes calculated for the unrodded core occur above core midplane at the fifth fuel block level due to the axial temperature gradient in the core. The top of the core is cooler than the bottom. Although the highest fast flux occurs in the core fuel blocks, the high fuel block temperatures (800–1,000ºC) prevent the use of irradiation test facilities (tubes) and control rods (sleeves) with metallic components in the fuel blocks. Rather, all irradiation test positions are located in the inner and outer graphite reflector blocks, where the reflector blocks are much cooler (500–600°C) and experiments can be directly cooled by primary helium coolant.

Maximum thermal and fast fluxes are presented in Table III for three irradiation positions. The center loop position is a graphite block column with a centrally located, thick-walled, steel pressure tube. The Ring 3 irradiation positions are those three high-flux irradiation positions up against the Ring 2 fuel blocks (see Fig. 2). The Ring 4 positions consist of three irradiation positions and three loop positions up against the Ring 3 fuel blocks. The maximum thermal flux occurs in the Ring 3 positions and is calculated to be 3.90E+14 n/cm$^2$/s. These high thermal flux test positions could have a thin-walled, low thermal-neutron-absorbing containment tube for “drop-in” capsule experiments.

Table III. Maximum fast and thermal irradiation fluxes by position.

| Irradiation Position | Core Ring | Maximum Thermal Flux (n/cm$^2$/s) | Maximum Fast Flux (n/cm$^2$/s) |
|---|---|---|---|
| Center loop | 1 | 1.61E+14 | 1.17E+14 |
| Outer reflector | 3 | 3.90E+14 | 5.24E+13 |
| Outer reflector | 4 | 2.82E+14 | 2.28E+13 |

The cycle length for the baseline HTG-TR is calculated to be 110 days. Assuming a 4-week shutdown time between cycles, the HTG-TR has a maximum availability factor of 80%. The fuel rod average burnup ranges from 4.62 to 9.56% FIMA with a core average of 7.36% FIMA. The 110-day cycle length could potentially be extended by increasing the PF. A penalty will be paid in lower thermal-neutron irradiation fluxes by factors of 1.33 and 1.74, respectively, for PF=25% or 35% (Fig. 5). The cycle lengths, however, can be substantially extended to 210 and 281 days, respectively (Fig. 6). Variable cycle length through changes in PF could be a useful feature of the HTG-TR. Average compact burnups will also increase to approximately 8.85 and 9.26% FIMA for PF=25 and 35%, respectively.

A preliminary control rod design consists of $B_4C$ compacts in an 800H alloy sleeve. $^{10}$B enrichment of 30 to 50% would be sufficient. A total of 15 control rods are located in the outer graphite reflector block—three control rods in Ring 3 and 12 control rods in Ring 4. The total worth of the 15 rods is $50.2; hot shutdown requires $30.8, and hot-to-cold shutdown requires $35.0. Cold shutdown can be achieved with two out of the three Ring 3 rods and ten out of the twelve Ring 4 rods, showing sufficient shutdown margin for stuck rods or accidental rod withdrawals.

An important mission of the HTG-TR is the irradiation of a variety of primary coolant fluids from alternative reactor technologies. Alternative loop fluids may include light water, liquid salt, liquid metal, and gases or steam. The reactivity impact of 38-L volumes of pressurized light water, FLiBe, and liquid sodium were evaluated separately based on assumed placement in the central irradiation loop facility. The negative core reactivity incurred for each fluid inserted in place of helium coolant ranged from -0.12 $ to -1.57 $ (i.e., minor reactivity impacts to the overall core reactivity).

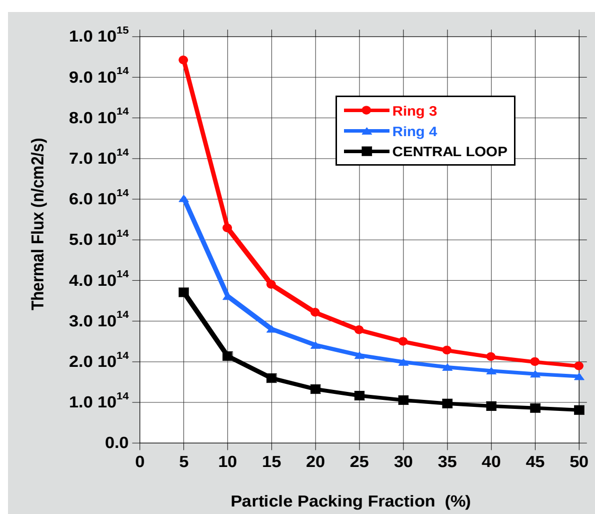


Fig. 5. Thermal flux versus packing fraction.

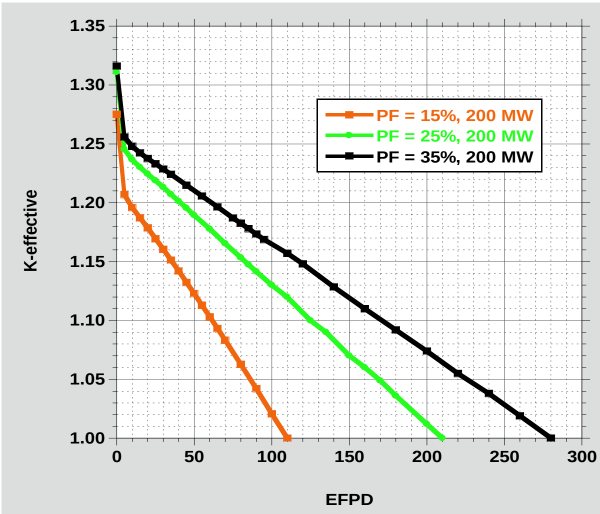


Fig. 6. Reactivity letdown versus burnup.

## V. CORE THERMAL HYDRAULICS

Steady-state calculations were performed using the RELAP5-3D computer code[4] to characterize the core and reactor vessel temperatures. Nominal and sensitivity cases were studied. The RELAP5-3D nodalization diagram is shown in Fig. 7. Coolant flow enters near the bottom of the reactor vessel cylinder (Component #100), flows up through the annulus between the core barrel and reactor vessel (#110), and then enters the upper plenum (#125). Helium then flows down through a number of parallel channels in the core: the coolant holes in the fuel blocks (#140 and #150), the gaps between the hexagonal blocks (#135, #145, #155, #165, #175, and #185), the gap between the permanent side reflector and the core barrel (#190), and gaps between the graphite reflector blocks and the control rods or irradiation tubes. These flow paths all meet in the lower plenum (#195), from which the coolant exits the reactor vessel (#200).

Steady-state calculations were performed for a range of reactor powers; block-to-block gaps of 2, 3, and 4 mm; and 4, 6, and 8 axial levels of fuel blocks. A 2-mm gap between blocks is about as close together as they can be loaded in the core. Through thermal cycling and irradiation, the gaps are expected to widen over the core life. Therefore, block-to-block gap widths of 2, 3, and 4 mm were modeled to provide an indication of how the response might change during the core life. To be consistent with keeping the peak fuel temperatures below 1,300°C during steady state, the nominal power levels are 100 MW for the four-level core, 150 MW for the six-level core, and 200 MW for the eight-level core.

Fig. 8 shows peak fuel temperatures from these calculations. At a constant power, the peak fuel temperature may increase by about 100°C over the life of the reflector as the gap between the blocks increases; this suggests that higher powers and fluxes could be tolerated earlier in the core life, when the gaps are smaller.

Calculated steady-state, thermal-hydraulic conditions are provided in Table IV for the 2-mm gap case. The effective core bypass is all of the flow that does not flow through either a fuel block coolant channel or a gap around a fuel block.

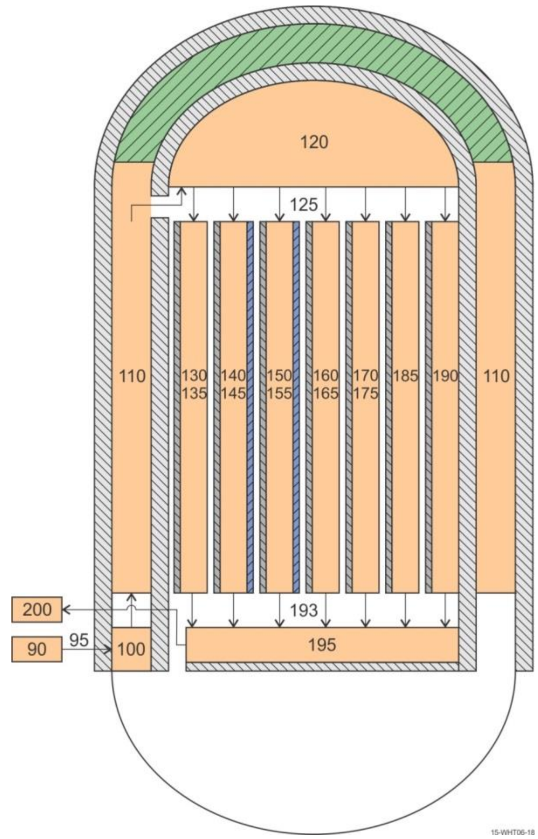


Fig. 7. Nodalization of the reactor vessel for the HTG-TR RELAP5-3D input model.

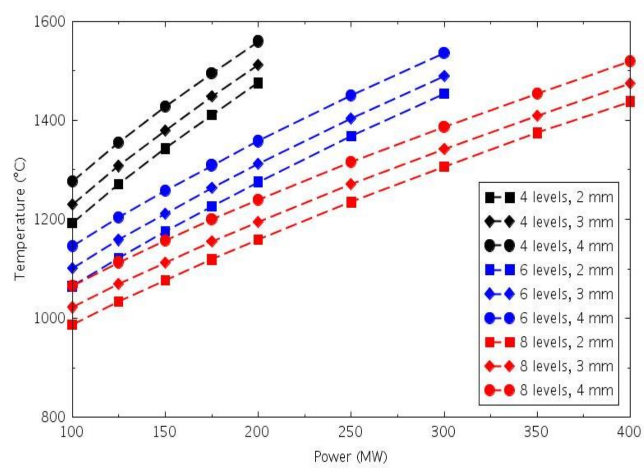


Fig. 8. Steady-state, peak-fuel temperature versus core power and block-to-block gap width.

Table IV. Steady-state conditions for eight-level, 200-MW core with 2-mm gaps between blocks.

| Parameter | Value |
|---|---|
| Coolant inlet temperature (°C) | 325 |
| Coolant outlet temperature (°C) | 650 |
| Coolant flow rate (kg/s) | 117.2 |
| Effective core bypass at core outlet (%) | 27 |
| Peak fuel temperature (°C) | 1,159 |
| Center reflector peak temperature (°C) | 648 |
| Ring 3 reflector peak temperature (°C) | 585 |
| Core barrel peak temperature (°C) | 329 |
| Reactor vessel peak temperature (°C) | 317 |
| Reactor Cavity Cooling System (RCCS) heat removal (MW) | 0.44 |
| Irradiation loop heat removal (MW) | 0.17 |

The reflector temperatures decrease with increasing power, because the coolant flow increases to maintain the same temperature rise across the core. The increased flow means the velocity is higher, which results in higher heat transfer coefficients from the structures to the coolant.

Heat removed by the coolant in the central irradiation loop reduces the temperature of the central reflector but has little effect on the fuel and fuel-block temperatures, because the amount of heat removed is very small compared to the heat being generated in the fuel and removed by the flow through the coolant channels.

Sensitivity calculations were used to investigate the impacts of trying to reduce the core bypass flow in the outer reflector, increasing the coolant inlet and outlet temperatures, and changing the axial power shape. Only the increased coolant temperatures had a significant impact on the fuel and structure temperatures, but the impact was no greater than would be expected.

## VI. HTG-TR SAFETY BASIS

The prismatic core design provides particle fuel radionuclide retention in a passively safe reactor that requires no energized systems for long-term decay heat removal. The large thermal capacity of the core results in long transients on the order of days. The primary safety feature is the use of TRISO fuel. The coatings on the fuel particles have been shown to prevent fission product release both historically and during recent irradiation testing in ATR, and the use of coatings in this reactor is within the fuel qualification envelope. If some fission products escape the coating, the fuel matrix would be the next barrier to fission product release. The fuel compacts are sealed in the graphite fuel blocks that are not structurally challenged by the temperatures achieved during the most severe accidents. The reactor building provides the final barriers to fission product release to the environment.

Use of an inert gas for both the primary coolant and the gap between the irradiation loops and the experiment wells precludes any chemical interactions with the structures in the plant. Use of inert gas also means that there would be no adverse coolant interactions if a leak from an experiment irradiation loop develops. One challenging feature of the helium coolant is that it does not provide radiation shielding. This means that removal of irradiated experiments will require portable shielding or casks for movement of the test specimens from the reactor to a storage area.

The neutronic characteristics of the core and large graphite reflector reduce the fast neutron fluence to the core barrel and reactor vessel. The thermal-neutron fluence to these components can be reduced by using borated pins in the side reflector.

Decay heat can be removed using only passive systems and physical processes. Decay heat from the core is transferred radially to the reactor vessel, primarily by radiation and conduction. From the reactor vessel, radiation and natural convection in the reactor cavity transfer energy to the water-cooled Reactor Cavity Cooling System (RCCS). Flow through the RCCS is provided by natural convection from a large pool located higher than the reactor cavity.

### VI.A. Safety Performance

Generally accepted criteria for TRISO fuel are peak temperatures below 1,250°C during steady-state operation and within the time-at-temperature envelope established by AGR fuel testing in the ATR during an accident or transient. As is shown below, the peak transient temperatures are lower than those during steady state, so increasing the power and flux during steady state may be possible if further fuel testing shows that operating temperatures above 1,250°C result in no challenges to the fuel integrity.

The operational events and accidents for this HTG-TR will be similar to those for a commercial prismatic block reactor: increases or decreases in coolant flow, changes in the reactor inlet temperature, reactivity-initiated events (such as control rod withdrawals), and changes in coolant system pressure. Accidents of particular interest are water/steam ingress events (potential reactivity insertion) and total losses of forced convection cooling.

A test reactor introduces some additional accidents to be considered. Most accidents initiated in the loops, such as a loop blowdown or loss of cooling, will be seen in the reactor as a perturbation in the reactivity that may be bounded by control-rod-driven reactivity events; detailed analyses would need to be performed as the reactor design matures. Failure of one of the irradiation loops could result in the release of radioactive material to the reactor building. Liquid-metal or molten-salt loops would be at low pressure, making piping failure less likely. Specific transport analyses would need to be performed for the reactor building layout and systems to determine if limits on loop source terms would need to be imposed on the experiments to ensure that atmospheric releases are within established safety limits.

The most likely initiator for a water/steam ingress event is a rupture in the steam generator tube. Designs for commercial plants isolate the steam generator and have included a non-safety-grade feedwater dump system to mitigate this event, and those approaches could be used for this reactor as well. A water loop in the core is also a potential source for this event; however, failure of both the experiment pressure boundary inside the reactor vessel and the primary coolant system boundary in the same test well may fall within the beyond-design basis event realm, because it involves independent failures of two ASME pressure vessel boundaries.

Total losses of forced convection cooling are referred to as conduction cooldown transients, because the heat in the core is conducted (and radiated) to the reactor vessel and then to the RCCS. In a pressurized conduction cooldown (PCC), the primary coolant system pressure boundary remains intact. In a depressurized conduction cooldown (DCC), the system is depressurized, and the general assumption is that a loss-of-coolant accident has occurred. The DCC typically produces the limiting fuel temperatures. Both DCC and PCC transients were simulated with RELAP5-3D.

The DCC transients were simulated by imposing a 1-s blowdown and flow coastdown on the system; only the core outlet was open to atmospheric pressure. Reactor scram was also assumed to occur at the beginning of the transient. Fig. 9 presents the peak fuel temperatures from the DCC transient. The maximum values are about 150°C lower than the steady-state values and well within the AGR time-at-temperature envelope.

The PCC accident was modeled by imposing a 5-s flow coastdown in the primary coolant and the irradiation loop, initiating a reactor scram, and maintaining the normal operating pressure. Peak fuel temperatures, shown in Fig. 10, are about 100°C lower than those in the DCC transient; reductions in temperatures were observed in the other structures as well.

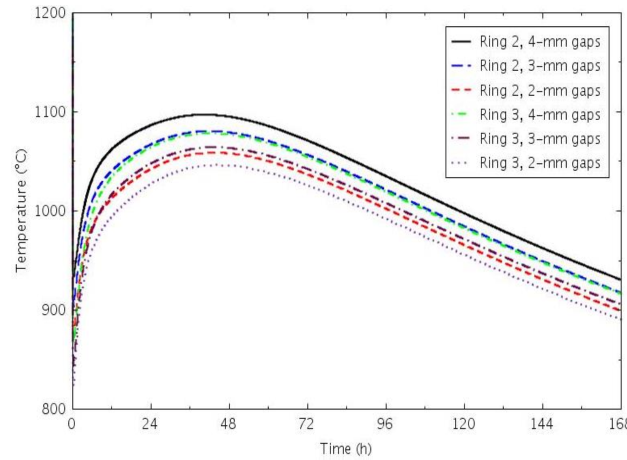


Fig. 9. Peak fuel temperatures for a DCC transient.

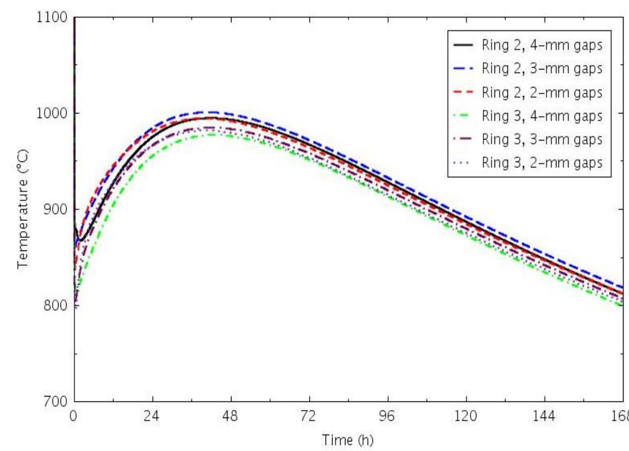


Fig.10. Peak fuel temperatures for a PCC transient.

## VII. REFUELING AND REPLACEMENT OF TEST ARTICLES

The availability and operational effectiveness of a test reactor is a strong function of the ease and speed with which fuel and test articles can be replaced. High-flux reactors must (in general) be refueled more frequently than power reactors. The test samples must be removed after a specified time in the operating core per experimental demands. The refueling and test insertion/replacement schedules may not conveniently overlap at a given power level, so cycle planning becomes a balancing exercise between the required test exposure, loop power and temperature, and fuel reactivity. Even if online test insertion and removal (e.g., with a pneumatic shuttle system) is available, frequent outages are the norm. Design emphasis is placed on simplicity of reloading operations and accessibility to core structures.

The large size of the HTGR fuel and reflector blocks, control rods, and test spaces require appropriately sized handling equipment. The large size of the core and vessel translates into ample space above the core for maneuvering these structures. Access to the fuel and removable reflector blocks is achieved through the vessel head penetrations, most of which are used for the control rod drive assemblies and which are large enough to accommodate a block. All fuel blocks can be retrieved using a fuel-handling machine of similar design to the one that that was successfully operated at the FSV reactor.

Because access to the fuel, test spaces, and control rods is likely to be from above the core, the pressure vessel head will be crowded with penetrations, as shown in Fig. 11. Each penetration for Ring 4 would service two independently controlled control rods. Preliminary evaluation indicates that the required penetration configuration is workable,[2] but further design calculations are needed to optimize the location of penetrations and interior structures.

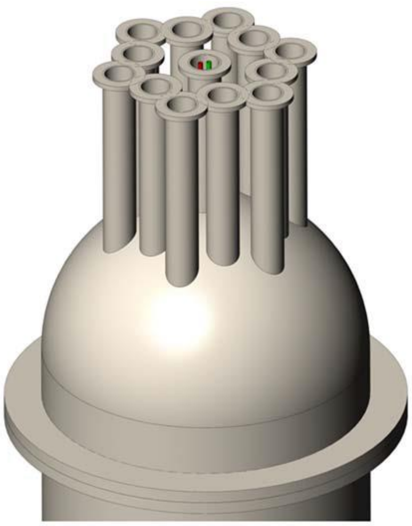

Fig. 11. Vessel head penetrations.[2]

## VIII. CONCLUSIONS

The use of an HTGR with materials irradiation as its *primary* mission is unprecedented. This study presented a recent point design for a graphite-moderated HTG-TR developed by INL. The 200-MW HTG-TR is predominantly a thermal-neutron spectrum reactor with a sizable graphite pile cooled by helium gas. The core features a large number of irradiation positions with large test volumes and long test lengths, providing an ideal environment for thermal neutron irradiation of large test articles. The HTG-TR core has been specifically designed and optimized to provide the highest possible thermal and fast-neutron fluxes, which occur in ring 3 (3.90E+14 n/cm$^2$/s) and ring 1 (1.17E+14 n/cm$^2$/s), respectively.

The large irradiation volumes and long (110-day) cycle length, plus the competitive thermal neutron-irradiation flux and large operational safety margins are the main strengths of the HTG-TR. This translates into greater flexibility for a variety of irradiation experiments and test materials. The HTG-TR can support independent irradiation loops containing a variety of coolant fluids (e.g., liquid metal, liquid salt, light water, and other gases or steam); the reactor design is passively safe; and peak fuel temperatures during design-basis conduction cooldown (loss of forced cooling) accidents remain below the steady-state operating temperatures and well below safety limits.

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