INL/EXT-16-37661

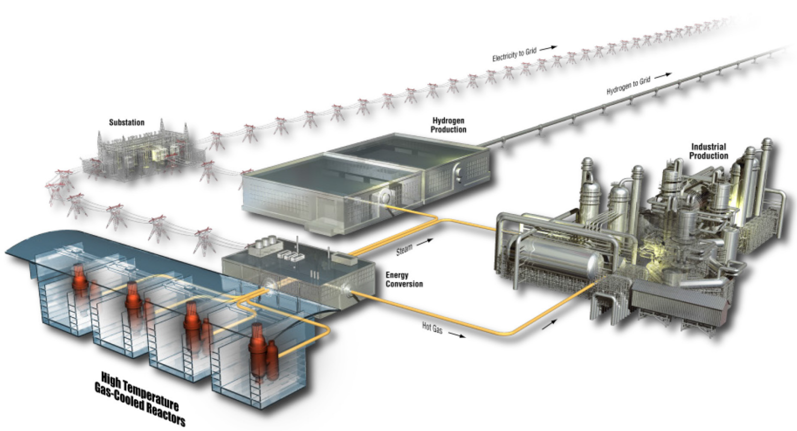
High Temperature Gas-Cooled Test Reactor Point Design: Summary Report

J. W. Sterbentz
P. D. Bayless
L. Nelson
H. D. Gougar
J. Kinsey
G. Strydom

January 2016

The INL is a
U.S. Department of Energy
National Laboratory
operated by
Battelle Energy Alliance

Idaho National Laboratory



INL/EXT-16-37661

High Temperature Gas-Cooled Test Reactor Point Design: Summary Report

J. W. Sterbentz
P. D. Bayless
L. Nelson
H. D. Gougar
J. Kinsey
G. Strydom

January 2016

Idaho National Laboratory
INL ART TDO Program
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

INL ART TDO Program

High Temperature Gas-Cooled Test Reactor Point Design: Summary Report

INL/EXT-16-37661

January 2016

Authors:

James W. Sterbentz — 1/29/2016
Nuclear Engineer — Date

Paul D. Bayless — 1/29/16
Nuclear Engineer — Date

Approved by:

Diane V Croson for
Hans D. Gougar — 1/29/16
INL ART TDO Deputy Technical Director — Date

Michelle T. Sharp — 1/29/16
INL ART TDO Quality Assurance — Date

ABSTRACT

A point design has been developed for a 200-MW high-temperature gas-cooled test reactor. The point design concept uses standard prismatic blocks and 15.5% enriched uranium oxycarbide fuel. Reactor physics and thermal-hydraulics simulations have been performed to characterize the capabilities of the design. In addition to the technical data, overviews are provided on the technology readiness level, licensing approach, and costs of the test reactor point design.

CONTENTS

FIGURES

TABLES

ACRONYMS

AGR	Advanced Gas Reactor
ART	Advanced Reactor Technologies
ATR	Advanced Test Reactor
CP	construction permit
DCC	depressurized conduction cooldown
DOE	Department of Energy
FHR	fluoride salt-cooled reactor
FIMA	fissions of initial heavy metal atoms
FOAK	first-of-a-kind
FSV	Fort St. Vrain
GA	General Atomics
HFIR	High-Flux Isotope Reactor
HTGR	high-temperature gas-cooled reactor
HTGR-TR	high-temperature, gas-cooled test reactor
HTR	high-temperature reactor
INL	Idaho National Laboratory
LWR	light water reactor
MHTGR	modular high-temperature gas-cooled reactor
NA	not applicable
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
O&M	operations and maintenance
PCS	primary coolant system
PF	packing fraction
PSR	permanent side reflector
R&D	research and development
RCCS	reactor cavity cooling system
SAR	safety analysis report

TCI	total capital investment
TEDE	total effective dose equivalent
TR	test reactor
TRISO	tristructural isotropic
TRL	technology readiness level
UCO	uranium oxycarbide
U.S.	United States

High Temperature Gas-Cooled Test Reactor Point Design: Summary Report

1. SUMMARY

A point design for a graphite-moderated, high-temperature, gas-cooled (HTGR) test reactor (TR) (HTGR-TR) has been developed by Idaho National Laboratory (INL) as part of a United States (U.S.) Department of Energy (DOE) initiative to explore and potentially expand the existing U.S. TR capability. This report provides an initial summary description of the design and its main attributes. Although there are no HTGRs operating today in the U.S., the design of the HTGR-based 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 HTGR-TR has drawn heavily on recent advancements in tristructural isotropic (TRISO) particle fuel, graphite, and in-core HTGR materials from the very successful DOE Advanced Gas Reactor (AGR) Program and associated U.S. Nuclear Regulatory Commission (NRC) interactions. These advancements, along with recent and past HTGR technology, have been incorporated into the design of the HTGR-TR.

The HTGR-TR core is composed of hexagonal prismatic fuel blocks and graphite reflector blocks. Figure 1 shows a cross section of the reactor vessel and core. Twelve fuel columns (96 fuel blocks total) are arranged in two hexagonal rings (Rings 2 and 3) to form a relatively compact, high-power density, annular core sandwiched between inner, outer, top, and bottom graphite reflectors. The fuel columns are 8 blocks high. TRISO particle fuel from the DOE AGR Program has been adopted with the larger 425-μm uranium oxycarbide (UCO) kernel with an enrichment of 15.5-wt% ^{235}U. The reactor power is 200 MW and has a power cycle length of 110 days. Assuming a four-week shutdown time between cycles, it also has a maximum availability factor of 78%.

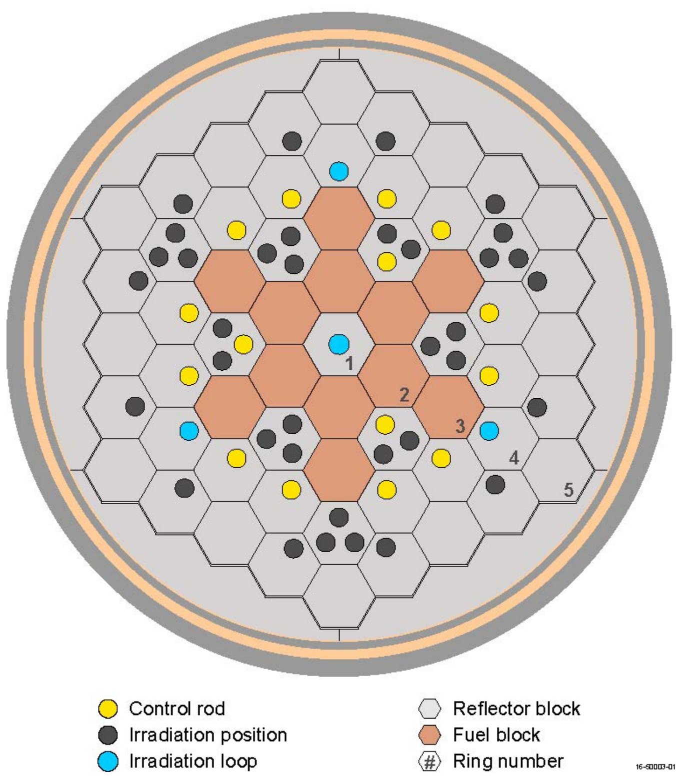


Figure 1. Reactor vessel cross section in core region.

The HTGR-TR is predominantly a thermal-neutron spectrum reactor with a sizable graphite pile cooled by helium gas. The highest thermal-neutron flux occurs in the outer reflector (Ring 3). High fast-flux irradiation levels are more difficult to achieve. The maximum fast-flux levels are produced in the annular core, but—due to excessive temperatures in the high-power density core under accident conditions—all the irradiation test facilities have been initially located in the inner and outer reflectors where fast neutrons are moderated and fast-flux levels decline. Fast flux can be enhanced in the central reflector column (Ring 1) with the removal of graphite from the column blocks, and this is where the maximum fast flux occurs.

The core features a large number of irradiation positions with large test volumes and long test lengths, ideal for thermal-neutron irradiation of large test articles (e.g., full length partial fuel rod assemblies). Up to four test loop facilities can be accommodated with pressure tube boundaries to isolate test articles and test fluids from the primary helium coolant system. One of these test loop facilities is located in the center of the core (Ring 1) and has a maximum thermal and fast flux of 1.61E+14 n/cm^2/s and 1.17E+14 n/cm^2/s (E_n>0.18 MeV), respectively. The three other loop facilities can be located in the outer reflector (Ring 4) with a maximum thermal and fast flux of 2.82E+14 n/cm^2/s and 2.28E+13 n/cm^2/s (E_n>0.18 MeV). The in-core loop facilities have test volumes of about 14 L.

It is expected that one of these loop locations in the outer reflector would contain a pneumatically-driven rabbit system. The core can also accommodate at least 36 irradiation positions for drop-in test capsules in the outer graphite reflector. In Ring 3 these positions have a maximum thermal and fast flux of 3.90E+14 n/cm^2/s and 5.24E+13 n/cm^2/s, respectively. All test positions can be the full length of the active core (6.34 m), and the Ring 3 and 4 positions could be up to 16 cm in diameter. The 8-cm-diameter irradiation positions shown in Figure 1 each have a test volume of 30 L, resulting in a total test volume over 1100 L. The positions shown in Figure 1 are just one example of a possible configuration; larger or smaller diameter facilities could be accommodated without much difficulty.

A modern commercial HTGR will operate at relatively high gas pressure (7 MPa) and high outlet gas temperature (750–850ºC). The point design TR is also designed to operate at 7 MPa, but at a lower outlet gas temperature (650°C). The lower outlet temperature was selected to ensure sufficient thermal margin under normal operating conditions to prevent melting of metallic in-pile tubes during accident conditions. Penetration of the top-head reactor pressure vessel boundary by both control rod guide tubes and loop pressure tubes could potentially result in top-head crowding. Future engineering assessments will need to consider not only possible crowding issues, but penetration design and maintenance of the pressure boundary integrity due to frequent loading and unloading of fuel and experiments.

The primary mission of the HTGR-TR is material irradiation and therefore the core has been specifically designed and optimized to provide the highest possible thermal and fast neutron fluxes. A helium-cooled 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). Power levels and coolant conditions are such that it can serve as a test bed supporting developments in high efficiency electricity production (steam and Brayton cycle), as well as process heat-driven energy products including hydrogen. Other secondary missions such as isotope production can also be supported. The range of temperatures and test loop coolants afforded by the HTGR-TR would be most useful to molten salt and gas-cooled reactor developers. Loop experiments for investigating fuel, material, and coolant interactions in a radiation field are supported by only a few facilities in the U.S. and around the world. Because of the large volumes within the multiple loop positions, advanced water-cooled reactor fuels can also be tested. Much of the customer base of INL’s Advanced Test Reactor (ATR) could also be served with the HTGR-TR with half-sized or even full-sized fuel assemblies for smaller light water reactor (LWR) concepts such as Nuscale being accommodated.

The HTGR-TR has strong negative fuel and moderator temperature coefficients. Under normal critical operation and over the entire power cycle length, the reactor will operate safely because of strong negative temperature feedback and high-thermal inertia of the graphite. One aspect of the reactor control that was not considered in the design is the use of burnable poisons. Burnable poisons will eventually play an important role in holding down the initial core excess reactivity over the 110-day power cycle and for flattening the power profile. As the primary performance goal in this study was to maximize the irradiation flux, optimization of the burnable poison loading was omitted but will be required in the next design phase. Axial and radial placement of the burnable poison rods (B_4C) in the fuel columns will need to be done judiciously so as to minimize any effect on the flux profiles in the irradiation spaces. Once burnable poisons are incorporated into the reactor design, the movable control rod pattern can be adjusted to optimize core performance.

The reactor design is passively safe and peak fuel temperatures during design-basis conduction cooldown (loss of forced cooling) accidents are below the steady-state operating temperatures and well below safety limits. Long-term decay heat removal is provided by a natural-circulation driven, water-cooled system such that no energized systems are required. Heat is transferred from the reactor vessel to the cooling system by passive radiation and natural convection mechanisms.

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 HTGR test reactor. This translates into greater flexibility for a variety of irradiation experiments and test materials. Another potential strength is possibly to increase the cycle length. Although the HTGR test reactor meets the 90-day metric criterion, a much longer cycle length (up to 280 days) can readily be achieved with simple increases in the TRISO particle packing fraction (PF=35%). Longer irradiations can potentially accumulate fluence faster with fewer reactor shutdowns, despite a slightly reduced flux.

As part of the overall Advanced Test/Demonstration Reactor Options Study, an assessment of the maturity of Generation IV reactor technologies was conducted by a multi-laboratory panel of experts. A technology readiness scale developed by DOE was used to evaluate the HTGR-TR system. For the HTGR, the lowest technical maturity scores were assigned to certain metallic components inside the pressure vessel. When exposed to core conditions under accident conditions, these may be subjected to failure. If coolant temperatures are limited to 850°C, SA508/533 (the steel alloy used in LWRs) is adequate for the pressure vessel. Metallic control rod drive tubes and seals, however, may fail in the event of the most severe loss-of-forced-cooling events, with subsequent depressurization of the core. While this is not expected to cause significant fuel particle degradation, circulating radiological inventory would be released and expensive core repairs would be necessary. Qualification of new alloys or even the use of carbon or silicon carbide composites for the guide tubes may be needed. For these reasons, the reactor enclosure subsystem for the demonstration plant was assigned a technology readiness level (TRL) of 5. The overall conclusion of the panel was that the HTGR, with outlet temperatures limited to 850°C, is suitable for near-term deployment as either a test or demonstration reactor.

This TRL is probably too low. The control rod guide tubes are not part of the primary coolant system pressure boundary. The control rod drive housing connections on the reactor vessel upper head will not see the high temperatures that the core will during the limiting design basis accidents, and thus would not be expected to fail. The pressure boundary at risk would be that between the primary coolant and the irradiation loops. While the temperature of these components may be high enough that they may need to be replaced, they would not be expected to fail, and thus the primary pressure boundary would remain intact.

The capital, operating, and decommissioning costs for the HTGR-TR are based on the information presented in the *Next Generation Nuclear Plant (NGNP) Pre-Conceptual Design Report* for a 350-MW first-of-a-kind (FOAK) reactor with a single reactor module, and include indirect costs and contingencies.[1] The detail cost model utilized for this cost estimate was developed as part of the NGNP

Project using data from three vendors. The total capital cost for an HTGR is comprised of the following cost categories: preconstruction costs, direct costs, indirect costs, and project contingency. Operating costs include staffing requirements, annual fees, insurance, taxes, material supplies, outage costs, and administration and general cost overhead. The total capital investment (TCI) required to build a 200-MW HTGR-TR is estimated at $3,942 million, within a −50% and +50% uncertainty range of $1,971–5,913 million.

The HTGR-TR aligns with the NRC's definition of a Test Facility (TR), as found in 10 CFR 50.2. Test reactors are one of the types of non-power reactors that the NRC license under the authority of Subsection 104c of the Atomic Energy Act, and are therefore issued "Class 104c" licenses. Congress directed the NRC to impose the minimum amount of regulation on Subsection 104(c) research reactor and TR licensees. In keeping with this direction, the NRC staff utilizes NUREG-1537 as the primary guidance for review of research reactors and TR technologies and license applications. DOE and NRC established a joint initiative in July 2013 to develop guidance for advanced reactor developers and other stakeholders on how the existing General Design Criteria (GDC) reflected in 10 CFR 50, Appendix A, can be adapted to non-LWRs. A proposed set of GDC adaptations specific to modular HTGRs was developed by a DOE/national laboratory team and submitted to NRC for review in December 2014.

A self-assessment has been performed on the HTGR test reactor scoring against the DOE-developed criteria. It is shown in Appendix A that the test reactor scores 89 out of a possible total of 117, i.e. 76%.

2. TEST REACTOR OBJECTIVES AND MOTIVATION FOR CONCEPT SELECTION

The primary objective of the HTGR-TR design was to provide a versatile, multi-purpose, high flux facility for advanced reactor fuels and materials irradiations. Currently, such capability in the U.S. is provided mainly by the High-Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory and ATR at INL. HFIR and ATR are both LWRs with over 40 years of safe and reliable operating and irradiation experience. Table 1 provides a comparison of pertinent test positions and reactor data for HFIR, ATR, and the HTGR-TR design.

Table 1. Comparison of irradiation characteristics of High-Flux Isotope Reactor, Advanced Test Reactor, and high-temperature gas-cooled test reactor.

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	50.8	2-10E+14	≤1.5E+14 (E_n>0.111 MeV)	85	1251	23
ATR	Flux trap	13.3	121.9	4.4E+14	2.2E+14 (E_n>0.1 MeV)	110	116	30–60
HTGR-TR	Graphite reflector	≤16.0	640.0	3.9E+14	1.2E+14 (E_n>0.18 MeV)	200	23	110

Flux levels in the HTGR-TR are below those of HFIR and ATR but not substantially lower despite the large differences in core power density. Note that in the HFIR center flux trap the thermal flux is much higher, with an average 2.35E+15; these super-high flux positons are usually reserved for isotope production (^{252}Cf). The 110-day power cycle length of the HTGR-TR is substantially longer than the 23-day HFIR cycle and 30- to 60-day ATR cycles. Furthermore, the product of the flux and irradiation time and relatively large number of test positions and large test volumes available in the HGTR-TR help increase the usefulness of the HTGR-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 an LWR, fast reactor, or fluoride salt-cooled reactor.

Another very important and useful feature of the HTGR-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 primary coolant gases and is estimated to have small or minimal reactivity impact on the relatively large HTGR core.

Still other useful features of the HTGR-TR include the ability to generate electricity and produce isotopes. The electricity could be sold to a local utility for revenue and any surplus supplied to the national laboratory reactor site. The production of commercial isotopes could also 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 (R&D). 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.

3. 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, followed by a brief discussion of the ex-vessel systems.

3.1 Reactor Fuel Form and Configuration

The HTGR-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 (GA) design[2] that goes back to the Fort Saint Vrain (FSV) fuel block design (Figure 2). This block design offers great flexibility in enrichment zoning, particle packing fraction (PF) zoning, placement of burnable poison rods, and cooling. Figure 3 shows a detailed computer model rendering of the FSV fuel block used in the test reactor physics analysis. Optimization of the fuel block dimensions, fuel rod pitch, fuel rod diameter, and number of fuel and coolant channels remains for future work.

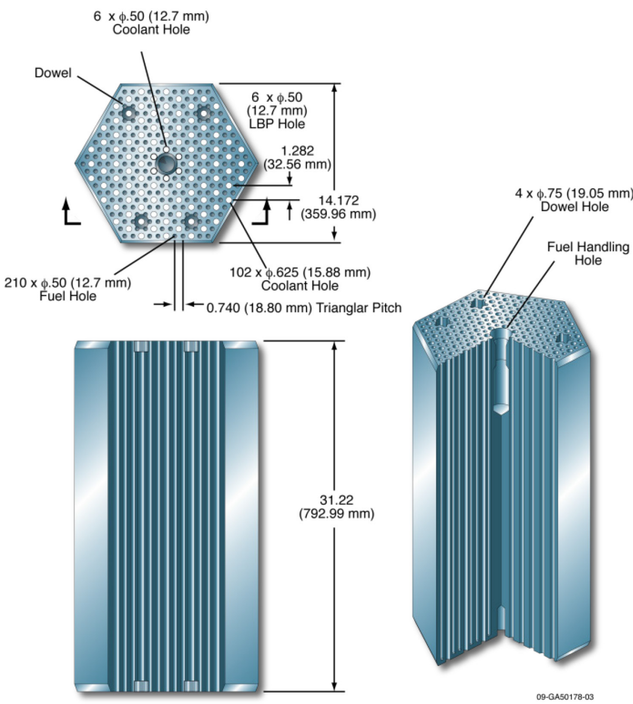


Figure 2. Fort Saint Vrain fuel block.

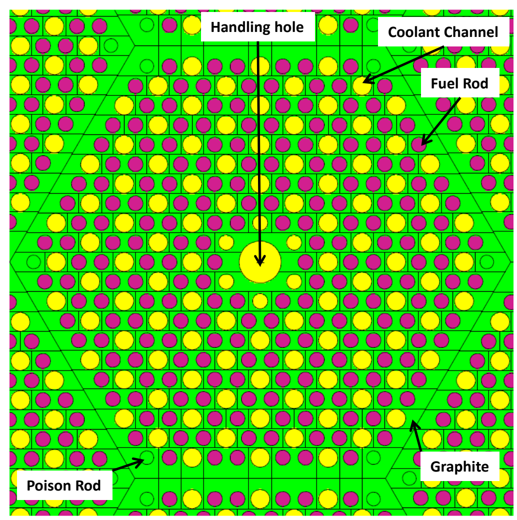


Figure 3. Fuel block model.

The TRISO particles matrixed in cylindrical fuel compacts form an integral high-temperature ceramic system specifically designed for the NGNP HTGR commercial reactors. The same TRISO fuel is used for the TR. Recent irradiation testing of the TRISO fuel on the DOE AGR Program has demonstrated the robustness and high performance of the fuel under high temperature (1300ºC), burnup (20% fissions of initial heavy metal atoms [FIMA]), and fast fluence (5.5E+21 n/cm^2) conditions. The tests have been very successful with in most cases no fuel particle degradation. The AGR-1, AGR-2, and AGR-3/4 irradiation tests have included a variety of particle designs that have provided substantial particle performance data. The specific TRISO particle design adopted for the TR will be based on the up and coming AGR-5/6/7 qualification test particle design that features a large 425-µm-diameter UC0.5O1.5 kernel, 15.5-wt% enrichment, and PF=25 or 38%.

Compacts for the TR, however, will have a much lower particle PF (PF=15%) to boost the irradiation fluxes. Four relatively recent and notable particle and compact design improvements include:

- The larger kernel diameter (425 versus 350 μm)
- Higher UCO density (11.04 versus 10.40 g/cm^3)
- Higher graphite binder density (1.70 versus 1.2 g/cm^3)
- Higher bulk graphite density (1.83 versus 1.74 g/cm^3).

These improvements boost HTGR core reactivity.

The TR core configuration (baseline) is shown in Figure 3 and features the following characteristics:

- Prismatic hexagonal fuel and graphite reflector blocks
- High-leakage annular core
- Block pitch of 36 cm
- Five-ring core: Ring 1 (inner reflector), Rings 2 and 3 (annular core), Rings 4 and 5 (outer reflector)
- 12 fuel columns
- Eight fuel blocks per column
- 210 fuel and 108 coolant channels per fuel block
- Core height of 9.2 m with an active height of 6.4 m
- Core diameter of 3.4 m
- 200-MW thermal power.

The baseline point design is similar in many respects to modern commercial HTGRs. Both are large graphite piles with annular, high-leakage cores formed by prismatic fuel and graphite hexagonal blocks. The helium coolant, pressure, temperature, down flow, and flow path through the pressure vessel are essentially the same. Both have inner, outer, top, and bottom graphite reflectors. The TR core configuration, however, diverges from the much larger commercial reactor in the number of fuel blocks and power as the TR mission changes to include the material irradiation. To boost irradiation flux in the outer reflectors where the irradiation test facilities are located, the 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 (6–8 W/cm^3).

The TR fueled core is an annular core sandwiched between an inner and outer graphite reflector. The annular core has only 12 fuel columns: six in Ring 2, and six more in Ring 3 where the fuel blocks alternate with graphite blocks around Ring 3 (Figure 4). Each fuel column is eight fuel blocks high. Modern commercial cores can have up to 102 fuel columns and are 10 blocks high. 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 Hex Ring 4 are all graphite block columns; 12 with control rods and the other six with additional irradiation test positions. Hex Rings 4 and 5 are the outer graphite reflector. Beyond Ring 5 is the permanent side reflector (PSR), graphite blocks to form-fit the core barrel. The core is approximately 3.4 m in diameter and 9.2 m in total height. The total core thermal power is 200 MW.

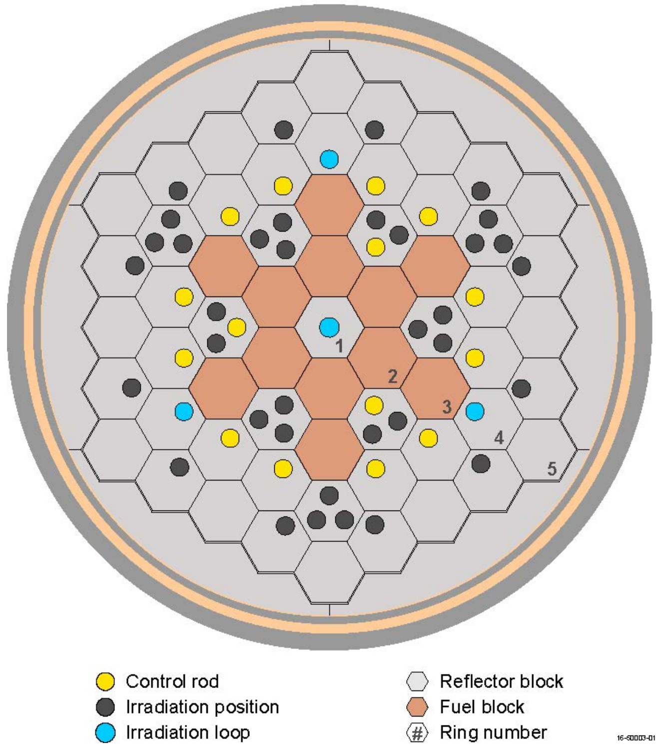


Figure 2. Baseline test reactor core configuration.

The central irradiation loop in Ring 1 is a dedicated pressure loop facility. It has a stainless-steel 316L tube with an outer diameter of 11.34 cm (4.5 in.) and wall thickness of 1.35 cm (0.531 in.). The outer irradiation positions in Rings 3 and 4 can have a similar pressurized loop facility. Loops can accommodate relatively large test specimens cooled by various fluids including high-pressure light water, liquid salt, liquid sodium, or different gases (e.g., helium). In addition, the Ring 3 and 4 irradiation positions can have thinner-walled metallic containment tubes (molybdenum, zirconium, titanium) for drop-in type capsule experiments. These tube facilities can have diameters up to approximately 16 cm. For the TR design evaluation, these facilities have an outer diameter of 10.16 cm (4.0 in.), the same as the control rod holes in the graphite blocks. Table 2 lists the irradiation facility by ring, type, and pertinent metrics.

Table 2. Irradiation facilities and characteristics.

Hex Ring No.	Number of Loops	Number of Tubes	Test Diameter (cm)	Test Length (m)	Test Volume per Facility (L)	Total Test Volume (L)
1	1	0	5.4	6.34	14	14
2	0	0	—	—	—	—
3	0	15	8.0	6.34	30	450
4	3	9	5.4/8.0	6.34	14/30	42/270
5	0	12	8.0	6.34	30	360
Total	4	36	—	—	—	1136

There are total of 15 control rods in the outer reflector (Figure 4). The combined worth of these rods is approximately −$50; enough negative reactivity to shut the core down under both hot and cold conditions. Sufficient margin exists to shut down, even if two or three rods are stuck out. The introduction of burnable poisons, irradiation tubes, and other in-core hardware will also introduce negative core reactivity and enhance the control rod shutdown margin.

Control rod and loop penetrations through the top head of the reactor pressure vessel may compete for the limited room available in the TR 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 TR design with its compact core configuration specifically located the control rods in the outer reflector to address this potential problem. The key reactor parameters are summarized in Table 3.

Table 3. Key reactor parameters.

Reactor thermal power	200 MW
Primary coolant	Helium gas
Primary coolant system (PCS) 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
Number of primary coolant loops	1
Fuel format	Prismatic block with coolant channels and fuel rods (compacts)
Fuel columns	12
Fuel blocks per column	8
Fuel blocks per core	96
Fuel type	$UC_{0.5}O_{1.5}$ TRISO-coated particle
Fuel PF	15.0%
^{235}U enrichment	15.5 wt%
Average core power density	23.4 W/cm^3
Power cycle length	110 days
Reflector material	graphite
Reactor vessel internals material	• Alloy 800H (control rod sheath) • Stainless-steel 316L (irradiation loop pressure tube) • Molybdenum, zirconium, titanium (irradiation tubes in outer reflector)
Core structural material	Graphite
Control rod material	• B_4C in graphite • Boron-10 enrichment 30–50%
Vessel material	Steel
Core fueled height	6.4 m
Core outer diameter	3.4 m
Core total height	9.2 m

3.2 Reactor Physics

The proposed TR design shown in Figure 4 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, all in Hex Rings 1, 2, and 3 only for compactness. Allowing fuel columns in Ring 4 would have required an additional outer reflector hex ring or additional 30 graphite columns in Ring 6, plus more PSR blocks. This would also increase in the core and pressure vessel diameter by 0.72 m. Since 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.

In the physics analyses, there were six primary design variables:

- Core power (50–250 MW)
- Particle PF (5–50%)
- Power cycle length
- Arrangement of fuel columns in core
- Number of fuel columns (6, 7, 12, and 18)
- Number of fuel blocks in a fuel column (4, 5, 6, 7, and 8).

Some design variables were fixed. While these fixed variables simplified the design analyses, it left open the possibility for a more optimized TR design for future designs. The fixed design variables included:

- FSV fuel block design
- Single 15.5% enrichment
- AGR-5/6/7 particle design.

There were also TRISO particle fuel and thermal hydraulic limits that had to be considered:

- Particle power (<400 mW)
- Compact burnups (<20% FIMA)
- Compact fast fluence (<5.5E+21 n/cm^2 at E_n>0.18 MeV)
- Fuel rod power-peaking (<2.0 peak-to-average)
- Peak fuel temperature (1250°C) normal steady-state operations
- Peak fuel temperature (within the AGR time-at-temperature envelope) accident conditions.

The following assumptions were also made in the physics evaluations:

- Uniform core PF
- Unrodded core
- No burnable poison rods
- Compact fuel radius of 0.6225 cm
- Homogenized compacts.

The combination of design variables, fixed-value variables, and limitations resulted in a complex interplay between the design variables where some variables were diametrically opposed to one another, while others were closely aligned. In all cases, variable ranges were restricted by the fuel and thermal hydraulic limitations.

3.2.1 General Physics Design Characteristics

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 (Figure 5), and the reduction of the number of fuel blocks in the core either 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 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 (>1250°C). High power density also leads to excessive ^{235}U fuel burnup rates and shorter power cycle lengths. The TR design attempted to maximize the power density while observing the fuel and the temperature limitations and cycle length goals.

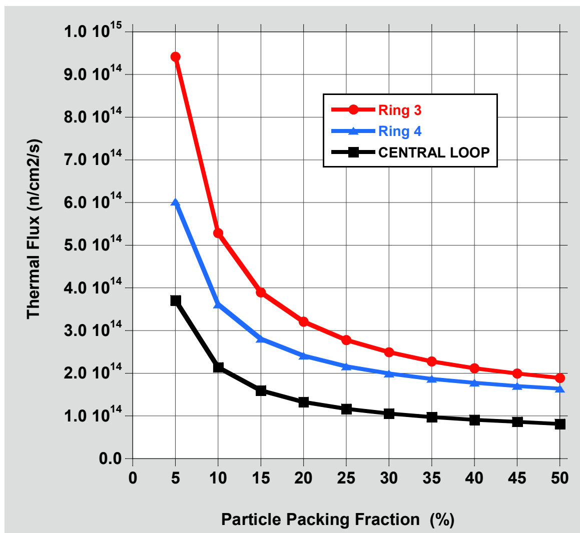


Figure 3. Thermal flux versus packing fraction.

The final optimal TR core configuration or fuel column arrangement and number of fuel columns is shown in Figure 4. The design balances core power, particle PF, particle power, fuel temperature, and cycle length. The result is a 12-fuel column, 8-block-high core configuration, uniform particle PF=15%, and a total core power of 200 MW. A single-batch core load can sustain a 110-day power cycle length. Calculated results for these design parameters are presented below.

3.2.2 Calculated Physics Results

The neutronic calculations used the Monte Carlo N-Particle 5 Version 1.60 computer code[3] and INL-developed depletion methods and software. Detailed Monte Carlo N-Particle core models were developed based on the GA FSV fuel block design and the baseline core configuration depicted in Figure 3. The calculated results are specifically for the core configuration in Figure 4 at 200 MW, particle PF=15%, and 8-block-high fuel columns with no burnable poisons, enrichment grading, PF grading, or control rod insertion (except in the section on control rod worth).

3.2.2.1 Maximum Irradiation Flux.
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 annular core fuel blocks, the excessively-high fuel block temperatures (800–1000ºC) 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 for three irradiation positions in Table 4. 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 (Figure 4). 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 4. Maximum fast and thermal irradiation fluxes by test 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

It should be noted that a thick-walled pressure containment tube (stainless steel) for loop experiments can reduce the local thermal flux by a factor of 2. The highest useable fast flux is calculated to be 1.17E+14 n/cm^2/s (E_n>0.18 MeV) and occurs in the central loop facility. This fast flux is achieved by removing the graphite mass in the Ring 1 graphite blocks. Without the graphite removal, the fast flux is 4.64E+13 n/cm^2/s.

3.2.2.2 Cycle Length and Burnup.
The cycle length for the baseline TR is calculated to be 110 days. The fuel rod average burnup ranges from 4.62 to 9.56% FIMA with a core average of 7.36% FIMA. These burnups are slightly less than the AGR-2 UCO burnups that ranged from 4.90 to 10.30% FIMA with an average burnup of 8.18% FIMA. The AGR-2 TRISO particles were also 425°µm-diameter UCO kernels, but with a slightly lower enrichment of 14-wt% ^{235}U and a higher PF=36%. The AGR-5/6/7 qualification and margin tests will use a 425-µm-diameter UCO kernel with an enrichment of 15.5-wt% ^{235}U, just like this TR, but with higher PFs of 25 and 35%. The AGR-5/6/7 compacts should sustain burnups of 8.0–18.6% FIMA, which is substantially higher than the 9.56% FIMA maximum burnup predicted at end-of-cycle for the TR.

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% (Figure 5). The cycle lengths however can be substantially extended to 210 and 281 days, respectively (Figure 6). Variable cycle length through changes in PF could be a useful feature of the TR. Average compact burnups will also increase to approximately 8.85, and 9.26% FIMA for PF=25 and 35%, respectively.

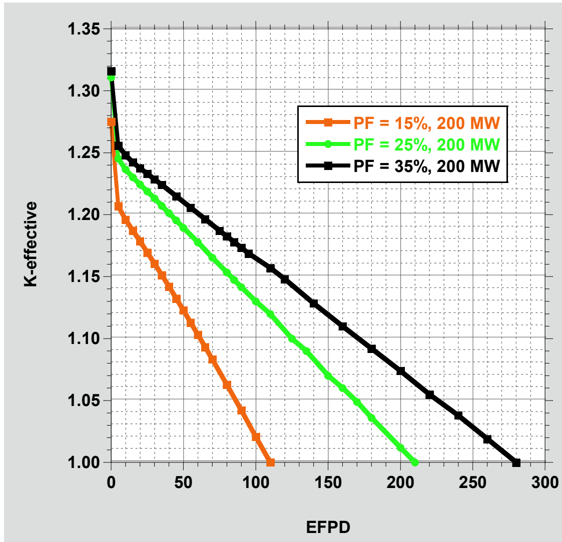


Figure 6. Reactivity letdown versus burnup.

The use of burnable poison rods in the six available corner positions in the prismatic fuel blocks can reduce power-peaking at the core-reflector interfaces. Poison rods designed to be graphite containing B_4C with very low concentrations of boron-10 (<1%) should be sufficient to hold down interface reactivity and local power-peaking.

3.2.2.3 Control Rod Worth. A preliminary control rod design consists of B4C compacts in an 800H alloy sleeve. Boron-10 enrichment of 30–50% 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 2/3 Ring 3 rods and 10/12 Ring 4 rods, showing sufficient shutdown margin for stuck rods or accidental rod withdrawals.

3.2.2.4 Reactivity of Alternative Loop Test Fluids. An important mission of the TR is the irradiation of a variety of primary coolant fluids from alternative reactor technologies. Alternative primary coolant fluids may include light water, liquid salt, liquid metal, and gases or steam. To evaluate the reactivity impact to the TR core, 38-L volumes of pressurized light water, FLIBE, and liquid sodium were separately evaluated based on assumed placement in the central irradiation loop facility. Table 5 gives the negative core reactivity incurred for each fluid inserted in place of the primary helium coolant in the central test loop (Ring 1).

Table 5. Test fluid reactivity impact.

Fluid Type	Fluid Symbol	Temperature (°C)	Pressure (MPa)	Reactivity ($)
Helium gas (primary coolant)	He	650	7.0	—
Liquid sodium	Na	300–600	1.01E-4	−0.17
Light water	H_2O	329	15.0	−1.57
Liquid salt-FLIBE[a]	LiF_2-BeF_2	548	0.3	−0.21

a. ^{7}Li enrichment = 99.99%

The introduction of these three test fluids has minor reactivity impact to the core overall. Other fluids, especially liquid salts, would also be easily accommodated in the central loop facility. The introduction of gases or steam into the central loop would also be a small or negligible reactivity effect.

3.2.3 Physics Parameter Summary

Table 6 provides a summary of the key reactor physics parameters related to the baseline design.

Table 6. Summary of reactor physics parameters.

Fuel columns	12
Fuel blocks per column	8
Fuel blocks per core	96
Fuel block type:	Prismatic hexagonal
- Height	79.3 cm
- Flat-to-flat width	36 cm
Fuel rods per fuel block	210
Coolant channels per fuel block	108
Poison rods per fuel block	0
Compact diameter	1.245 cm
Compacts per block	3,126
Compacts per core	300,096
Compact binder matrix graphite density	1.70 g/cm^3
Particle UCO kernel diameter	425 m
Particle PF	15.0%
Particles per compact	2,706
Particles per core	812M
^{235}U enrichment	15.5 wt%
^{235}U loading per fuel block	525.6 g
^{238}U loading per fuel block	2,865.5 g
Power cycle length	110 days
Maximum fuel burnup	9.56% FIMA
Maximum thermal flux	3.90E+14 n/cm^2/s
Maximum fast flux	1.17E+14 n/cm^2/s
Maximum fast fluence	1.72E+21 n/cm^2
Peak fuel temperature under normal operation	<1250°C
Average particle power	246 mW
Temperature coefficients of reactivity:	—
– Isothermal	−9.7 to −9.5 pcm/°C (20–1000°C)
- Fuel	−4.9 to −2.4 pcm/°C (20–2500°C)
- Moderator	−3.2 to −5.2 pcm/°C (20–1800°C)
Control rods	3 (Ring 3) + 12 (Ring 4)
Control rod worth	\$50.2
Beta-effective	0.0073
Neutron generation time	642 s

3.3 Core Thermal Hydraulics

The RELAP5-3D computer code[4] was used to calculate the thermal-hydraulic conditions. Coolant flow enters near the bottom of the reactor vessel cylinder, flows up through the annulus between the core barrel and reactor vessel, then enters the upper plenum. Helium then flows down through a number of parallel channels in the core: the coolant holes in the fuel blocks, the gaps between the hexagonal blocks, the gap between the permanent side reflector and the core barrel, and gaps between the graphite reflector blocks and the control rods or irradiation tubes. These flow paths all meet in the lower plenum, from which the coolant exits the reactor vessel.

Generally accepted criteria for TRISO fuel are peak temperatures below 1250°C during steady-state operation. 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. Figure 7 shows the peak fuel temperatures at different powers for these three different gap widths between the blocks. Peak fuel temperatures are below the limit for powers up to 200 MW. Higher powers, and thus higher fluxes, could be tolerated early in the core life when the gaps are smaller.

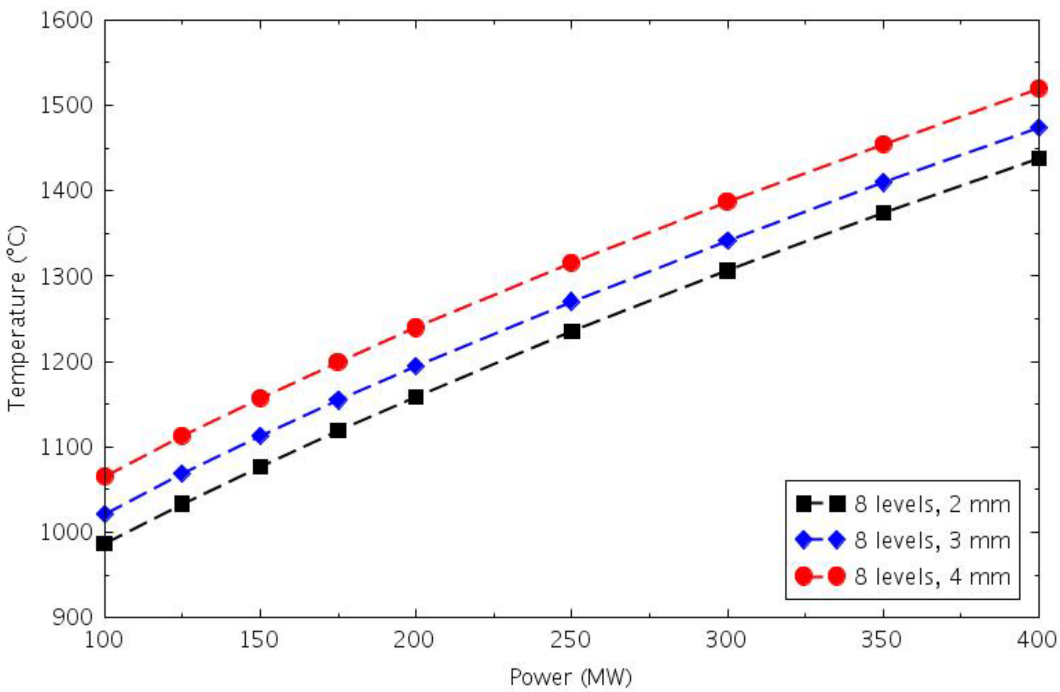


Figure 4. Steady-state peak fuel temperature versus core power.

Calculated steady-state thermal-hydraulic conditions are provided in Table 7. 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. The peak fuel temperatures increase with larger block-to-block gaps, as more flow bypasses the coolant channels. The larger bypass flows also result in generally lower reflector temperatures. The insensitivity of the PSR, core barrel, and reactor vessel temperatures-to-changes in the fuel temperature indicate that these structures are driven by the coolant inlet temperature rather than by radial heat transfer from the fuel.

Table 7. Steady-state conditions for 8-level, 200-MW core.

Parameter	2-mm Gaps	3-mm Gaps	4-mm Gaps
Coolant inlet temperature (°C)	325	325	325
Coolant outlet temperature (°C)	650	650	650
Coolant flow rate (kg/s)	117.2	117.3	117.3
Effective core bypass at core inlet (%)	26	29	33
Effective core bypass at core outlet (%)	27	31	35
Peak fuel temperature (°C)	1159	1194	1240
Center reflector peak temperature (°C)	648	645	651
Ring 3 reflector peak temperature (°C)	585	567	558
Ring 4 reflector inner peak temperature (°C)	562	550	548
Ring 4 reflector outer peak temperature (°C)	392	383	380
Ring 5 reflector peak temperature (°C)	357	348	343
PSR peak temperature (°C)	336	332	331
Core barrel peak temperature (°C)	329	328	328
Reactor vessel peak temperature (°C)	317	317	317
RCCS heat removal (MW)	0.44	0.44	0.44
Irradiation loop heat removal (MW)	0.17	0.17	0.17

3.4 Other Systems

The design effort thus far has been focused on the core and reactor vessel; design of most of the ex-vessel systems has not been addressed yet. Except for test-specific systems and hardware, all systems and components are prototypic of either a commercial or demonstration gas-cooled prismatic block reactor. The fuel and reflector blocks are full size. The reactor vessel and steam generator will be somewhat smaller because of the lower core power. The PCS consists of the reactor vessel, a hot duct or pipe leading to a steam generator, a compressor or gas circulator, and a cold return through an outer annulus around the hot duct to the reactor vessel inlet. An alternate parallel path to provide high temperature helium to a process heat application could also be included, if desired.

The secondary coolant system is expected to include a turbine-generator set to generate electricity and a condenser. For licensing reasons, not all of the available steam would be used for electricity generation. The remaining steam could be used to drive the process heat test bed or could simply be sent directly to the condenser. The steam generator is expected to be used for normal decay heat removal. Alternatively, a dedicated cooling system could be included like those in commercial plant designs in which helium is drawn from the bottom of the reactor vessel, cooled, and returned to the vessel.

Functional containment of fission products is provided by the fuel particles, compacts, and interstitial graphite. Therefore, a reactor building is preferable to a full containment structure. In the event of a large break in the primary coolant boundary, the reactor building sis vented to relieve pressure while the core cools down and mitigating actions can be taken. The levels of radiological materials that accumulate in the primary circuit during operation are low enough to be vented to the atmosphere without exceeding site boundary limits even though much of the material released from the circuit will remain within the building. The reactor vessel is expected to be in one compartment or cavity, and the steam generator and compressor/circulator in another.

A reactor cavity cooling system (RCCS) will be used for passive, long-term decay heat removal if normal active systems are unavailable. It consists of a series of cooling panels located around the reactor vessel. Water flowing on the inside of the panels removes the heat transferred to the outside of the panels by radiation and convection from the reactor vessel. A large pool of water located higher than the reactor cavity, but not directly above it, provides water to the bottom of the cooling panels and receives the warmer water exiting the top. Flow through the panels is driven by natural convection, with no pumps required. The pool could be actively cooled during normal operation, and could be designed to be cooled by air natural circulation when active cooling is not available. Given the relatively small amount of heat removed by the RCCS during steady state and the large thermal margins shown in the accident simulations in Section 4, an air-cooled system may also be a viable option.

Detailed instrumentation and control systems have not been considered yet. However, it is anticipated that the plant instrumentation and controls would be those required to ensure safe and efficient operation of the reactor and associated systems, and to satisfy NRC's requirements to monitor the in-core conditions within a reasonable level of accuracy. Additional instrumentation that would be used strictly for computer code assessment will not be included. Instrumentation needed to control a specific experiment, such as thermocouples and flow control valves, would be provided with that experiment. Except for the center irradiation loop location, the test facilities have all been shown as off-center in the reflector blocks, so that standard fuel and reflector block handling tools can be used. For the center loop, either the loop could be moved to one side of the block, or special handling equipment could be developed to move the reflector blocks.

The experiment loops would most likely be located in experiment wells. The well is attached to the upper head, open at the top, closed at the bottom, forming the PCS pressure boundary. The experiment loop is then lowered into the well, with its piping and associated components (pumps, heaters, etc.) located outside the pressure vessel. Helium flow would be provided between the experiment loop and the well to provide cooling to the well wall.

4. TEST REACTOR 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.

4.1 Safety Characteristics

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 its use in this reactor is within the fuel qualification envelope. Should 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 PCS and reactor building provide 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. It also means that there would be no adverse coolant interactions should a leak develop from an experiment irradiation loop. 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 PSR.

The steam generator is expected to be used for normal decay heat removal. Alternatively, a dedicated cooling system could be included like those in commercial plant designs in which helium is drawn from the bottom of the reactor vessel, cooled, and returned to the vessel. While these active systems would be used for convenience, they are not required.

Decay heat removal can be accomplished 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 RCCS. Flow through the RCCS is provided by natural convection from a large pool located higher than the reactor cavity.

4.2 Safety Performance

Generally accepted criteria for TRISO fuel are peak temperatures below 1250°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 will be shown below, the peak transient temperatures are lower than those during steady-state, so it may be possible to increase the power and flux during steady-state, should further fuel testing show that operating temperatures above 1250°C result in no challenges to the fuel integrity.

The operational events and accidents for this 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 TR 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 steam generator tube rupture. 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, although failure of both the experiment pressure boundary inside the reactor vessel and the PCS boundary in the same test well may fall into the beyond design basis event realm, as it involves independent failures of two American Society of Mechanical Engineers pressure boundaries.

Total losses of forced convection cooling are referred to as conduction cooldown transients, as the heat in the core is conducted (and radiated) to the reactor vessel and then to the RCCS. In a pressurized conduction cooldown, the PCS pressure boundary remains intact. In a depressurized conduction cooldown (DCC), the PCS is depressurized; the general assumption is a loss-of-coolant accident. 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.

Figure 8 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. Average reflector temperatures over the fueled length are shown in Figure 9 for the 4-mm gap case. The central reflector temperatures are very close to the fuel temperatures, with temperatures steadily decreasing moving radially outward toward the reactor vessel. Figure 10 presents the peak reactor vessel wall temperatures. The peak temperature increases about 100°C during the transient.

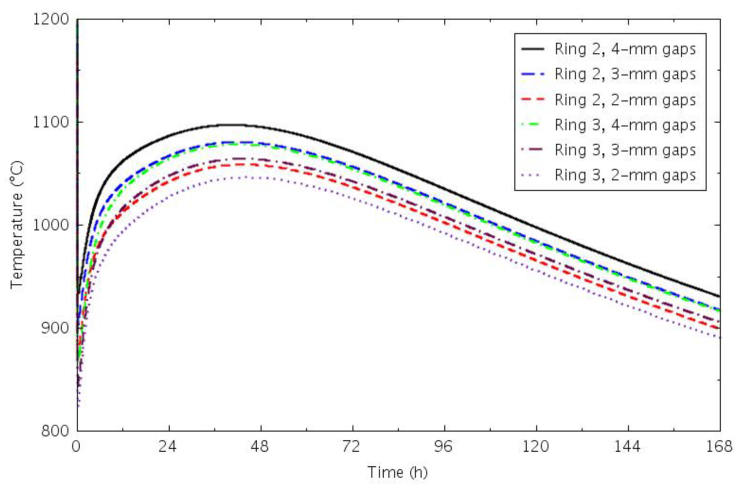


Figure 5. Peak fuel temperatures for depressurized conduction cooldown transient.

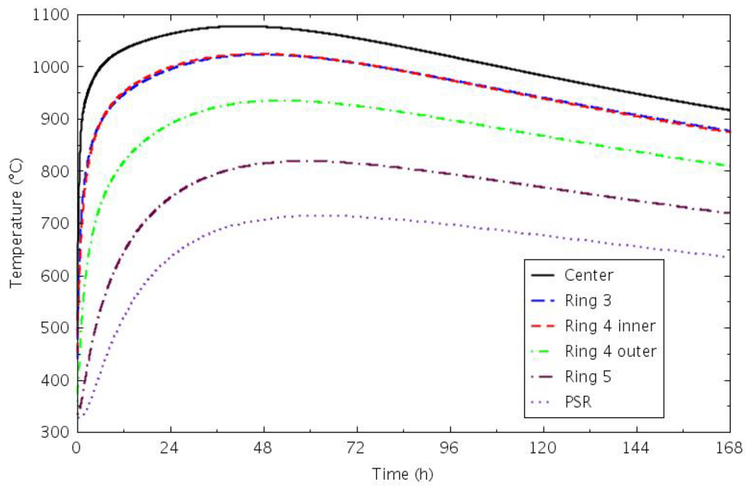


Figure 6. Axial average reflector temperatures for depressurized conduction cooldown transient with 4-mm gaps.

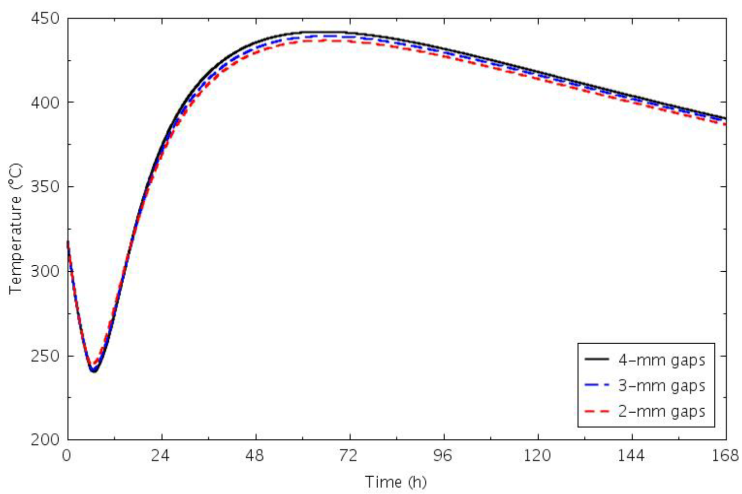


Figure 7. Peak reactor vessel wall temperatures for depressurized conduction cooldown transient.

The DCC simulations were insensitive to the axial power profile, a 1- or 10-s delay in reactor scram, blocking some of the bypass paths, changing the temperature of the helium flowing back in through the break, or modeling a double-ended break instead of a single-ended break. Increasing the steady-state coolant temperatures to 350°C inlet and 750°C outlet resulted in about a 20°C increase in the peak temperatures. The largest impact on the fuel temperature was when cooling flow was provided inside the center irradiation tube; this also reduced the temperature of the central reflector, while leaving the outer reflector temperatures unchanged.

The pressurized conduction cooldown 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 Figure 11, are about 100°C lower than those in the DCC transient; reductions in temperatures were observed in the other structures as well.

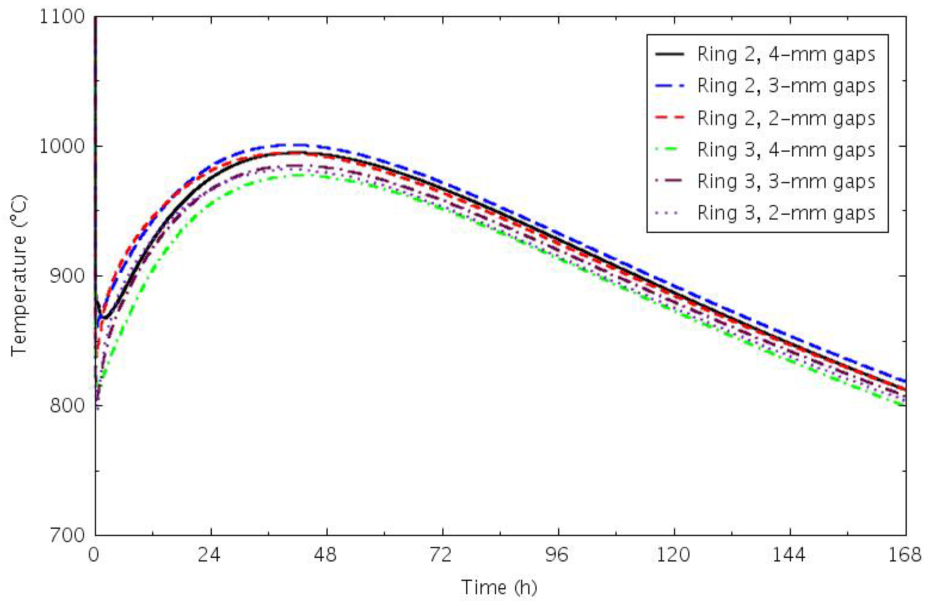


Figure 8. Peak fuel temperatures for pressurized conduction cooldown transient.

One concern identified during the transient simulations is the condition of the irradiation tubes. Without an internal cooling mechanism, the tubes will be at about the same temperature as the reflector blocks in which they are located. For flow-through tubes, this is not an issue, as they do not contain internal pressure and can be made of higher melting point materials such as titanium or molybdenum. Pressure-containing irradiation facilities, however, will likely have to be constructed of steel. While the transient temperatures shown in Figure 9 are well below the steel melting point, many are above the design temperature for pressure-bearing systems. Some cooling would need to be provided to these facilities during the cooldown transients to ensure their continued viability, or they may need to be replaced should such an accident occur, depending on their location and temperature histories.

5. TECHNOLOGY READINESS OF TEST REACTOR CONCEPT

The timeline (and cost) for design and deployment of a TR is strongly dependent on the technical maturity of the reactor technology. Of all of the non-LWR concepts proposed in recent decades for power and other applications, the HTGR is among the most technically mature. Indeed HTGRs supplied power to the electric grid in the U.S. (FSV) and in Germany (Thorium Hochtemperatur Reaktor) in the 1980s. More modern engineering-scale HTGRs employing passive decay heat removal are operating in Japan (high-temperature test reactor) and China (HTR-10).

As with the Power Reactor Innovative Small Module reactor, the U.S. Government collaborated with industry to develop a small, modular version of the high-temperature reactor (HTR) in the 1980s. The GA modular HTGR (MHTGR) was subjected to pre-application safety review by the NRC. (Fuel development by the federal government continued under the New Production Reactor Program, which had as its mission the production of tritium for the weapons program.) An important difference between the MHTGR and FSV plant was the lower core power density enabling the ability to reject decay heat passively to the environment even in the most severe loss of coolant accident. This is achieved by limiting the core power (about 600 MW) and building a tall core with a relatively small diameter, thus providing a short conduction path from the core to the vessel. All HTRs designed since then have adopted this inherent safety feature. The detailed technical status of the MHTGR and its pebble bed counterpart developed in Germany, the HTR Modul, are described in Reference [5]. The HTR Module design was submitted to the German regulator in the late 1980s, but was never built. A two-unit pebble-bed modular HTR power plant based on the German design is under construction in China.

As part of the overall Advanced Test/Demonstration Reactor Options Study, an assessment of the maturity of Generation IV reactor technologies was conducted by a multi-laboratory panel of experts and documented in Reference [6]. Panel's assessment results of the HTGR are summarized in this section.

A technology readiness scale developed by DOE was used to evaluate the HTGR and other systems. The numerical scores for the different systems and subsystems of the HTGR are tabulated in Table 8. The scale, using the DOE-defined readiness levels as described in Reference [5], ranges from 9 for technologies with operational experience down to 1 where the technologies basic principles have been observed and reported. The overall TRL was obtained by taking the minimum value of the TRLs of the key subsystems shown in shaded cells in Table 8. Key subsystems are those that were determined by the panel to be critical to safe operation and performance of the concept. A more detailed explanation of the scoring process and rationale for the score is provided in Reference [6].

Table 8. Technology readiness levels for each high-temperature gas-cooled test reactor system and subsystem for test reactor deployment (key subsystems are shaded).

System	HTGR TR
Nuclear Heat Supply	5
Fuel element (fuel, cladding, assembly)	6
Reactor internals	6
Reactivity control	6
Reactor enclosure	5
Operations/inspection/maintenance	5
Core instrumentation	6, 3
Heat Transport	5, 3
Coolant chemistry control/purification	6
Primary heat transport system (hot duct)	6
Intermediate heat exchanger (if applicable)	NA[a]/3

Table 8. (continued).

System	HTGR TR
Pumps/valves/piping	5
Auxiliary cooling	6
Residual heat removal	5
Power Conversion	6
Turbine	7
Compressor/recuperator (Brayton)	NA
Reheater/superheater/condenser (Rankine)	7
Steam generator	7
Pumps/valves/piping	6
Process heat plant (e.g., H_2)	NA/3
Balance of Plant	6
Fuel handling and interim storage	6
Waste heat rejection	7
Instrumentation and control	6
Radioactive waste management	6
Safety	6
Inherent (passive) safety features	6
Active safety system	6
Licensing	3, 6[b]
Safety design criteria	3
Applicability of previous licensing experience	3
Safety and analysis tools	4
Fuel Cycle	NA
Recycled fuel fabrication technology	
Used fuel separation technology	
Safeguards	3
Proliferation resistance—intrinsic design features (e.g., special nuclear material accountability)	3
Plant protection—intrinsic design features	3
Overall Technology Readiness Level	**5**

a. NA = not applicable.

b. The SDC have recently been further developed based on a broad set of industry and DOE inputs and comments (NRC feedback on this is due out later in 2016). A TRL of 6 is suggested as a more accurate reflection of this status, but the TRL of 3 is retained to be consistent with the detailed TRL assessment that is reported in Reference [6].

For the HTGR design, the lowest technical maturity scores were assigned to certain metallic components inside the pressure vessel. When exposed to core conditions under accident conditions, these may be subjected to failure. If coolant temperatures are limited to 850°C, SA508/533 (the steel alloy used in LWRs) is adequate for the pressure vessel. Metallic control rod drive tubes and seals, however, may fail in the event of the most severe loss-of-forced-cooling events, with subsequent depressurization of the core. While this is not expected to cause significant fuel particle degradation, occupational exposure, or releases to the environment, the circulating radiological inventory could be released and expensive core repairs would be necessary. Qualification of new alloys or even the use of carbon or silicon carbide composites for the guide tubes may be needed. The control elements are not anticipated to reach failure temperatures. For these reasons, the reactor enclosure subsystem for the demonstration plant was assigned a TRL of 5.

The overall conclusion of the panel was that the HTGR, with outlet temperatures limited to 850°C, is suitable for near-term deployment as either a test or demonstration reactor.

6. TEST REACTOR LICENSING, DEVELOPMENT, AND DEPLOYMENT PLANS

The HTGR-TR aligns with the NRC's definition of a Test Facility (TR), as found in 10 CFR 50.2[7]:

> *A thermal power level in excess of 1 megawatt, if the reactor is to contain a circulating loop through the core in which the applicant proposes to conduct fuel experiments.*

TRs are one of the types of non-power reactors that the NRC license under the authority of Subsection 104c of the Atomic Energy Act, and are therefore issued "Class 104c" licenses. An additional restriction regarding this class of reactor license is that the facility must be used so that no more than 50%t of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than R&D or education or training.

Congress directed the NRC to impose the minimum amount of regulation on Subsection 104(c) research reactor and TR licensees. In keeping with this direction, the NRC staff utilizes NUREG-1537[8] as the primary guidance for review of research reactors and TR technologies and license applications. NUREG-1537 is based on the key constituents from 10 CFR 50, and use of the historically-applied "two-step" NRC licensing is summarized in Subsection 6.3.

NRC's research reactor and TR reviews are typically performed using a "performance-based" approach, rather than the more prescriptive-review approach used when licensing commercial power reactors under the NRC's Standard Review Plan for LWRs.[9] This approach is expected to be characterized by the establishment and implementation of flexible and tailored technical requirements specific to the advanced technologies being developed.

6.1 Test Reactor Dose Limits

The occupational and public dose limits for TRs are found in 10 CFR 20.1201 and 10 CFR 20.1301; accident doses for workers should be compared with the limits found in 10 CFR 100.[10],[11] These limits are summarized Table 9.

Table 9. Dose limits applicable to the high-temperature gas-cooled test reactor.

TRs—Occupational, Public, and Accident Dose Limits			
Applicability	Requirement Document	Dose Limit	
Occupational Dose—Annual Limit	10 CFR 20.1201	TEDE	≤5 rem
		Organ dose	≤50 rem
Public Dose Limit	10 CFR 20.1301	Annual TEDE	≤100 mrem
		Hourly external dose	≤2 mrem
Accident Dose Limit—Worker	10 CFR 100.11	Whole body	≤25 rem
		Thyroid dose	≤300 rem
Key: TEDE = total effective dose equivalent.			

6.2 Design Criteria for Modular High-Temperature Gas-Cooled Reactors

DOE and NRC established a joint initiative in July 2013 to develop guidance for advanced reactor developers and other stakeholders on how the existing General Design Criteria (GDC) reflected in 10 CFR 50, Appendix A, can be adapted to non-LWRs. A proposed set of GDC adaptations specific to modular HTGRs was developed by a DOE/national laboratory team and submitted to NRC for review in December 2014.[12]

Initial NRC feedback on those adaptations is pending, and expected to ultimately be reflected in an NRC Regulatory Guide that will be formally issued in late 2016. These adapted criteria provide guidance and direction to be considered by both TR developers and the NRC staff reviewing the related license application(s).

6.3 Research and Development Needed for Licensing

The Advanced Reactor Technologies (ART) Program team has recently reviewed and discussed the R&D needed to support commercial licensing for the modular HTGR and sodium fast reactor advanced reactor design types. Results of those efforts are reflected in the ART Regulatory Technology Development Plan.[13] The Regulatory Technology Development Plan identifies, assesses, and prioritizes key ART research opportunities with respect to their associated regulatory impact, and recommends research priorities that specifically consider the needs of the NRC independent safety review process. Although this plan focused on supporting development and deployment of commercial facilities, it can be used as a resource for identifying key R&D needed to support TR licensing, since many of the issues to be addressed are largely the same. Key R&D topical areas identified include:

- Accident progression modeling
- Primary system and containment performance
- Fission product behavior modeling
- Core heat removal
- Thermal-fluid dynamics
- Nuclear analysis
- Fission product transport
- Event sequence frequency.

In addition to the above key R&D topical area, it will be necessary to identify and assess any adverse effects that may result from the interactions between the TR and associated test mediums (e.g., molten and salt), and from feedback/interaction from "over-the-fence" TR output end-users.

6.4 Test Reactor Deployment Schedule

The NRC's license application review process for research reactors and TRs is built around the 10 CFR 50 ("two-step") licensing process, consisting of series applications for a construction permit (CP) followed by an operating license. Although no NRC regulations apply specifically to the submittal of a safety analysis report (SAR) for non-power reactors, the SAR format has historically been used in direct support of this process and is expected to provide the most efficient and effective method for facilitating the NRC's review of the TR discussed in this report.

The application for a CP must contain the following three types of information:

1. Preliminary safety analyses
2. An environmental review
3. Financial and anti-trust statements.

NRC also conducts an environmental review, in accordance with the National Environmental Policy Act, to evaluate the potential environmental impacts and benefits of the proposed TR facility.

Final design information and plans for operation are developed during construction of the facility. The applicant then submits an application to NRC for an operating license. The application contains a final SAR, an updated environmental report, and a description of the plans for operation, including technical specifications. The review process is similar to that applied to the CP application (Figure 12), but may exclude certain steps (hearing, etc.) depending on petitions received.

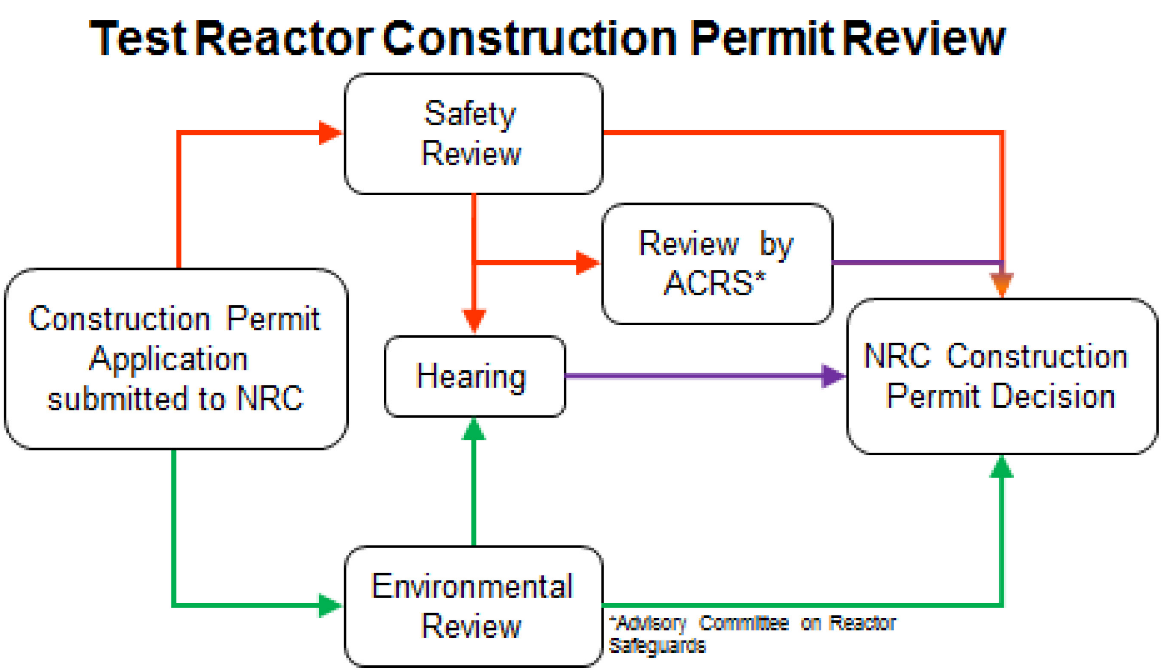


Figure 12. Nuclear Regulatory Commission test reactor construction permit review process.

The design, procurement, and construction processes follow a system engineering philosophy of progression. This includes the conceptual design (2 years), preliminary design (2 years), final (detailed) design (3 years). Key activities in these areas are summarized below, and a representative schedule is reflected in Figure 13:

- ***Conceptual Design.*** The purpose of conceptual design is to fully define the selected concept. Feasibility questions have been adequately addressed (sometimes subject to future ongoing development activities with known fallback options), and key design decisions have been made. The product of conceptual design is a moderately detailed plant design concept for which all main features have been identified and key analyses have been performed. This information is used to support pre-application discussions with the NRC, including the establishment of the licensing technical requirements that will be applied to the NRC's review of the license application.
- ***Preliminary Design.*** The purpose of preliminary design is to advance the design process to the point of project commitment to manufacture/construction. Any adjustments needed as a result of conceptual design work are made and the second design iteration is performed. In general, the design activity during preliminary design is more complete and thorough, using more complete design information and more mature analysis tools and models. The preliminary design would form the basis for the bulk of the content of the SAR submitted to the NRC in support of the CP application.
- ***Construction and Startup Testing.*** Construction activities can begin once a plant CP has been obtained. A CP is granted based on NRC review of the preliminary SAR. In accordance with our schedule, this review is to last a maximum of 3 years and conclude as the plant final design phase comes to completion.

The reference TR plant construction schedule is 6 years. At the end of the sixth year, the TR is installed and ready for fuel loading and shake-down phase operation. Initial operating license conditions or restrictions (e.g., reduced power level) are addressed during this shake-down phase.

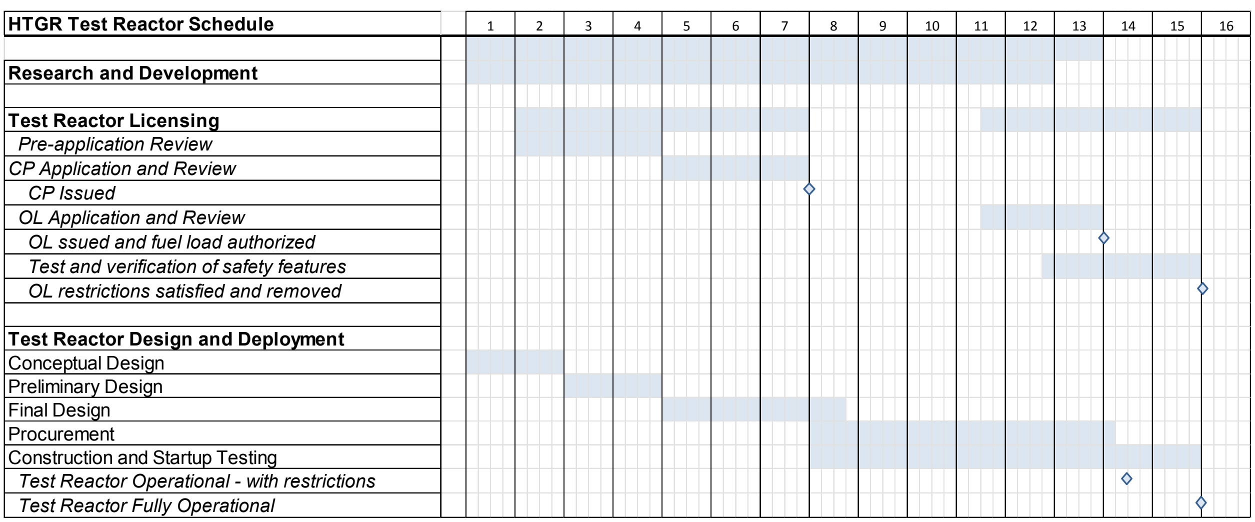


Figure 13. High-temperature gas-cooled test rector design, licensing, and deployment timeline.

7. ECONOMICS

The capital, operating, and decommissioning costs for the HTGR-TR are based on the information presented in the NGNP pre-conceptual design report[1] for a 350-MW FOAK reactor with a single reactor module. The detail cost model utilized for this cost estimate was developed as part of the NGNP Project using data from three vendors, as described in Reference [14]. Unless otherwise stated, all costs are presented in 2015 dollars and are scaled using the Chemical Engineering Plant Cost Index. The total capital cost for an HTGR includes preconstruction costs, direct costs, indirect costs, and project contingency. Operating costs include staffing requirements, annual fees, insurance, taxes, material supplies, outage costs, and administration and general cost overhead.

The AREVA SC-HTGR Demonstration Reactor Report estimated a total overnight cost of $3,963 million for the 625-MW design, while the NGNP cost model produces an estimate of $3,669 million, based on a 600-MW Rankine-cycle FOAK design.[15] The two estimates are based on two slightly different designs, but are still within 8% of each other, well within the uncertainty range of both estimates. Hence, it can be concluded that the NGNP cost model for the large (600 MW) design still represents a valid benchmark point for the estimates that are presented in this section.

The cost estimate for the 200-MW TR is based on the 350-MW NGNP demonstration reactor data, since this was the closest match to the selected power level. The NGNP cost model included several complex scaling factors for calculating the cost of a scaled 350-MW design from the larger 600-MW design that was considered as the reference NGNP design, and further scaling down to 200 MW was not recommended by the AREVA review team or INL personnel involved in the NGNP cost model, since the uncertainties related to the cost trade-offs of the higher complexity of the TR design versus a simpler power reactor design are significant. The extrapolating of the NGNP correlations developed for the 600- to 350-MW costs can also not be measured using existing vendor data. The 350-MW costs were therefore applied “as-is” for the estimate for the 200-MW TR costs.

The best-estimate (point) cost of the 200-MW TR design is summarized in Table 10, together with the costs estimate uncertainties of −50%[1] and +50%, which is consistency with the level of project definition of 0 to 2% (an AACE International Class 5 estimate). The TCI is calculated by summing the preconstruction costs, direct costs, indirect costs, and project contingency. The TCI required to build a 200-MW HTGR test reactor is estimated at $3,942 million, within an uncertainty range of $1,972–5,913 million.

The following assumptions were used to determine these costs estimates:

- ***Reactor outlet temperature.*** The reactor outlet temperature of the HTGR-TR is selected at 650°C, but the correlations available in the NGNP cost model are only specified between 750 and 950°C. An extrapolation down to 650°C was not recommended, due to possible non-linear variances in material costs when the lower temperatures are utilized. The lowest available model point, 750°C, was therefore selected for the 200-MW TR estimate and provides justification for decreasing the lower bound of the cost-estimate uncertainty range.
- ***Non-scaling factors.*** The NGNP cost model[1] shows that building, support facilities, licensing and design costs components do not scale with a change in reactor power, since the activities and durations required for a large or small reactor design is comparable. These components make up $1,325 million of the total cost (34%).

---

1. A relatively lower bound of the cost-estimate uncertainty of −50% was selected for the cost study to reflect the lower power level of the TR (200 MW) as compared to the size of the reactor used for the point estimate (350 MW). This lower bound is still within the acceptable range of cost-estimate uncertainty.

Table 10. Summary of lower, best-estimate, and upper cost estimates for 200-MW first-of-a-kind high-temperature gas-cooled test reactor

Item	−50%	Best Estimate	50%
Capital Cost summary (Million 2015 $)			
Preconstruction costs			
Land and land rights	0	0	0
Licensing and application costs	122	244	366
Total preconstruction costs	122	244	366
Direct cost			
Selected configuration direct cost total	447	894	1,341
Balance of equipment adder	112	224	336
Test loops/facilities	100	200	300
Total direct cost	659	1,318	1,977
Indirect costs			
Total design cost	540	1,081	1,621
Construction services	112	224	336
Home office and engineering services	90	180	270
Field office and engineering services	55	109	164
Owner's costs	64	129	193
Total indirect cost	861	1,723	2,585
Construction cost	1,642	3,285	4,927
Project contingency	328	657	985
Total Capital Investment	**1,971**	**3,942**	**5,913**
Yearly Operations and Maintenance (O&M) Summary (Million 2015 $)			
Total yearly O&M cost	18	36	54
Fuel Cost Summary (Million 2015 $)			
Refueling cost per core	30	60	90
Total average yearly fuel cost	20	40	60
Decommissioning Cost Summary (Million 2015$)			
Total decommissioning cost	39	78	117

- ***Cost of additional test features and increased complexity.*** The 200-MW TR includes several features that are not present in the 350-MW demonstration reactor: test loops, irradiation positions and the peripheral supporting elements (pumps, vessel head penetrations, extraction and sample storage space, etc.). The cost of these features cannot be assessed within the scope of this point design study, but an estimate of $200 million has been added to account for this increase in design complexity. This is roughly based on available data from the INL ATR Program, where the addition of a single gas test loop to the current design has been estimated at approximately $80 million. If it is assumed that the addition of four test loops would achieve some economy of scale, $200 million could be an approximate estimate of this factor, possibly within the same −50% and +50% uncertainty band as the rest of this estimate.

For each of the major cost categories, a short summary of the major contributors is provided below.

7.1 Capital Costs

The total capital cost is comprised of the cost categories addressed in the following subsections.

7.1.1 Preconstruction Costs

The main contributor to the preconstruction cost category is NRC licensing and application. Design certification costs are included in the licensing costs for the test plant. Costs for land and land rights costs are normally included in this cost category, but have been assumed to be zero for this analysis because the proposed reactor would be constructed on government land. Generally, it may be assumed that a single HTGR requires 50 acres of land.

7.1.2 Direct Costs

Direct costs for the reactor plant are associated with the cost of materials and installation for the equipment items that make up the reactor plant. Given the nascent stage of the HTGR design, previous reactor cost estimates provided by selected reactor design suppliers were assessed to determine the reactor plant equipment items that make up the majority of the direct costs. The following equipment items make up approximately 80% of the installed equipment costs: reactor building, vessel, initial core, metallic and graphite internals, reactor cavity cooling system, core refueling equipment, heat rejection system, intermediate heat exchanger, and the power generation equipment (Rankine cycle assumed). (The TR will most likely not include an intermediate heat exchanger, but an estimation of the cost for this component is beyond the scope of this work). To account for the balance of equipment costs not included in the direct cost estimate, the total cost of the items above were multiplied by a factor of 1.25. This factor was based on the assessment of previous cost estimates provided by the reactor design suppliers, in which the remaining equipment items contribute 20% of the installed equipment costs.

7.1.3 Indirect Costs

The capital required for construction overhead and other costs not included in the direct costs are included in the indirect costs. Given the early stage of HTGR design and costing efforts, it is necessary to estimate the indirect costs as a percentage of the direct costs based on previous reactor design supplier estimates and historical indirect costs for LWR designs. Indirect costs cover the following:

- Construction services: construction management, procurement, scheduling, cost control, site safety, and quality inspections
- Home and field office and engineering services: costs for estimating, scheduling, project expediting, project general management, design allowance, field office, field engineering, field drafting, field procurement, and project fees
- Owner's costs: project fees, taxes, and insurance; spare parts and other capital expenses; staff training and startup costs; and administrative and general expenses
- Design costs: covers the conceptual, preliminary, and final design activities, as well as R&D activities associated with these stages of reactor design and licensing. A breakdown of these components is provided in Table 11.
- R&D costs cover the R&D needed to increase the TRLs for each HTGR system.

Table 11. Design costs.

Phase	$ Million
Conceptual design	90
Preliminary design	194

Final design	315
R&D costs	482
Total	**1,081**

7.1.4 Contingency

A project contingency of 20% was selected for the HTGR capital cost analysis for all project phases.

7.2 Operating Costs

Operating costs are estimated for the HTGR-TR and include O&M costs and refueling costs. O&M costs were estimated based on methodology presented for a study of advanced reactor technologies, specifically the study of O&M staffing and costs conducted by nuclear industry partners. O&M costs are assumed to include staffing requirements, fees, taxes and insurances, material supplies, outage costs, and administration and general cost overhead.

HTGR fuel costs were calculated for the prismatic fuel configuration provided by GA for the NGNP pre-conceptual design.[1] It is assumed that the refueling cost is scaled linearly with the reactor power rating.

7.3 Assessment of Potential Revenue

A preliminary assessment of the potential revenue and resulting economic considerations associated with construction and operation of an HTGR was performed. The analysis assumed that 50% of the heat generated by the HTGR would be available to generate electricity, due to the NRC license restrictions on TRs. The remaining heat can be used for experimental setups or just dumped to the final heat sink. The results of the assessment are summarized in Table 12.

Table 9. Possible revenue generation for a high-temperature gas-cooled test reactor with Rankine cycle to generate electricity.

Selling Price of Electricity[a,]		Estimated Electrical Sales Revenue Generated[b.] ($M)
Area	($/MW)	
Northwest U.S.	22.75	6.2
Northeast U.S.	44.36	12.1

a. Selling price of electricity based on average January 2016 spot price from the Energy Information Administration website (www.eia.gov).[16] The highest and lowest U.S. prices are presented in the table to illustrate the electricity selling price range.

b. Annual revenue from electricity sales for a 200-mW test reactor with a capacity factor of 78%, 50% of reactor power rating used to generate electricity, 39.8% efficiency of 550°C Rankine steam cycle to produce electricity (see Reference [17] and electricity prices as noted in the table).

8. REFERENCES

1. INL, 2007, *Next Generation Nuclear Plant Pre-Conceptual Design Report*, INL/EXT-07-12967, Idaho National Laboratory, November 2007.

2. DOE, 1986, *Preliminary Safety Information Document for the Standard MHTGR*, Stone & Webster Engineering Corp., HTGR-86-024, 1986.

3. X-5 Monte Carlo Team, 2003, *MCNP—A General Monte Carlo N-Particle Transport Code, Version 5*, Volume I (LA-UR-03-1987) and Volume II (LA-CP-03-0245), Los Alamos National Laboratory, Los Alamos, New Mexico, April 2003.

4. INL, 2015, RELAP5-3D Code Development Team, *RELAP5-3D Code Manual*, Volumes I–V, INL/MIS-15-36723, Rev. 4.3, October 2015.

5. Gougar, H., 2014, *Baseline Concept Description of a Small Modular High Temperature Reactor*, INL/EXT-14-31541, Rev. 1, Idaho National Laboratory, May 2014.

6. Gougar, H.D., et al, 2015, *Assessment of the Technical Maturity of Generation IV Concepts for Test and Demonstration Reactor Applications*, INL/EXT-15-36427, Idaho National Laboratory, Rev. 2, October 2015.

7. 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*, U.S. Nuclear Regulatory Commission.

8. NRC, 1996, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, NUREG-1537.

9. NRC, 2007, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*, NUREG-0800.

10. CFR 20, *Standards for Protection Against Radiation*, U.S. Nuclear Regulatory Commission.

11. 10 CFR 100, *Reactor Site Criteria*, U.S. Nuclear Regulatory Commission.

12. J.C. Kinsey, M. R. Holbrook, 2014, *Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors*, INL/EXT-14-31179, Rev. 1, December 2014.

13. J.C. Kinsey, 2015, *Advanced Reactor Technology – Regulatory Technology Development Plan (RTDP)*, INL/EXT-14-32837, Rev. 0, May 2015.

14. Gandrik, A.M., 2012, *Assessment of High Temperature Gas-Cooled Reactor (HTGR) Capital and Operating Costs*, TEV-1196, Idaho National Laboratory, Rev. 1, January 2012.

15. Mayer, J., L. Lommers, and F. Shahrikhi, 2016, *Steam Cycle - High Temperature Gas-Cooled Demonstration Reactor*, AREVA Inc., Technical Data Record 12-9251108-000, January 2016.

16. U.S. Energy Information Administration, Independent Statistics and Analysis, www.eia.gov.

17. Nelson, L., et al. 2011, *Integration of High Temperature Gas-Cooled Reactors into Industrial Process Applications*, Idaho National Laboratory, INL/EXT-09-16942, Idaho National Laboratory, Rev. 3, September 2011.

Appendix A

Self-Assessment Against Test Reactor Metrics

Appendix A

Self-Assessment Against Test Reactor Metrics

Summary of Self-Assessment

Metric	INL Score
T1.1.1	1
T1.1.2	5
T1.1.3	9
T1.1.4	9
T1.1.5	9
T1.2.1	9
T1.2.2	9
T1.2.3	9
T2.1.1	5
T2.1.2	5
T2.2.1	5
T2.3.1	5
T3.1.1	9
Total	**89/117 (76%)**

Metric T1.1.1. Fast-flux conditions.

Metric	>5 × 10^{15} n/cm^2-s fast (>0.1 MeV)	5 × 10^{14} to 5 × 10^{15} n/cm^2-s fast (>0.1 Mev)	<5 × 10^{14} fast (>0.1 MeV)
INL Score	—	—	1

Metric T1.1.2. Thermal flux conditions (0.625 eV).

Metric	>5 × 1014 n/cm2-s thermal	1–5 × 1014 n/cm2-s thermal	<1 × 1014 thermal
INL Score	—	5	—

Metric T1.1.3. Irradiation volumes and length for largest test location.

Metric	Volume >10 L Length >2 m	Volume 5–10 L Length 0.5–2 m	Volume <5 L Length <0.5 m
INL Score	9	—	—

Metric T1.1.4. Maximum sustainable time at power, to provide a time-at-power for a single irradiation (i.e., cycle length).

Metric	>90 Days	45–90 Days	<45 Days
INL Score	9	—	—

Metric T1.1.5. Provisions for testing prototypic and bounding conditions (temperature, coolant, chemistry).

Metric	Does the reactor allow for testing at prototypic and bounding conditions?	
INL Score	Yes = 9	—

Metric T1.2.1. Number of test zones.

Metric	>25 Locations	10–25 Locations	<10 Locations
INL Score	9	—	—

Metric T1.2.2. Number and type of distinct irradiation test loops each with a different cooling system independent of the primary reactor coolant.

Metric	3 or More	1 or 2	None
INL Score	9	—	—

Metric T1.2.3. Ability to insert/retrieve irradiation specimen while staying at power.

Metric	At Power (e.g., rabbit)	Limited Handling Capability	Only at Shutdown
INL Score	9	—	—

Metric T2.1.1. Project cost.

Metric	<$2.5 B	$2.5–4 B	>$4.0 B
INL Score	—	5	—

Metric T2.1.2. Construction schedule—time from site preparation to first operation.

Metric	Within 5 Years from Site Preparation	5–10 Years from Site Preparation	Greater Than 10 Years After Site Preparation
INL Score	—	5	—

Metric T2.2.1. Annual operating costs.

Metric	<$100 M/yr	$100–150 M/yr	>$150 M/yr
INL Score	—	5	—

Metric T2.3.1. Availability factor.

Metric	>80%	60–80%	<60%
INL Score	—	5	—

Metric T3.1.1. Number of secondary missions.

Metric	Sale of Energy Products	Other Secondary Missions	None
INL Score	9	—	—