INL/EXT-14-31541
Revision 2

Baseline Concept Description of a Small Modular High Temperature Reactor

Hans D. Gougar

October 2014

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

INL/EXT-14-31541
Revision 2

Baseline Concept Description of a Small Modular High Temperature Reactor

Hans D. Gougar

October 2014

Idaho National Laboratory
Very High Temperature Reactor
Technology Development Office
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

Baseline Concept Description of a Small Modular High Temperature Reactor

INL/EXT-14-31541
Revision 2

October 2014

Approved by:

Hans D. Gougar
Reactor Physics Analysis and Design

10/10/14
Date

Kirk W. Bailey
Quality Assurance

10 10 2014
Date

Diane V. Croson
Advanced Reactor Technologies
Deputy Director

10/10/14
Date

David A. Petti
Advanced Reactor Technologies
Director

10/10/14
Date

EXECUTIVE SUMMARY

The objective of this report is to provide a description of generic small modular high temperature reactors (herein denoted as smHTR), summarize their distinguishing attributes, estimate the technical readiness, and lay out the research and development (R&D) required for commercialization. The generic concepts rely heavily on the modular high temperature gas-cooled reactor designs developed in the 1980s that were never built but for which pre-licensing or certification activities were conducted. The concept matured more recently under the Next Generation Nuclear Plant (NGNP) project, specifically in the areas of fuel and material qualification, methods development, and licensing. As all vendor-specific designs proposed under NGNP were all both 'small' or medium-sized and 'modular' by International Atomic Energy Agency (IAEA) and Department of Energy (DOE) standards, the technical attributes, challenges, and R&D needs identified, addressed, and documented under NGNP are valid and appropriate in the context of Small Modular Reactor (SMR) applications.

The format of the report roughly follows that of the Technical Review Panel report submitted to the Department of Energy (DOE) in September of 2012. Part 1 provides the general background and overview of the HTR, part 2 consists mainly of a table listing the important technical parameters and features of the reference concepts, part 3 includes a table showing the technical readiness of the major systems, structures, and components and the estimated costs to build a plant, and part 4 describes the attributes of the concepts in terms of the Technical Review Panel Criteria specified in the TRP report.

Although the term High Temperature Reactor (HTR) is commonly used to denote graphite-moderated, thermal spectrum reactors with coolant temperatures in excess of 650^{o}C at the core outlet, in this report the historical term High Temperature Gas-Cooled Reactor (HTGR) is used to distinguish the gas-cooled technology described herein from its liquid salt-cooled cousin. Moreover, in this report it is to be understood that the outlet temperature of the helium in an HTGR has an upper limit of 850^{o}C, which corresponds to the temperature to which certain alloys are currently being qualified under DOE's Advanced Reactor Concepts (ARC) program. Although similar to the HTGR in just about every respect, the Very High Temperature Reactor (VHTR) may have an outlet temperature in excess of 850^{o}C and is therefore farther from commercialization because of the challenges posed to materials exposed to these temperatures. The VHTR is the focus of R&D under the Generation IV program and its specific R&D needs are included in this report when appropriate for comparison.

The distinguishing features of the modular HTGR are the refractory (TRISO) coated particle fuel, the low-power density, graphite-moderated core, and the high outlet temperature of the inert helium coolant. The low power density and fuel form effectively eliminate the possibility of core melt, even upon a complete loss of coolant pressure and flow. The graphite, which constitutes the bulk of the core volume and mass, provides a large thermal buffer that absorbs fission heat such that thermal transients occur over a timespan of hours or even days. As chemically-inert helium is already a gas, there is no coolant temperature or void feedback on the neutronics and no phase change or corrosion product that could degrade heat transfer. Furthermore, the particle coatings and interstitial graphite retain fission products such that the source terms at the plant boundary remain well below actionable levels under all anticipated nominal and off-normal operating conditions. These attributes enable the reactor to supply process heat to a collocated industrial plant with negligible risk of contamination and minimal dynamic coupling of the facilities (Figure E-1). The exceptional retentive properties of coated particle fuel in a graphite element were first demonstrated in the DRAGON reactor, a European research facility that began operation in 1964.

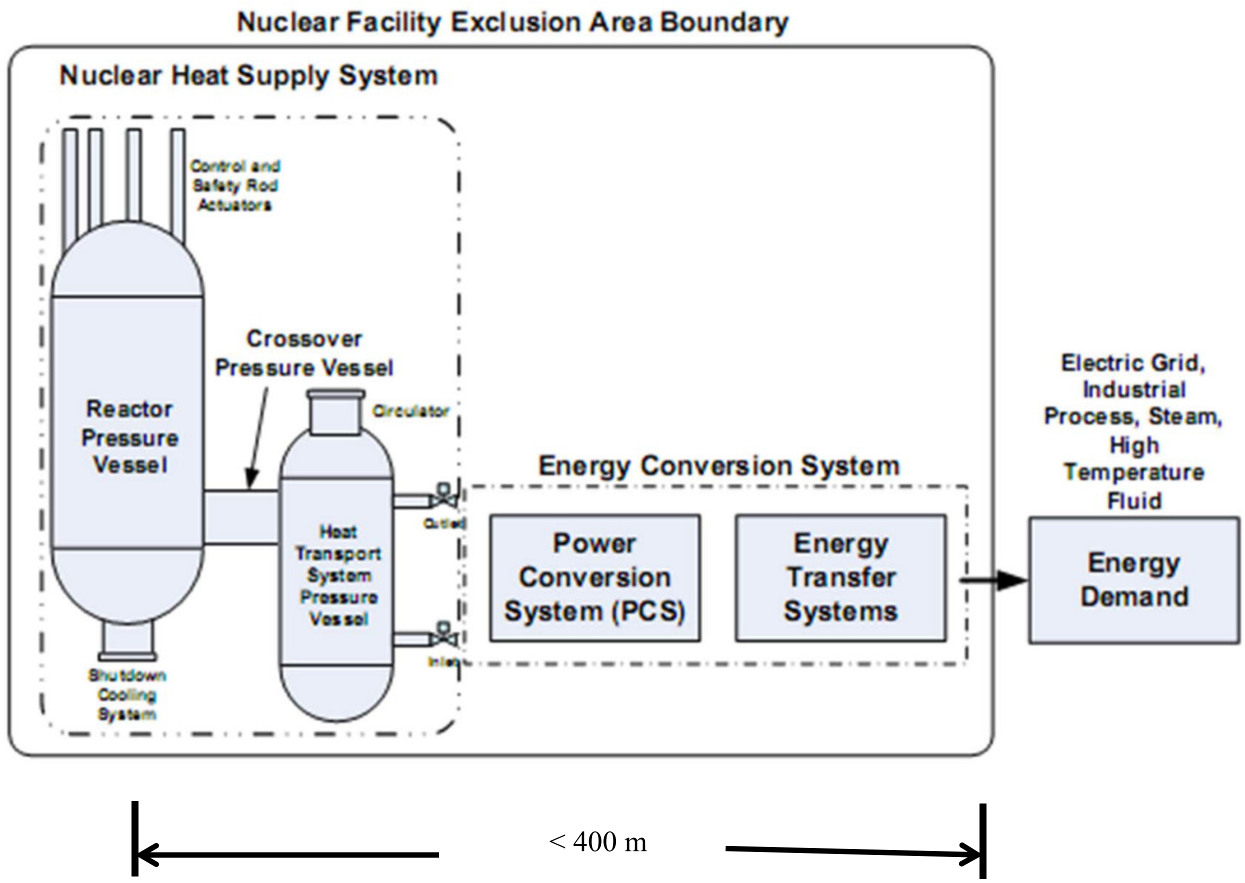

Figure E-1. Collocated smHTR and industrial plant.

Although the conceptual designs originally submitted for consideration under NGNP were originally intended to drive gas turbine (Brayton cycle) power conversion systems, feedback from industrial partners and potential NGNP customers indicated a preference for process heat in the form of steam in the 750-850°C temperature range. This reduced the demand to qualify the components and high temperature alloys required of high pressure, high temperature gas turbine cycles. For higher temperature applications, alloys such as 800H and Inc-617 were identified as suitable for vessels, heat exchangers and other metallic components with a few (<10) years of additional testing and qualification. This testing started under the NGNP program and continues under ARC. The use of an indirect steam (Rankine) power conversion system does increase the risk of water ingress in the graphite core with the possibility of limited graphite interaction and elevated fission product releases. Further studies must be conducted to confirm that this would have a manageable effect on source terms. It should be noted that first generation HTGRs—notably the Arbeitsgemeinschaft Versuchs Reaktor (AVR) in Germany and Fort St. Vrain in Colorado—recovered fully from massive water ingress events and with minimal radionuclide releases. The adequacy of the steam cycle notwithstanding, the very high temperatures and inert primary coolant would support a highly efficient Brayton cycle power conversion system using either helium or supercritical carbon dioxide as a working fluid, assuming the development of a robust heat exchanger with the primary and secondary loops.

The performance of TRISO fuel underpins the passive safety case for the modular HTGR concept. The fuel developed in Germany for the first generation HTGRs showed excellent integrity and fission product retention at burnups as high as ~10% FIMA and temperatures as high as 1600°C. All HTGRs developed since then, starting with the first generation modular HTGRs in the mid-1980s, were designed such that the fuel temperature would never exceed this value even in the event of a complete loss of forced cooling and pressure. It should be noted that, unlike in cladding failure in light water reactors (LWRs) and other reactor concepts, exceeding the 1600°C 'threshold' in HTGR fuel does not result in catastrophic fuel damage. Rather, the rate of release of fission products through degraded or failed particle coatings increases and leads to increased contamination of the primary circuit and, if the primary pressure boundary is breached, actionable environmental releases. In typical modular HTGR core geometries (annular, tall and thin) with low power density and large graphite thermal inertia, the rise in fuel temperatures occurs over several hours, and only a small fraction of all fuel kernels are exposed to peak fuel temperatures in excess of 1600°C.

Recent heating tests performed on highly irradiated fuel under the Advanced Gas Reactor (AGR) Fuel Program indicate the progress made in the design and fabrication of this fuel form. No non-negligible fission product releases were observed in statistically significant quantities of fuel particles at temperatures as high as 1800°C, indicating a safety margin even higher than the German standard. These results will be confirmed through the additional testing planned under ARC.

Most current modular HTGR concepts are designed to burn low-enriched uranium (8-16%) in a 'once-through' cycle to fractional burnups as high as 16%. In addition to being a highly robust ('accident-tolerant') fuel form, the TRISO particle is expected to perform very well in encapsulating spent fuel in a repository. If stored in the surrounding graphite matrix, the spent fuel is high in volume but also low in heat loading. The graphite matrix, however, can be removed to leave a lower volume waste form with a correspondingly higher thermal loading. Research is underway in a number of countries into the decontamination and recycling of used graphite.

HTGRs have been fueled with thorium and other fuels. Both Fort St. Vrain (USA) and the Thorium HochTemperatur Reaktor (Germany) operated on a combination of highly enriched uranium and fertile thorium. Under the early German HTR program, a number of different closed fuel cycles were considered. A plutonium-burning HTGR has been the subject of R&D collaboration between the U.S. and Russia as well as in Europe. TRISO fuel can be (and has been) reprocessed. Chemical removal of the graphite matrix and outer pyrolytic coating reveals the more durable silicon carbide layer which can be breached through electrical or mechanical means. Once breached, the ceramic fuel kernel can be leached out for processing via standard treatments. Proliferation concerns and the relatively low price of enriched uranium have removed any real motivation for closed cycle HTGRs. Also, as a thermal spectrum reactor, the HTGR can effectively reduce plutonium stockpiles but minor actinides will accumulate even in recycled fuel. Most new modular HTGRs and the VHTR are designed to burn only fresh low-enriched uranium (LEU) although thorium cycles are still being explored.

The graphite used in the 1st generation HTGRs no longer exists. New grades produced by a few vendors are the subject of extensive characterization and irradiation testing under the NGNP (now ARC) program. Preliminary results of these tests indicate that the quality of the newer graphite grades is superior to, and the material properties are more uniform than those of, their predecessors.

The methods used to simulate plant behavior for design and licensing have matured under the DOEs NGNP program but still lag behind the industry's ability to simulate LWRs in terms of fidelity and accuracy. Codes are being developed or updated to capture HTGR physics but there remain gaps in the neutronic and thermal fluid database needed to validate these codes and models to modern regulatory standards. A few integral and separate effects experiments are underway to fill the gaps but any design submitted under a near-term license application will need to reflect considerable safety margins to be accepted (achieved at the expense of performance).

A complete report of the status of HTGR technology (including references) is provided in the accompanying status report [1].

The technical barriers to small modular HTGR deployment will be largely overcome with the completion of the fuel and material qualification programs and, to a lesser extent, the experimental confirmation of thermal fluid behavior under accident conditions. The major non-technical challenges are licensing and economics. Licensing is inhibited by the LWR-centric requirements in the federal regulations, the lack of extensive operating plant and experimental data, the approximate models used in simulation, and the lack of familiarity with the HTGR among regulators. The economic challenge arises from: (1) the lack of an industrial infrastructure supporting HTGR fuel and plant manufacturing, (2) the uncertainties in modeling that must be compensated by more conservative designs, and (3) the availability of inexpensive fossil fuels for electricity generation, transportation and heating, and industrial process heat.

CONTENTS

FIGURES

TABLES

ACRONYMS

AGR	Advanced Gas Reactor
ARC	Advanced Reactor Concepts
AVR	Arbeitsgemeinschaft Versuchs Reaktor
BOP	Balance of Plant
CANDU	Canadian Deuterium/Uranium
CFD	Computational Fluid Dynamics
CR	Control rods
DOE	Department of Energy
FBP	Fixed Burnable Poison
FP	Fission Products
GA	General Atomics
HLW	High level waste
HTR	High Temperature Reactor
HTTR	High Temperature engineering Test Reactor
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
IPyC	Inner pyrolitic carbon
KMP	Key measurement points
LBP	Lumped Burnable Poison
LEU	low enriched uranium
LWR	Light water reactors
MHTGR	Modular High Temperature Gas Reactor
NOAK	N_{th}-of-a-kind
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
OPyC	Outer pyrolitic carbon
PBMR	Pebble Bed Modular Reactor
PBR	Pebble bed reactor
PCS	Power Conversion System
R&D	Research and development
RCCS	Reactor Cavity Cooling System
RSC	Reserve shutdown control
RSS	Reserve shutdown system

SMR	Small Modular Reactor
SSC	Systems, structures, and component
TDRM	technology development roadmap
TRISO	tristructural isotropic
UCO	Uranium Carbide/Oxide
VHTR	Very High Temperature Reactor
VSOP	Very Superior Old Programs

Baseline Concept Description of a Small Modular High Temperature Reactor

1. Technical Description of Reference Concepts

Although they share the attributes described above, two design variants of the modular HTGR have been developed and are sufficiently different in other aspects to be described in parallel in this report as separate reference concepts (Figure 1). General Atomics (GA) began development of the *prismatic* (block) modular HTGR in the 1960s and, with support from the U.S. government, pursued variations of the concept in support of different missions. Two HTGRs designed by GA—the 115 MWt Peach Bottom-1 and the 750 MWt Fort St.Vrain—delivered power to the U.S. grid. One prismatic reactor is in operation today, the 30 MWt High Temperature engineering Test Reactor (HTTR), built by the Japan Atomic Energy Agency and located at the Oarai Research Laboratory. In the 1980s, GA developed the Modular High Temperature Gas Reactor (MHTGR) and commenced preliminary licensing activities with the Nuclear Regulatory Commission (NRC). The MHTGR is the basis of the prismatic reference concept described in this paper.

The other class of HTGR is the pebble bed reactor (PBR) pioneered in Germany, following roughly the same development trajectory as the prismatic core. The 46 MWt Arbeitsgemeinschaft VersuchsReaktor operated for 20 years, primarily as a demonstration system and a testbed for pebble fuel but it also delivered power to the grid. It was succeeded by the Thorium HochTemperatur Reaktor (THTR), which operated only for a few years but also generated electricity. One PBR is in operation today, the 10 MWt HTR-10, built by China's Institute for Nuclear Energy Technology. In the 1980s, Interatom, a German industrial consortium, commenced licensing activities (in Germany) on the HTR Modul, a 200 MWt modular PBR design that featured online fueling. China adopted this design and is in the process of building a two-unit plant that is scheduled to commence operation in 2016. The HTR Modul is the basis of the pebble bed reference concept described in this paper.

Neither the MHTGR nor the HTR Modul would require active decay heat removal to ensure fuel integrity. The TRISO-coated particle fuel, the large graphite mass, low power density, and natural heat removal mechanisms are sufficient to maintain fuel integrity under all postulated scenarios.

As of 2014, no TRISO fuel form is qualified for commercial use in the U.S. Under the Next Generation Nuclear Plant (NGNP)/Advanced Gas Reactor (AGR) (now Advanced Reactor Concepts [ARC]) program and in cooperation with a fuel vendor, TRISO fuel is being qualified through a series of irradiations and heating tests under an ASME NQA-1-2008; 1a -2009 program. Any near-term smHTR project outside of China will therefore be constrained to use AGR fuel. For this reason, it is assumed that AGR fuel will be used for both of the reference concepts.

Tristructural-Isotropic (TRISO) fuel particles are bonded in a graphite matrix to form either a cylindrical 'compact' or a spherical pebble (Figure 1). TRISO particles consist of various layers acting in concert to provide a containment structure that limits radioactive product release. They include a fuel kernel, porous carbon layer, inner pyrolitic carbon (IPyC), SiC, and outer pyrolitic carbon (OPyC). The buffer layer allows for limited kernel migration and provides some retention of gas compounds. The silicon carbide layer ensures the structural integrity of the particle under constant pressure and also helps retain metallic fission products. Details of the TRISO particle and compact designs are given in table 1 in part 2. Compacts are inserted into hexagonal graphite blocks to assemble a prismatic fuel element. For pebbles, a 5-mm layer of graphitic matrix material is forms a protective shell around the inner fueled zone (see Figure 1).

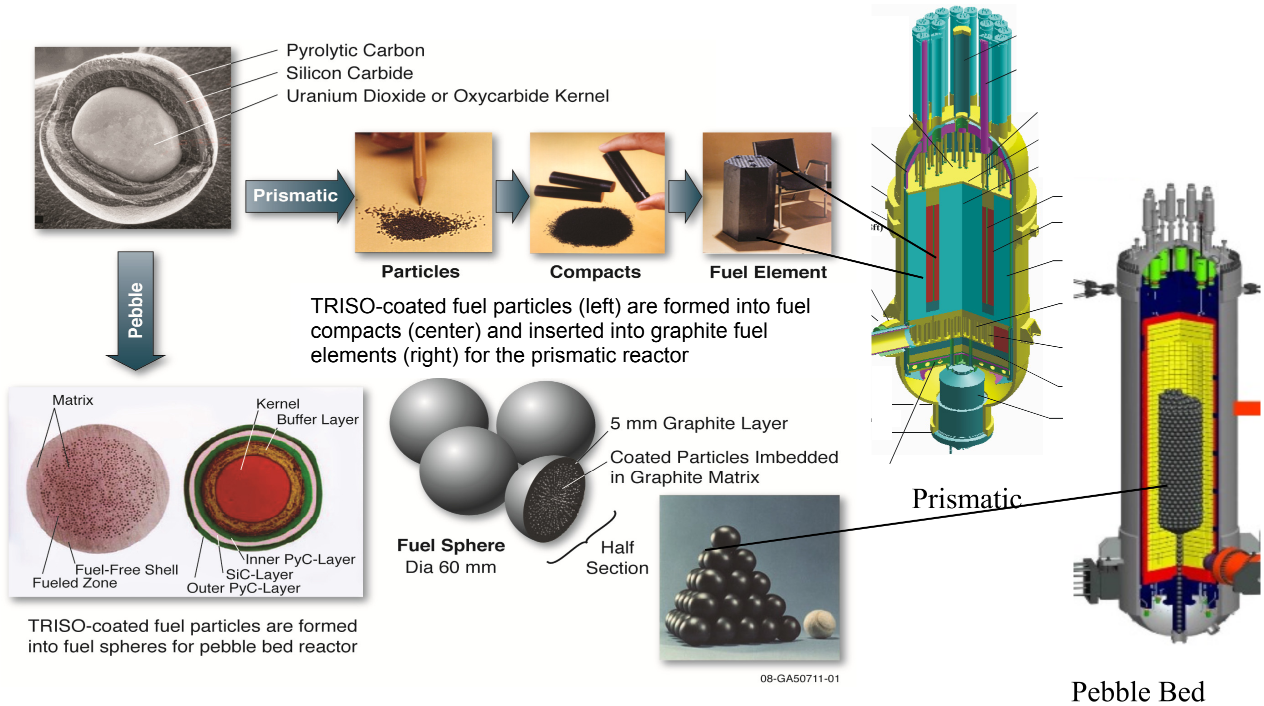

Figure 1. TRISO fuel as loaded into a prismatic or PBR.

1.1 MHTGR (Prismatic Reference Concept)

The MHTGR 350 is a GA design that was developed in the 1980s. The major features of the power plant are shown in Figure 2 and the main characteristics of the design are summarized in table 5 in Section 3. The reactor vessel contains the reactor core, reflectors and associated neutron control systems, core support structures, and shutdown cooling heat exchanger and motor-driven circulator. The steam generator vessel houses a helically-coiled steam generator bundle as well as the motor-driven main circulator. The pressure-retaining components are constructed of steel and designed using existing technology.

The reactor vessel is uninsulated to provide for decay heat removal under loss-of-forced-circulation conditions. In such events, heat is transported to the passive Reactor Cavity Cooling System (RCCS), which circulates outside air by natural circulation within enclosed panels surrounding the reactor vessel. No valves, fans, or other active components or operator actions are needed to remove heat using the Reactor Cavity Cooling System (RCCS).

The reactor core and the surrounding graphite neutron reflectors are supported within a steel reactor vessel. The restraining structures within the reactor vessel are a steel and graphite core support structure at the bottom and a metallic core barrel around the periphery of the side reflectors.

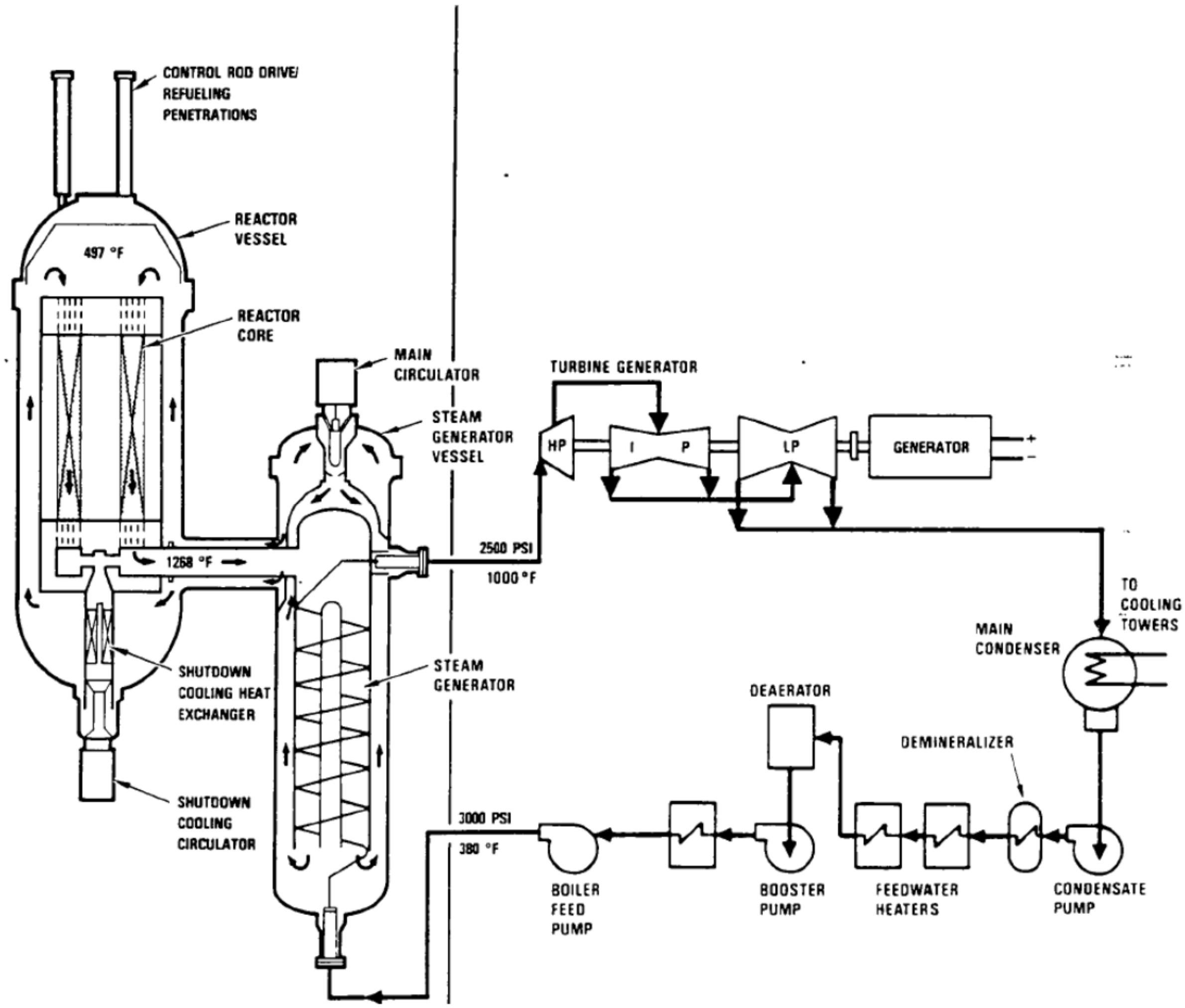

Figure 2. Schematic of the MHTGR power plant (best available drawing).

The core is designed to provide 350 MWt at an average power density of 5.9 MW/m^3. A core elevation view is shown in Figure 3. The design of the core consists of an array of hexagonal fuel elements in a cylindrical arrangement surrounded by a single ring of identically sized solid graphite replaceable reflector elements, followed by a region of permanent graphite reflector elements all located within a reactor pressure vessel. The permanent reflector elements contain a 10-cm thick borated region at the outer boundary, adjacent to the core barrel. The borated region contains B_4C particles of the same design as in the Fixed Burnable Poison (FBP), but dispersed throughout the entire borated region with a volume fraction of 61%.

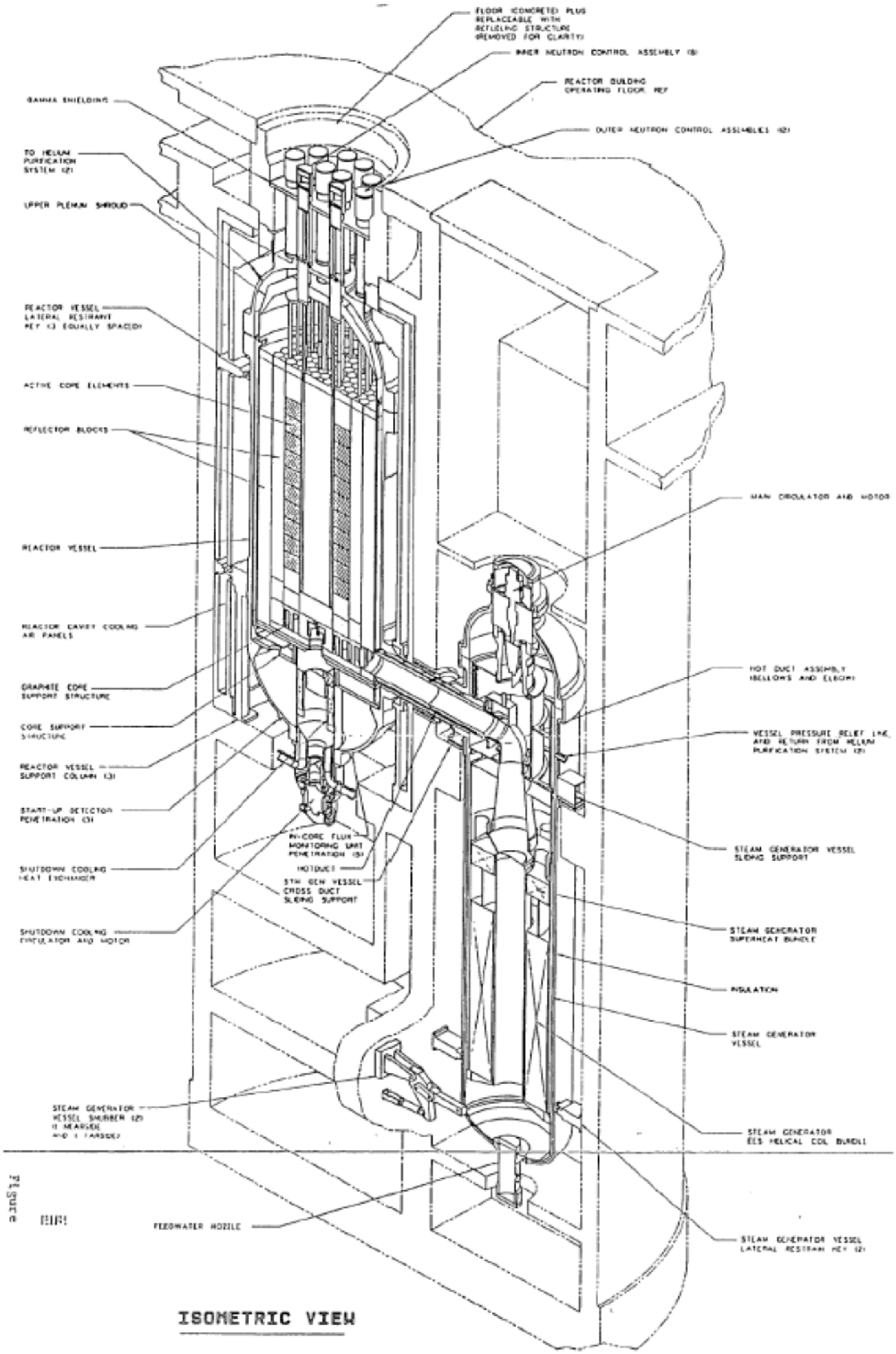

Figure 3. MHTGR reactor vessel and internals.

The active core consists of hexagonal graphite fuel elements containing blind holes for fuel compacts and full-length channels for helium coolant flow. The fuel elements are stacked to form columns (10 fuel elements per column) that rest on support structures. The active core columns form a three-row annulus with columns of hexagonal graphite reflector elements in the inner and outer regions. Thirty reflector columns contain channels for control rods, and 12 columns in the core also contain channels for the reserve shutdown material.

The annular core configuration (Figure 4) was selected, along with the average power density of 5.9 MW/ m^3, to achieve maximum power rating and still permit passive core heat removal while maintaining the SiC temperature below ~1600°C during a conduction cooldown (also known as depressurized loss-of-forced cooling) event. The active core effective outer diameter of 3.5 m is sized to maintain a minimum reflector thickness of 1 m within the 6.55-m inner diameter reactor vessel. The radial thickness of the active core annulus was specified on the basis of ensuring that the control rod worths of the reflector located rods would meet all shutdown and operating control worth requirements. The choice of reflector control rods was made to ensure that the control rod integrity is maintained during passive decay heat removal events. These radial dimensions also allow for a lateral restraint structure between the reflector and vessel. The height of the core with ten elements in each column is 7.9 m, which allows maximum power rating and axial power stability over the cycle.

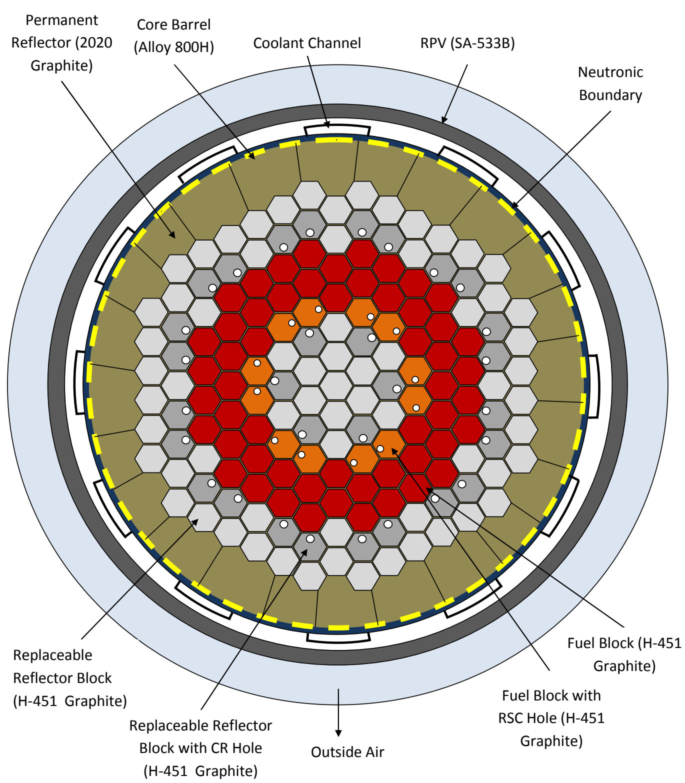

Figure 4. Planar view of the MHTGR core.

There are two types of fuel elements, a standard element, and a reserve shutdown (Figure 5) element that contains a channel for reserve shutdown control (RSC). The fuel elements are right hexagonal prisms of the same size and shape as the Fort St. Vrain HTGR elements.

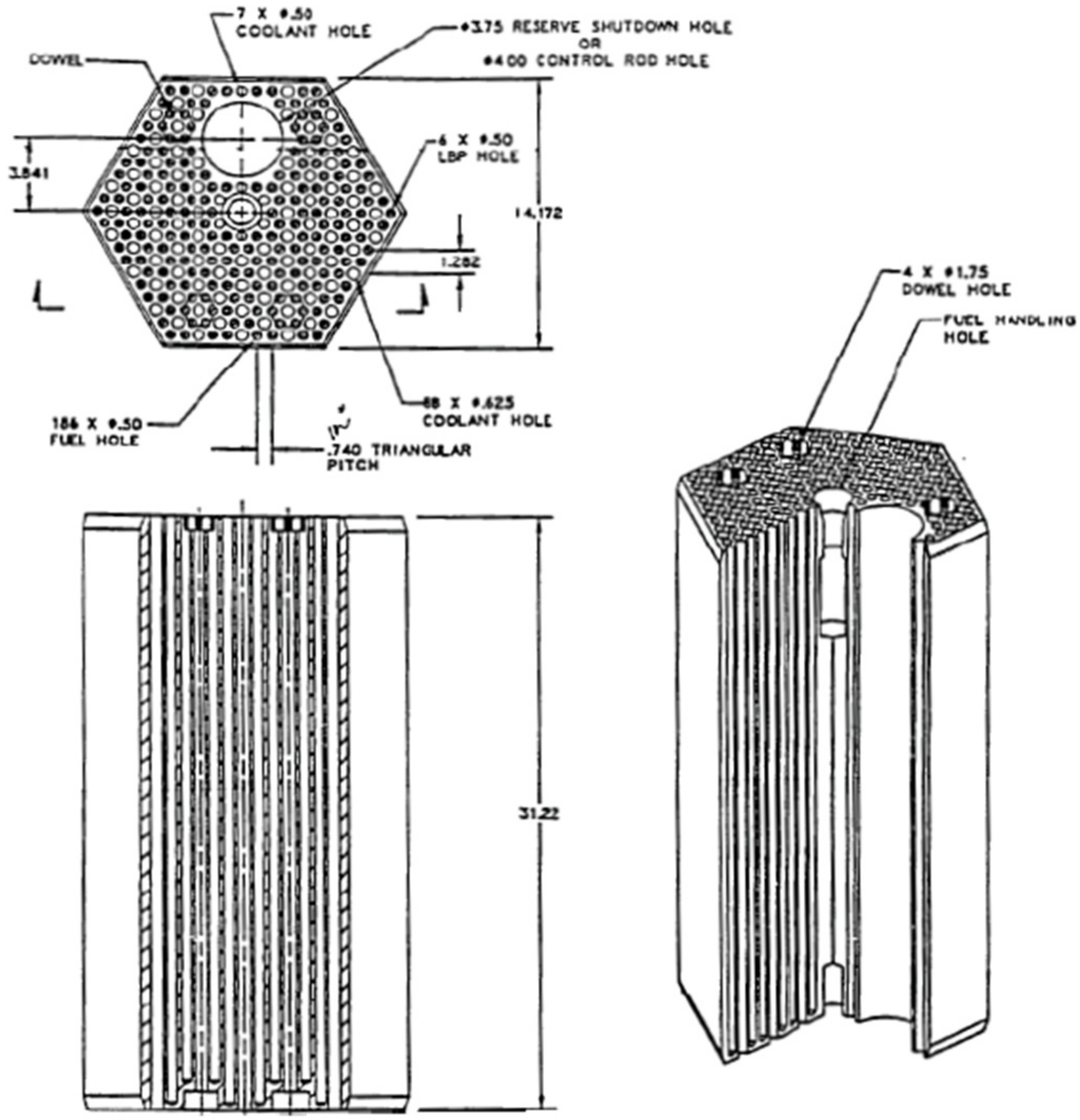

Figure 5. Reserve shutdown control element.

The fuel and coolant holes are located in parallel through the length of the element. The standard fuel element contains a continuous array of fuel and coolant holes in a regular triangular array of two fuel holes per one coolant hole. The six corner holes contain lumped burnable poison compacts.

At each element-to-element interface in a column, there are four dowel/socket connections, which provide alignment of coolant channels. A 3.5-cm diameter fuel-handling hole, located at the center of the element, extends down about one-third of the height, with a ledge where the grapple of a fuel-handling machine engages.

The core reactivity is controlled by a combination of lumped burnable poison (LBP), movable poisons, and a negative temperature coefficient. The fixed poison is in the form of LBP compacts inserted into fuel blocks near the vertices of the hexagon. The movable poison is in the form of metal-clad control rods inserted into the inner and outer outer radial reflectors. Should the control rods become inoperable, a backup RSC is provided in the form of borated pellets that may be released into channels in the active core. Details of the control element construction are available in references [9].

1.2 HTR Modul (Pebble Bed Reference Concept)

The HTR Modul originally designed by Interatom in the 1980s is now being built, with some modifications, as a 2x250 MWt plant in China. The major features of the power plant are shown in Figure 6 and Figure 7. The reactor vessel contains the reactor core, reflectors and associated neutron control systems, core support structures, and motor-driven circulator but no shutdown cooling system. The steam generator vessel houses a helically-coiled steam generator bundle as well as the motor-driven main circulator. The pressure-retaining components are constructed of steel and designed using existing technology.

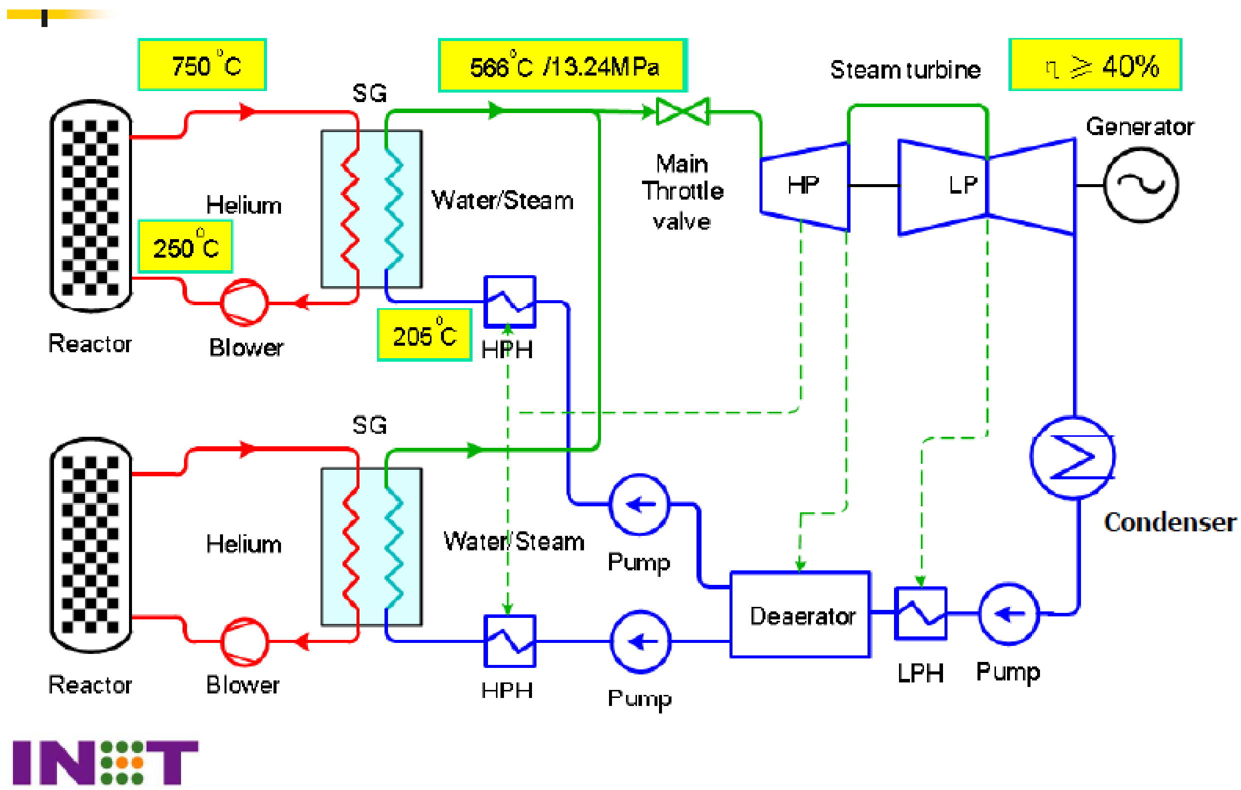

Figure 6. Major features of the HTR-PM (China) variant of the HTR Modul.

Each core is designed to provide 200 MWt at an average power density of 3.0 MW/m^3 [2]. The design of the core consists of cylindrical vessel lined with solid graphite replaceable reflector elements, followed by a region of permanent reflector elements all located within a reactor pressure vessel. The active core contains roughly 350,000 pebbles stochastically loaded and recirculated during operation. When a pebble drops from the bottom discharge chute, its burnup is measured. If it has not exceeded the burnup limit, it will be transferred pneumatically to the top of the core for another pass. Each pebble passes through the core about 15 times before final discharge to the spent fuel storage containers.

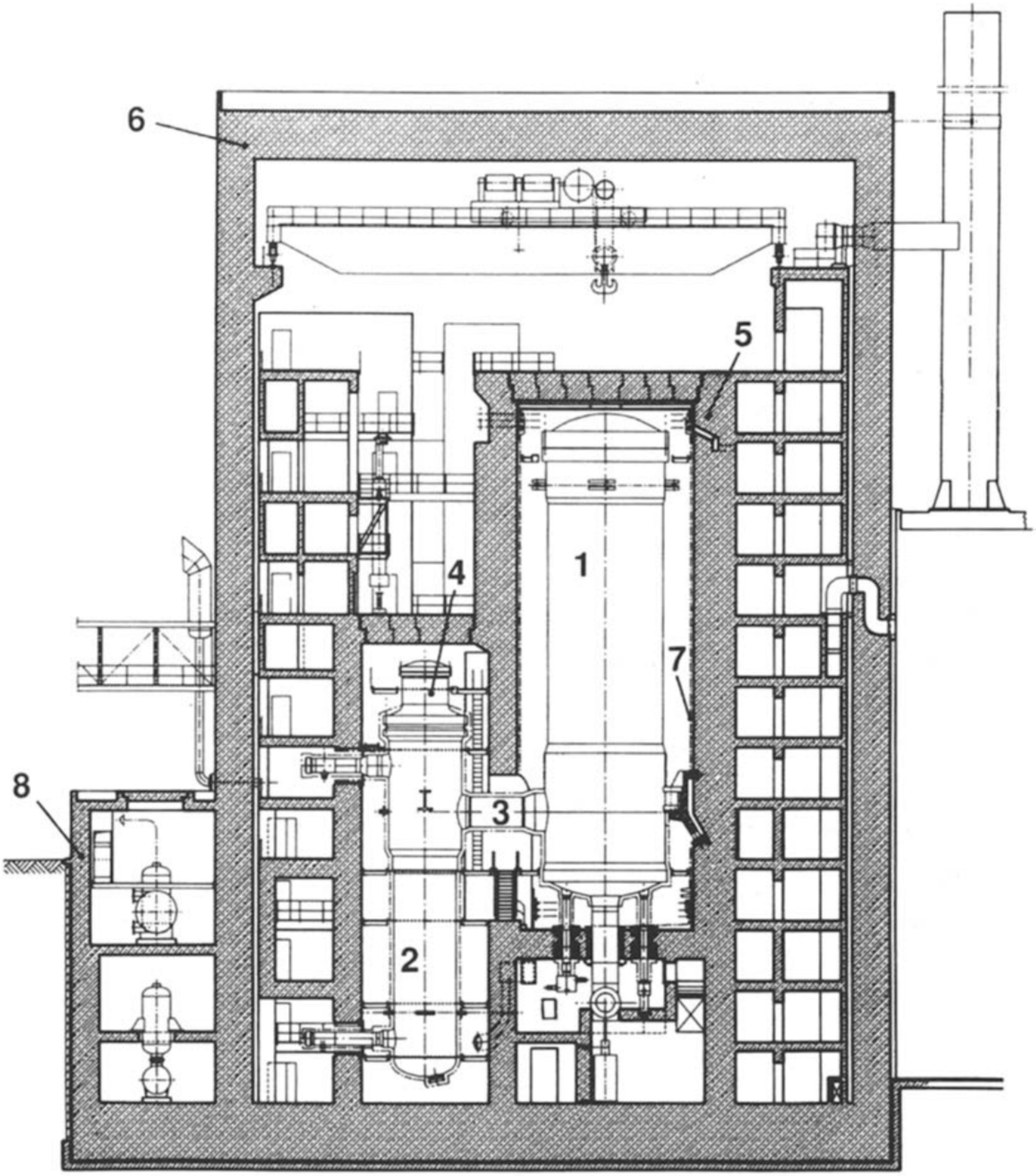

Figure 7. Layout of the HTR-PM (Modul) vessel and steam generator[3]

Figure Legend.

1	Reactor pressure vessel	5	Primary cell
2	Steam generator	6	Concrete shield
3	Cross vessel	7	Surface cooler (RCCS)
4	Primary circuit blower	8	Reactor building annex

The 3-meter diameter of the pebble bed permits passive core heat removal (via conduction and radiation) while maintaining the SiC temperature below ~1600°C during a loss of forced cooling event. The 1-m graphite side reflector also serves to absorb the heat of a thermal transient, pass it to through the core barrel to the pressure vessel where is radiates to the RCCS. The diameter of the active core is limited to ensure that the combined worth of the reflector-located rods would meet all shutdown and operating reactivity requirements.

The core operates with very little excess reactivity as the online fuel system adds fresh pebbles to the core at the minimum frequency to maintain overall criticality, possibly with rods partially inserted to provide xenon-override capability if rapid restart is desired after an unplanned shutdown. Thus no burnable poisons are required to hold down reactivity. Along with the complete burnup of each pebble, this enhances fuel economy over batch-loaded cores. The low excess reactivity also inhibits the use of the reactor for weapons material production as any drop in reactivity would have immediate, negative, and observable impacts on power production and fuel consumption.

The six control rods are fabricated from boron carbide (B_4C) containing 10w/o natural boron pressed between metal tubes that form a flexible train that is lowered into the core. These operating rods are used to control the power level and can maintain the required 1% $\Delta\rho$ shutdown margin indefinitely under hot conditions. B_4C spheres are injected into 18 reflector channels to provided sufficient excess reactivity to keep the core subcritial under ambient (cold) conditions.

An inadvertent withdrawal of all control rods is a limiting reactivity event which can be controlled simply by tripping the primary coolant circulators. The subsequent temperature rise shuts down the fission reaction while the excess heat is transported to the cavity cooling system (Figure 8).

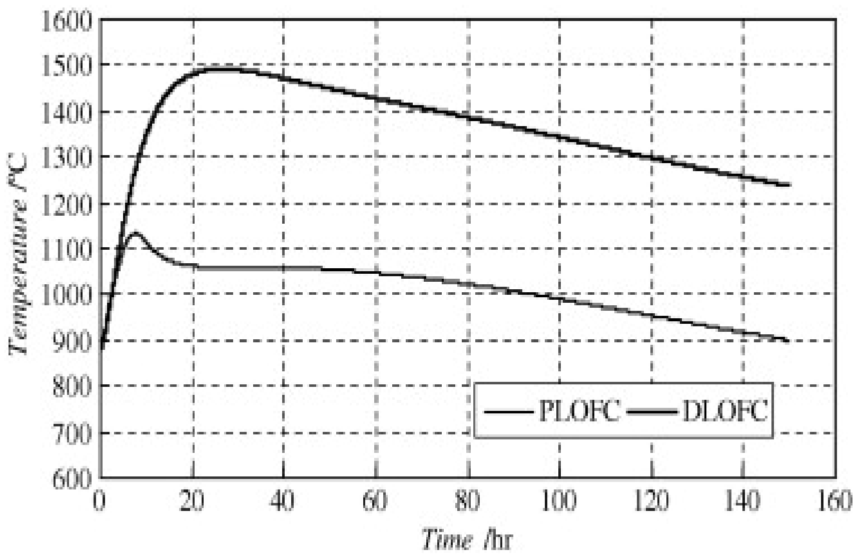

Figure 8. Peak fuel temperature during pressurized and depressurized loss of forced cooling transients [4].

2. Key Technical Parameters of the smHTR

The following table lists the major design parameters and attributes of the two concepts.

Table 1: Summary of features for the two smHTR concepts.

	Prismatic Core	**Pebble Bed Core**
1. Plant Configuration		
Primary System	Reactor Pressure Vessel connected to a Steam General by a Coaxial Cross Vessel (Duct)	Reactor Pressure Vessel connected to a Steam General by a Coaxial Cross Vessel (Duct)
Intermediate Loop(s)	1, steam	1, steam
Number of Turbines	3 (HP, IP,LP)	2 (HP, LP)
Reactor Building Characteristics	Below-grade primary/intermediate Seismic isolation for reactor	Below-grade primary/intermediate Seismic isolation for reactor
2. Energy Conversion and Balance of Plant (BOP)		
Power Conversion Cycle	Rankine - Superheated	Rankine
Thermal Efficiency	38%	40%
Turbine Building Characteristics	Above-grade adjacent to reactor No seismic isolation	
3. Construction Techniques		
General Approach	Traditional power plant construction enhanced with modular component fabrication	
Transportability	Not significantly different than small modular LWRs	
4. Key Plant Parameters		
Power (MWe/MWt)	134/350	80/200
Primary Coolant	Helium	Helium
Plant lifetime	60 years	60 years
Reactor Vessel, outer dia. (m) height (m)	6.9 22.5	
5. Core Performance and Safety		
Average Power Density (W/cc)	5.9	3.0
Cycle Length (years)	1.5	Online refueling
Capacity Factor (%)	>80*	>90
Cycle Reactivity Loss ($)	9[5]	NA
Fuel Enrichment (fissile/heavy metal)	15.5% LEU	8.0
Initial Loading (MT of Heavy Metal)	4.5	2.553

* Quoted from reference 2 (1987 report)

	Prismatic Core	Pebble Bed Core
Average/Peak Burnup (MWd/kg)	55/117	166[†]
Peak Fast Fluence (>0.1 MeV)	4.4* 10^{21} n/cm^2	3.5* 10^{21} n/cm^2
Delayed Neutron Fraction	Unavailable from references	
Power Coefficient (cents/°C)	-0.5	Unavailable from references
Doppler Coefficient (cents/°C)	Unavailable from references	
Fuel Temperature (°C)		
Steady-State (Mean/Peak)	680/1060	540/830
DLOFC (Peak)	1600	1540
6. Coolant and Thermal Performance		
Primary Coolant, T-in (°C) T-out (°C)	259 687	250 750
Primary Coolant, Pressure (MPa)	6.39	6.0
Primary Coolant Purification	Helium Purification System (charcoal beds, molecular sieve, oxidation bed, compressors, filters)	
Primary Coolant Corrosion Control	Unavailable from references	Unavailable from references
Primary Pumping Power Coolant Flow Rate	2x360kW 157.1 kg/s	 85 kg/s
Secondary Coolant, T-in (°C) T-out (°C)	293 541	170 530
Secondary Coolant, Pressure (MPa) Feedwater/Steam outlet	21.0/17.3	19
7. Fuel Properties		
Fuel Type	TRISO-coated particles in a carbon matrix	
Fuel Kernel	$UC_{0.4}O_{1.5}$ (UO_2/UC/UC_2 mixture)	
outer dia. (μm)	425	
Density (g/cm^3)	10.8	
Coatings	TRISO Thickness (mm)/Density (g/cm^3)	
Buffer (carbon)	100/1.05	
Inner Pyrolytic Carbon	40/1.90	
Silicon Carbide	35/3.19	
Outer Pyrolytic Carbon	40/190	
Fuel	Cylindrical Compact in Block	Spherical Pebble
Dimensions (cm)	4.928cm (height)	3.0 cm

[†] With online fueling, each pebble attains the target burnup (within 7%)

	Prismatic Core	Pebble Bed Core
	0.6225 cm (radius)	(radius)
Number of particles per element packing fraction	5986 35%	19000 9.5%
Matrix density (g/cm^3)	1.75	1.75
Number elements in the core	~2035800	360000
Fuel Block		NA
Height/Distance across flat (cm)	79.3/36	
Number Fuel holes, Standard/RSC	210,186	
Fuel hole radius (cm)	0.635 cm	
Number Large Coolant holes, Standard/RSC	102/88	
Number Large Coolant hole radius (cm)	0.794 cm	
Number Small Coolant holes, Standard/RSC	6/7	
Number Small Coolant hole radius (cm)	0.635 cm	
Fuel/Coolant Pitch	1.8796 cm	
LBP holes, diameter	6,1.143 cm	
Number per core, Standard/RSC	540/120	
Reflector Control Rods (Inner/Outer)	6/24	NA/6
Reserve Shutdown Channels	12	18
Sphere diameters (mm)/ w/o B_4C	Not available/40%	10/10%
Fuel Handling Machine	Transportable (not reside at site) In-vessel pantograph design located in single rotating plug	Onsite Fuel Handling Machine
9. Safety Systems		
Emergency Heat Removal	Shutdown Heat Removal system (if available) and conduction through core structures and thermal radiative heat transfer across gaps to the RCCS	
Reactor Shutdown Systems	Two independent control rod groups	
Inherent Safety Potential	Fission products are adequately retained in the fuel element under any conceivable loss of forced cooling event	
10. Containment System		
Primary Containment Boundary	TRISO (SiC) coating	
Secondary Confinement	Graphite element/Primary Coolant pressure boundary	
Functional Containment Design Basis	See the note following this table.	

	Prismatic Core	Pebble Bed Core
11. Decay Heat Management		
Normal Decay Heat Removal Path	Power operation: Intermediate loop, through the BOP A shutdown cooling system connected to a separate heat exchanger is activated at low power	
Backup Heat Removal System Design (RCCS)	Natural circulation-driven air-cooled cavity cooling panels (RCCS) – not necessary to maintain fuel integrity, only vessel	
Backup Decay Heat Removal Capacity	1.75 MWt (0.5% rated power)	0.85 MWt (0.425% rated power)
Ultimate Heat Sink	Outside air through RCCS/radiative heat transfer to earth in case RCCS fails	

Functional Containment Design Basis Under the NGNP Program, a 'functional containment' approach was defined comprising several barriers that limit the release of radionuclides to the environment (source term) for each postulated event, including normal operating conditions, abnormal operating conditions, and accident conditions. Such an approach is likely to be pursued for a smHTR. The multiple barriers include:

- individual fuel particle kernels,
- fuel particle coatings,
- the fuel matrix and fuel element graphite,
- the helium pressure boundary (primary circuit), and
- a vented low-pressure reactor building.

Each of these barriers contributes to limiting the release of radionuclides to the environment to meet the NGNP Project top-level radiological criteria. The contribution of each of the barriers in limiting the transport and release of radionuclides to the environment is calculated for each postulated event, depending on the response of the reactor to the event. The top-level radiological (or "design") criteria are derived from externally imposed requirements or guidelines, such as site boundary dose limits, occupational exposure limits, Environmental Protection Agency (EPA) Protective Action Guides (PAGs), etc.

The top-level radionuclide control requirements limit calculated dose under all licensing basis events so that regulatory requirements for protection of the health and safety of the public and protection of the environment are met at an exclusion area boundary (EAB) that is no more than a few hundred meters from the reactor (e.g., 400 to 425 meters). Limits on radionuclide release from the reactor building that are consistent with these top-level radionuclide control requirements are needed to establish the target values for all of the barriers to radionuclide release and ultimately to establish allowable in-service fuel failure and as-manufactured fuel quality requirements (e.g., allowable heavy metal contamination, SiC coating defects, etc.).

For frequent events expected to occur within a plant's lifetime, 10 CFR §20.1301 requires that the total effective does equivalent (TEDE) for a member of the public be limited to 100 mrem per year. For design basis events, 10 CFR §50.34(a)(1) requires that any reactor be designed such that: An individual located at any point on the EAB would not receive a radiation dose in excess of 25 rem TEDE for any 2-hour period following the onset of a postulated fission product release. For beyond design basis accidents, the quantitative health objectives (QHOs) are applied as the basis for determining achievement of the NRC's "Safety Goals for the Operations of Nuclear Power Plant Operation."

The above top-level regulatory criteria are derived from existing regulatory requirements and form the basis for NGNP operation under normal, accident, and severe accident conditions. However, the NGNP has also applied a design goal of meeting the EPA PAGs (≤ 1 rem TEDE or ≤ 5 rem Thyroid dose) at the EAB as means of demonstrating the safety margins provided by the functional containment.

3. Technical Readiness and Estimated Costs

The NGNP project issued a report that documents the Technology Readiness Assessment of critical systems, structures, and components along with technology development roadmaps (TDRM) to mature the technologies needed for a high-temperature gas reactor with an outlet temperature of 950°C, as well as other requirements consistent with those found in the NGNP requirement documents [6]. This report reconciled the assessment of Technical Readiness Levels (TRLs) and the technology development roadmaps developed by the gas reactor vendors participating in the project (AREVA, GA, Westinghouse). The results are summarized in Table 2. An update to the TDRM report was issued later in August 2009 to chart a path forward for a 750°C reactor outlet temperature (ROT) [7]. An assessment of reactor user interface TDRMs was also performed early in FY11 to evaluate the technology readiness of the interface components that are required to transfer high temperature heat from an HTGR to selected industrial applications [8].

Table 2: Technology Readiness Levels for Major SSCs for a VHTR with a 950°C Outlet Temperature.

NGNP / Area / System	Min TRL	Avg TRL
NGNP	3	3.8
Nuclear Heat Supply System (NHSS)	4	4.0
Reactor Pressure Vessel System	4	
Reactor Vessel Internals	4	
Reactor Core and Core Structure	4	
Fuel Elements	4	
Reserve Shutdown System	4	
Reactivity Control System	4	
Core Conditioning System	4	
Reactor Cavity Cooling System	4	
Heat Transfer System (HTS)	3	3.8
Circulators	4	
Intermediate Heat Exchangers	3	
Hot Duct - Cross Vessel	4	
High Temperature Valves	3	
Mixing Chamber	5	
Hydrogen Production System (HPS)	3	3.3
Power Conversion System (PCS)	4	4.0
Steam Generator	4	
Power Conversion Turbomachinery	4	
Balance of Plant (BOP)	3	3.5
Fuel Handling System	4	
Instrumentation & Control	3	

One of the vendors (AREVA) summarized the TRLs for their particular 750°C prismatic design that is similar to the MHTGR (Table 3). The differences between these tables clearly show the challenge of qualifying materials at the higher temperatures. The TRL for the entire plant, as given by the minimum SSC TRL is 3 for the 950°C VHTR and four for the 750°C HTGR. As the major SSCs are the same for the prismatic and pebble bed concepts, similar rankings would be expected for the HTR Modul. The values in Table 3 may be conservative as both prismatic and pebble bed commercial power plants have operated and that China is building a two-unit pebble bed HTGR, based upon the HTR Modul design that is scheduled to commence operation in 2016.

Table 3. TRLs for major components in the AREVA reference design with a 750°C outlet temperature.

System, Structure, or Component	TRL
Nuclear Heat Source	
• Vessel System	7
• Reactor Internals	4
• Reactor Core	4
• Control Rod Drives	4
• Nuclear Instrumentation	7
Main Heat Transport System	
• Main Helium Circulator	6
• Circulator Shutoff Valve	6
• Hot Duct	5
Power Conversion System	
• Steam Generator	6
Other Reactor Support Systems	
• Primary Loop Instrumentation	6
• Fuel handling System	6
• Reactivity Control System	5
Process Heat Transport System	
• Steam Reboiler System	8

A technical evaluation [14] was prepared as part of a study for the Next Generation Nuclear Plant (NGNP) Project to address estimating the capital, operating, and decommissioning costs of a high-temperature gas-cooled reactor (HTGR). The results should be considered preliminary and are not specific to the MHTGR or HTR-Modul, but can provide a rough estimate of the costs of building those reactors today. The level of project definition for this study was determined to be an International Class 4 estimate‡ with probable error range of -30%/+50%.

‡ Association for the Advancement of Cost Engineering

Table 4: Cost Summary for NGNP and NOAK HTGR with a Rankine power cycle, 850°C ROT.

Reactor Phase	NGNP		NOAK			
Reactor Size (MWt)	600	350	600 MWt		350 MWt	
Number of Modules	1	1	1	4	1	4
Capital Costs ($\$10^6$)						
Preconstruction Costs	233.50	233.50	76.50	91.00	76.50	91.00
Direct Costs	1253.70	941.02	764.15	2565.53	543.40	1818.70
Indirect Costs	1734.40	1554.73	459.10	1494.20	332.25	1065.06
Contingency	644.32	545.85	259.95	830.15	190.43	594.95
Overnight Cost ($\$10^6$)	3865.92	3275.11	1559.70	4980.88	1142.57	3569.72
Lower Bound	2706.14	2292.57	1091.79	3486.61	799.80	2498.80
Upper Bound	5798.88	4912.66	2339.55	7471.31	1713.86	5354.57
Overnight Cost ($/kWt)	6443.19	9357.44	2599.50	2075.37	3264.49	2549.80
Lower Bound	4510.24	6550.21	1819.65	1452.76	2285.14	1784.86
Upper Bound	9664.79	14036.17	3899.26	3113.05	4896.73	3824.70
Yearly O&M Costs ($\$10^6$)	37.54	34.39	37.54	99.60	34.39	87.00
Yearly O&M Costs ($/MWt-hr)	7.14	11.22	7.14	4.74	11.22	7.09
Yearly Fuel Costs ($\$10^6$)	57.28	33.41	33.47	133.88	19.52	78.10
Yearly Fuel Costs ($/MWt-hr)	10.90	10.90	6.37	6.37	6.37	6.37
Decommissioning Costs ($\$10^6$)	122.87	71.67	122.87	491.47	71.67	286.69

4. Attributes of the smHTR per the Technical Review Panel Criteria

The following table describes the concept in terms of the criteria specified in the Technology Review Panel report.

Table 5: Summary of Features by TRP Criteria

Criteria	Small Modular High Temperature Reactor (smHTR)
Category I. Safety	
1) Describe design features that address defense-in-depth, accident prevention, accident mitigation, and emergency planning.	A detailed discussion of the Defense-in-Depth approach for the NGNP is found in [10].
2) Provide sufficient design information on the shutdown and decay heat removal systems to allow an assessment of their reliability.	<u>Shutdown</u>: In both concepts, the strong fuel temperature feedback and the large fuel temperature margin during normal operation (the difference between the operating fuel temperature and the temperature at which the rate of failure of fuel particles becomes nonnegligible) is such that the core shuts down in response to any significant temperature increase unless compensated by an external reactivity injection. Indeed, the first step in one of the one of the shutdown sequences proposed for the HTR Modul is to turn off the primary coolant circulating pumps followed by insertion of the control rods. Rods are required, however, the keep the core in a low power or cold subcritical state during maintenance. The MHTGR (prismatic) features two independent sets of control rods (CR), one each in the inner and outer reflector regions. The absorber material consists of 40w/o B_4C granules dispersed in a graphite matrix and formed into annular compacts enclosed in Incoloy-800H tubes. The boron is enriched to 90 w/o B-10. The outer reflector contains 24 rods spaced evenly in the azimuthal direction. These rods are withdrawn in groups of three spaced at 120° intervals to limit azimuthal variations in power. The outer rods are used for reactivity control and power shaping during power operation. Fully inserted they can maintain the core in a Hot Zero Power condition. The rod is subdivided into 18 separate capsules that allow enough flexibility to accommodate any postulated offset between blocks, including seismically-induced shifting. A second set of six rods can be inserted into the inner reflector blocks near the core-reflector interface. These are fully withdrawn during power operation. Fully

Criteria	Small Modular High Temperature Reactor (smHTR)
	inserting the outer and inner reflector CR can keep the core subcritical at low temperatures. A reserve shutdown system (RSS) provides independent shutdown reactivity under all conditions. The RSS consists of natural boron particles embedded in a graphite matrix that is formed into spheres. The B_4C granules may be coated with pyrolytic carbon to limit oxidation and loss from the system during high temperature, high moisture events. When released into the reserve shutdown channel in the RSC fuel elements, the pellets have a packing fraction of ≥ 0.55. For the MHTGR, the weight percent of boron in the spheres is 40% in the MHTGR. In the HTR Modul PBR, there are six independently actuated CR containing B_4C that can be inserted into the side (outer) reflector to maintain and shape power. These are sufficient to hold the reactor at a Hot Zero Power condition. Another 18 channels in the outer reflector can be filled with Coated B_4C spheres to maintain the core in a subcritical state indefinitely at ambient conditions (RSS). <u>Decay Heat Removal</u>: Active core cooling is not necessary for the adequate removal of decay heat during an accident. Conduction and radiative heat transfer will remove heat at a sufficient rate to prevent significant fission product migration out of the fuel element. An unmitigated loss of forced cooling, however, can lead to excessive temperatures in the metallic components (such as control rod assemblies) with possible failure of those structures. In the absence of primary system cooling, the RCCS draws off sufficient heat to prevent damage to the pressure vessel or confinement building. Any available natural circulation of coolant in the primary will decrease the thermal power that the RCCS must dissipate. <u>Inherent Safety Approach</u> The HTGR design requirements specify the HTGR design shall have passive means of negative reactivity insertion and decay heat removal sufficient to place the reactor system in a safe stable state for any anticipated transient events without significant damage to the core or reactor system structure.
3) Describe the expected response of the ARC to normal and abnormal conditions. Demonstrate that the ARC design and associated instrumentation will provide operators with longer times than for current generation LWRs for system diagnosis before reaching safety systems challenge and/or exposure of vital equipment to adverse conditions.	Due to the low power density and large thermal inertia of the mostly-graphite core, all transients are characteristically slow compared to an LWR and none can result in fuel failure. In the bounding (peak fuel temperature) case, the depressurized loss of forced cooling resulting from a large break in the primary coolant boundary, the resulting core temperature rise and negative temperature coefficient immediately halts the fission reaction. Fission product decay causes the core temperature to increase over a span of tens of hours before beginning a slow decline. If no control elements are inserted, the core will go critical after a few days but at significantly reduced power level (usually less than 3 MW and corresponding to the maximum heat removal rate of the RCCS) . Eventually, the core will attain an equilibrium state in which the fission power is balanced by the power radiated to the ultimate heat sink. At no time, does the fuel temperature exceed the level at which significant (actionable) quantities of fission products are released. Pressurized loss of forced coolant flow, while somewhat less severe than a depressurized event, has been demonstrated numerous times in AVR, HTTR, and HTR-10. In the event of a large break in the primary, the reactor building is immediately vented to the atmosphere to reduce the system pressure to ambient levels. (The circulating activity is low enough such that worker and off-site does are well within acceptable levels.) The vents are then closed for the heat-up stage of the transient. At low pressure, any additional releases can be vented through filters. Preliminary safety analysis indicate that the inherent neutronic, hydraulic, and thermal performance characteristics of the HTGR design provide self-protection in <u>beyond-design- basis</u> sequences involving combined earthquake and DLOFC events to limit accident consequences without activation of engineered systems or operator actions.
4) Describe the design features that will reduce the probability for accidents, including accidents with potentially severe consequences. These design	The primary tenet of the inherent safety approach of HTGR is to utilize design features for prevention of core damage and related consequences, even for beyond-design-basis events that involve the failure of multiple safety grade systems The preliminary safety analysis shows that the inherent neutronic, hydraulic, and thermal performance characteristics of the HTGR design provide self-protection to

Criteria	Small Modular High Temperature Reactor (smHTR)
features should provide sufficient reliability, redundancy, diversity, and independence in safety systems to provide for either accident prevention or accident mitigation.	limit accident consequences *without activation of engineered systems or operator actions*. Specifically, the HTGR has the following design and inherent features that minimize the potential for severe accidents and their consequences: 1) Inherent characteristics of the reactor system: low power density, high thermal inertia of the active core, chemically inert gas-phase coolant, total negative reactivity coefficient; large temperature margin to fuel failure and robust fuel form 2) Passive shutdown heat removal system provides sufficient decay heat removal to insure fuel integrity under all anticipated scenarios. 3) Containment function is provided by the multiple coatings of the fuel particle which retain their radiological inventory under all anticipated loss of forced cooling scenarios. The graphite matrix, primary coolant boundary, and reactor building also serve to retain fission products in the event of the most severe events. 4) Chemically inert coolant and graphite are resistant to ingress of oxygen and water in the primary system. Severe water or air ingress may result in the surface degradation of some of the graphite structures (particularly the bottom reflector which is at the highest temperature) if the rate of air ingress is maintained at a high level. Bulk graphite does not burn but surface oxidation can occur under a narrow range of conditions. Circulating graphite dust (present mainly in PBRs with moving fuel) may transport small amounts of contamination which would be released in the event of a pipe break. Graphite dust has a very low combustibility index (far lower than wheat flour, for example). Partial oxidation of graphite may lead to the formation of carbon monoxide (CO) which, if vented to the reactor building, may combine with oxygen and burn under extreme scenarios. CO formation has not been observed in any operating HTGR. Steam ingress may also result in the oxidation of graphite structures and the release of small amounts of radioactive materials into the primary coolant. Both Fort St. Vrain and AVR experienced large core flooding events but recovered fully with no radiological consequences. 5) At least two independent control rod/absorber sphere systems, as described above. 6) In contrast to LWR design, where a combined earthquake event followed by a loss of forced cooling or off-site power would lead to significant distortions in the core geometry and the partial uncovering of the fuel assemblies and possible cladding failure, the same event in these HTGR designs would result in very small core geometry changes (the prismatic blocks and pebbles are already packed tightly, with a peak packing fraction increase from 0.61 to 0.63 expected in case of a severe earthquake). The loss of the helium coolant (i.e. uncovered HTGR fuel) would also not result in any particle failures. In the extreme scenario of exposed HTGR fuel (blocks or pebbles) directly in contact with the atmosphere in the aftermath of such an event, the graphite and TRISO layers act as significant delay barriers to the chemical interaction with either water or oxygen, in stark contrast to current LWR fuel cladding materials that are particularly vulnerable to chemical attack and the production of hydrogen gas. 7) Reactivity insertion events in both these HTGR designs (via control rod withdrawals or water ingress) are much less severe when compared to LWR designs, due to the low power density and total excess reactivity available. This is especially true for the online refueling schemes followed by the pebble bed designs.
5) Describe the design features that will minimize potential radiation exposures to plant personnel.	Radiological exposures are expected to be much lower than those of conventional LWR systems. The total exposure estimate for the MHTGR, as reported to the NRC in DOE-HTGR-86-024, is 149 person-rem which includes a 20% contingency to account for uncertainties. For a 4 module plant assuming 80% availability and estimated staffing levels(1986 assumptions), this translates into 0.38 person-rem/MWe-year or just over 208 person-rem annual dose, a factor of 4-7 lower than an comparable LWR in its day. Through 1983, Fort St. Vrain averaged 0.04 person-rem/MWe-year. Peach Bottom-1 averaged 0.12 person-rem/MWe-year in its last three years of operation [9]. No worker dose data were available for the HTR Modul. The superior performance observed in recent NGNP/AGR fuel irradiation and safety tests, should yield considerably lower doses for an smHTR built today.

Criteria	Small Modular High Temperature Reactor (smHTR)
	The low doses of the HTGR are attributed to the superior retention qualities of the TRISO fuel, graphite matrix, and inert coolant chemistry. A plant built today can also exploit remote technologies and repair features to further reduce exposure.
6) Describe how incorporation of defense-in-depth philosophy is accomplished in the ARC design. Specifically, describe how the multiple barriers are maintained to prevent radiation release, and thereby reduce the potential for, and consequences of, severe accidents.	The HTGR defense in depth approach is outlined in [10]. The principles of defense-in-depth are applied in the design, construction, and operation of existing and advanced nuclear power plants; the NGNP design is no exception. In the design and analysis process for the NGNP HTGR, the “historic” deterministic approach is integrated with a risk-informed performance-based evaluation methodology to ensure that selected design features provide the required level of safety and defense-in-depth. The result is a set of conservative design features combined with inherent reactor characteristics, passive design features, and active systems to (1) prevent transients and accidents, (2) ensure the performance of safety functions, (3) prevent the release of radioactive material, and (4) mitigate the consequences of accidents. The principles of multiple, independent, and concentric barriers to radionuclide transport and release are assessed for each significant source of radioactive material to assure that defense-in-depth has been maintained. In addition, the principles of design margin, redundancy, and diversity are applied in the design of the SSCs that support the required safety functions and serve to support and maintain the integrity and effectiveness of these barriers. The defense-in-depth strategies ensure that TLRC are met, adequate safety margins are achieved, deterministic principles of defense-in-depth are applied, and uncertainties in the reliabilities and capabilities of the SSCs providing the required safety functions are adequately addressed over the life of the plant. The five barriers to radionuclide release that form a functional containment system for modular HTGRs are as follows: 1. The fuel particle kernel, 2. The fuel particle coatings (silicon carbide and pyrocarbon coatings), 3. The core graphite and carbonaceous materials, 4. The helium pressure boundary, and 5. The reactor building. These barriers are described in detail in [10].
7) State how safety and security requirements will be considered together in the design process such that security issues (e.g., newly identified threats of terrorist attacks) can be effectively resolved through facility design and engineered security features, and formulation of mitigation measures, with reduced reliance on human actions.	The HTGR design requirements specify that the reactor shall be designed to minimize the risk of sabotage or proliferation, either through design features, or by proven safeguards and security techniques, or a combination of the two. Security is integrated into the design process. 1) The reactor module is located below grade in a silo structure that is hardened and provides a low profile. 2) The inherent safety features of the HTGR offer a high level of protection against malevolent events, as well as against accidents. 3) Core replacement is done remotely using in-vessel transfer systems and ex-vessel shielded casks in the case of the MHTGR. In the case of the HTR Modul, all fuel handling operations are performed remotely and automatically.
8) Describe the features that could result in a large release of radioactive materials, such as those that would prevent a simultaneous loss of containment integrity (including situations where the containment is by-passed), and the ability to maintain core cooling as a result of an aircraft impact. If prevention of release is not possible under this scenario, identify system designs that would provide a delay in	The MHTGR and HTR Modul design efforts did not include assessments of aircraft impact. The location of the reactor vessel and primary components below grade greatly decrease the probability of an impact.

Criteria	Small Modular High Temperature Reactor (smHTR)
radiological releases to facilitate any required emergency response both on-site and off-site.	
9) Describe the features that will prevent loss of onsite spent fuel storage facility integrity (if part of the ARC design), including consideration of an aircraft impact.	In the original designs, fuel elements discharged from either the MHTGR or HTR Modul core would be stored in water cooled storage pools for up to a year. These pools would be inside the reactor building which is resilient against impact. Beyond 1 year, the elements would be transferred to air-cooled storage containers which are much less vulnerable to failure due to impact. The failure of either the water or air spent fuel cooling facilities would not impact the low power density HTGR fuel form significantly as passive heat removal by conduction and radiation would be sufficient to prevent a large fraction of the more than 1 billion TRISO fuel particles in typical HTGR core to reach temperatures in excess of 1800^{o}C. Recent discussions with vendors indicvate that the short-term water-cooled storage could be replaced by a completely passive, air-cooled system.
10) Identify any R&D that would be needed to bring any of the safety-related technologies used in the design to a sufficient level of maturity to allow for industrial use.	Complete the AGR Fuel and AGC graphite Qualifications programs and Inconel 617 for higher temperatures.. Details can be found in [1].
Category II. Security	
1) Types of special nuclear materials (SNM) present and the security features that provide SNM protection.	The HTGR will utilize TRISO fuel. It is expected that the fuel will be LEU, with enrichment from 8% to 16%. There are two designs variants for the HTGR; they are the <u>prismatic HTGR</u> and the <u>pebble bed reactor (PBR)</u>. However, for the purposes of classifying SNM, the fuel is identical. Therefore, this section will treat the two designs as being identical. The fuel onsite can be classified in four ways. - The operational aspects of reloading the HTGR will be discussed in "Category°X – Nonproliferation." 1. <u>Fresh fuel onsite for loading</u> – This fuel is expected to be Uranium Oxycarbide (UCO) with a nominal enrichment of 15% (actual enrichment could vary from 8% to 16%). 2. <u>Reactor Core</u> – This category includes fuel that will be actively in use in the reactor core. As with other reactors, the composition of this fuel will change over the lifetime of the core. U-235 concentration will decrease and Pu concentration increase over time. - The reactor core will be located below grade. This will make access to the core more difficult for would-be saboteurs and provided added protection for certain attack scenarios, such as aircraft impact and missile launching from off-site locations. 3. <u>Newly Discharged Fuel</u> – There are two scenarios for fuel immediately after it has been removed from the core: - <u>Prismatic Core (MHTGR)</u> - Fuel that is removed from the core will initially be stored in spent fuel pools inside the reactor building. The used fuel pool will be located below grade and the fuel will be highly radioactive – and therefore, highly self-shielding – during the cooling period that it will be stored in the used fuel pool. - <u>Pebble Bed Reactor (HTR Modul)</u> – The details do not exist at this point. However, in the past, the newly discharged pebbles have been cooled in two ways. The original German design discharged the pebbles to cast iron air-cooled storage vessels inside the reactor building. The South African PBMR had the newly discharged pebbles cooling in water-cooled tanks inside the reactor building. 4. Medium-Term Storage – After cooling in the used fuel pool, the fuel will be transferred to medium term storage. There are two scenarios: - <u>MHTGR</u> – After cooling, the fuel assemblies will be transferred to air-cooled containers outside the reactor building. - <u>HTR Modul</u> – After cooling, the fuel will be transferred to air-cooled canisters outside the reactor building.

Criteria	Small Modular High Temperature Reactor (smHTR)
2) Consideration of security requirements together throughout the design process so that security issues (e.g., threats of theft, diversion, and sabotage) can be effectively anticipated, identified, and addressed at an early stage through integrated facility design and engineered security features, and through formulation of response/mitigation measures, ideally with reduced reliance on human actions compared with previous generations.	The HTGR is in an early stage of engineering development. As such, no explicit Physical Protection System (PPS) has been designed. However, it is expected that security experts will be made an integral part of the design/engineering team to ensure that Security By Design is implemented throughout the process in the future. Theft and sabotage scenarios will be considered. Regulatory requirements will be met or exceeded. Hardening of components and systems against both theft and sabotage will be accomplished via a process of interacting with safety systems designers. All aspects of plant design and operation will be considered during the development of the PPS. Cost-Benefit Analysis for design improvements will be included in any scenario analysis. Mitigation of the effects of any potential security events will be considered as part of the PPS design process. Diversion possibilities will be minimized via the use of a transparency process that will ensure regulators have up-to-date information. Information from operations, safety, and security will be shared with safeguards personnel and with the appropriate regulatory authorities to ensure that, overall, a transparent system is in place.
3) Features to prevent/mitigate sabotage threats, e.g., loss of integrity of onsite core fuel and spent fuel storage, including consideration of an aircraft impact and other relevant attack scenarios.	Both the reactor and the short-term storage are located underground, which makes access to the fuel difficult to achieve. In current designs, the fuel will be placed in above-ground canisters for medium-term storage. The plan is to keep the fuel in these canisters until such time this fuel can be shipped off-site to be placed in long-term storage. It may be prudent to revisit the used fuel storage scheme over the course of developing the plant/site design in order to make it easier to provide security at a lower cost. For example, providing a means of storing the used fuel underground for an extended period may decrease the need for armed security personnel. Other aspects of the PPS will be developed throughout the design process of the plant and will include security experts to ensure the protection of any onsite fuel.
4) Features to eliminate or reduce the potential theft of nuclear material.	Both the reactor and the short-term storage are located underground, which makes access to the fuel difficult to achieve. In current designs, the fuel will be placed in above-ground canisters for medium-term storage. The plan is to keep the fuel in these canisters until such time this fuel can be shipped off-site to be placed in long-term storage. It may be prudent to revisit the used fuel storage scheme over the course of developing the plant/site design in order to make it easier to provide security at a lower cost. For example, providing a means of storing the used fuel underground for an extended period may decrease the need for armed security personnel. The high volume fraction of graphite in the fuel elements requires the conspicuous movement of large amounts of material relative to the amount of special nuclear material contained within. Other aspects of the PPS will be developed throughout the design process of the plant and will include security experts to ensure the protection of any onsite fuel.
5) Any features that will require R&D to bring to maturity.	Conventional LWRs become more vulnerable during refueling. Thus, it is common for security experts to assess the refueling procedures for any reactor design. The MHTGR is expected to need refueling every 18-20 months, at which time the fuel blocks will be replaced. However, the fuel handling procedures that have been outlined for this design were developed about 30 years ago; these procedures will probably need to be updated. If so, it would be prudent to engage security professionals in the design process. In this way, the PPS can be developed in harmony with the safety and operational procedures that are required for refueling, thereby avoiding potentially costly security personnel and retrofitting of the plant. The HTR Modul will employ an online refueling methodology. With 15% enriched fuel, about 175 pebbles will be added to the core daily. This process will need to be closely monitored to ensure that theft of the material is not an issue, especially to guard against the insider threat that could exist. The use of SBD to aid in developing these fuel handling procedures and to ensure that these operations do not cause undue stress on other operational aspects of the plant would seem to be a necessary step.
Category III. Compatibility of the concept with traditional and advanced energy conversion systems/processes, maximizing energy production per fuel quantity used	
1) Compatibility between the pressure, temperature, and	Helium is inert and thus wholly compatible with the materials used in the primary system. Recent research indicates that a small amount of oxygen (a few ppm) is

Criteria	Small Modular High Temperature Reactor (smHTR)
chemistry of the ARC coolant and the fluid properties in the energy conversion system.	needed to retain a oxide layer on some metallic components to protect against certain types of degradation. The primary loop operates at around 7 MPA, comparable to a Boiling Water Reactor (BWR), and thus the pressure drops across the steam generator tubes are not excessive. Alloy 800H is currently code qualified for 300,000 hours (supporting 40-year plant life) and up to 760^{o}C in the nuclear section of the ASME Boiler and Pressure Vessel Code. Additional qualification is planned which would extend the ASME code for plant lives up to 60 years. An ASME Code Case for Alloy 617 is being developed for temperatures up to 950^{o}C and lifetimes up to 100,000 hours. The HTTR in Japan operates with a 950^{o}C outlet temperature and 10-year design lifetime, as did the AVR. At the very high temperature of 950^{o}C, structural components constructed from Alloy 617 would have to be designed as replaceable components for a 40-year or a 60-year plant design life.
2) Address whether off-the-shelf commercially available or newly developed energy conversion equipment (e.g. compressors, turbines, seals, bearings, printed circuit heat exchangers, etc.) can be used, or would it be necessary to develop new energy conversion machinery and equipment to satisfy the conditions proposed for the ARC?	Steam Rankine cycle power conversion technology is well developed and the system components are largely off-the-shelf items available from commercial vendors. For printed circuit heat exchangers, structural integrity of diffusion bonds, post-construction examination and in-service inspection, and ASME code stamping are the challenges that need to be overcome in order to allow it to be part of the nuclear boundary.
3) Availability of proven materials for anticipated primary and secondary system pressures and temperatures.	Alloy 800H and 2¼ Cr-1 Mo steel in the solution annealed condition are matured materials that can be readily procured. Alloy 800H is qualified in the nuclear section of the ASME code for up to 300,000 hours and 760^{o}C while 2¼ Cr-1 Mo steel (solution annealed) is qualified for up to 300,000 hours and 593^{o}C. They can be used to support the design and construction of the primary and secondary system of a 750^{o}C outlet smHTR for up to 40-year design life, with Alloy 800H for the hotter sections and 2¼ Cr-1 Mo steel for the colder sections of these systems. For example, some reference designs employ 2¼ Cr-1 Mo steel for the colder part (400^{o}C and below) of the steam generator tubes, with a dissimilar metal weld joining Alloy 800H in the hotter part of the tubes. Extension of these materials in the ASME code to 500,000 hours is required for the full lifetime of a 60-year design. For a 950^{o}C outlet smHTR, Alloy 617 is the construction material of choice for the IHX. Alloy 617 is being qualified for the nuclear section of the ASME code for up to 950C and 100,000 hours. At such a high outlet temperature, the IHX has to be designed as a replaceable component for smHTR lifetime of 40 or 60 years. The planned submittal of the Alloy 617 code case is at the end of FY15 and the code committee approval process generally takes about two years.
4) The potential susceptibility of the conversion system to traditional conversion system challenges, e.g., plugging, corrosion, etc.	Steam system: Ferritic steels are typically used in the steam generator. High purity feedwater is required to prevent deposition of impurities on heat transfer surfaces and to preclude intergranular stress-corrosion cracking of the heat transfer surfaces. This requires careful attention to the design and material selection for each component of the system, i.e., on-line water monitoring, full flow filters, full-flow demineralizers, and pH control. Graphite dust is known to build up in PBRs, mainly from abrasion of pebbles inside the fuel handling system. An AVR, dust accumulation on some of the colder parts of the coolant piping was observed after decommissioning but was not an operational issue. This may be an issue for the HTR Modul if compact heat exchangers are used to drive a Brayton cycle power conversion system Dust in the primary circuit has not been observed in significant quantities in prismatic reactors like the MHTGR.
5) If steam conversion systems are proposed, the compatibility of the ARC coolant with the traditional Rankine energy conversion cycle, in terms of fluid properties, pressure,	All HTGRs operated to date have driven Rankine cycle systems. The two reference systems were designed for the Rankine cycle. As discussed in criterion II.3 above, Alloy 800H will allow extended steam generator operation for up to 40 years and at temperatures up to 760°C. For longer lifetimes, extension of Alloy 800H in the ASME code beyond 300,000 hours is required. Similar extension is required if 2¼ Cr-1 Mo steel in the solution annealed condition is used in the colder section of the steam

Criteria	Small Modular High Temperature Reactor (smHTR)
temperature, chemistry and limits imposed on materials used in energy conversion systems.	generator tubes.
Category III. Ability to improve uranium resource utilization and minimize waste generation	
1) Uranium resource utilization.	The HTGR is configured for high outlet temperature and passive safety, not actinide management.
a) Uranium enrichment required (compared to existing LWR systems).	The reference concepts are designed to run on LEU but at a higher enrichment than LWRs. The fuel being qualified with ARC-VHTR is enriched to about 15%.
b) Design features, if any, that reduce uranium consumption.	The HTR Modul pebble bed core recirculates fuel to achieve maximum burnup for a given enrichment.
c) Is the use of reprocessed fuel planned for the ARC system in its fuel cycle?	Spent LWR fuel can be cast into TRISO particles for additional plutonium incineration (Deep Burn) but this is not envisioned for the reference concepts. The TRISO fuel form can be used to achieve high burnup.
d) What is the expected conversion ratio for the proposed ARC design? Can the ARC system be used for fissile material breeding?	CR<70%. Peach Bottom-1, Fort St. Vrain, and THTR all ran on a fissile-fertile mixture of thorium and highly enriched uranium to achieve high conversion ratios but this is not envisioned for the reference smHTGRs.
e) What are the R&D needs?	While some concepts have been demonstrated at lab scale, tf reprocessing is pursued, some development of fuel disassembly technology at larger scale may be required.
2) Estimate of waste generation (qualitatively compared to a once-through LWR). a) Ability to transmute long-lived products in nuclear waste: those produced in situ during reactor operation.	The HTGR is a thermal spectrum reactor running on LEU so actinide buildup will be comparable to an LWR. The HTGR spectrum is suitable for plutonium incineration in a so-called Deep Burn configuration but higher actinides will continue to accumulate. The substantially higher burnup achievable in TRISO fuel allows greater energy production from an equivalent input of heavy metal in an LWR. Spent fuel characteristics of the HTGR will not be significantly different from the LWR of comparable burnup.
b) Mass of discharged materials	The MHTGR running on AGR fuel achieves a burnup of about 100 MWD/kg. The HTR Modul recirculates the fuel and thus each pebble stays in the core until the target burnup level is reached, which for AGR fuel is about 160 MWD/kg. Online refueling also allows for a higher capacity factor (>90%).
c) Mass (qualitatively compared to an LWR discharge) of low heat, long-lived materials (examples Carbon 14, Technetium 99, Iodine 129).	C-14 discharges will be considerable higher than an LWR due to the buildup in the graphite(from transmutation of nitrogen in impurities). Fission products will be somewhat lower on a per MWD basis because of the higher efficiency and burnup.
d) Mass (qualitatively compared to an LWR discharge) of low heat, low longevity materials (Class A, B, C low-level waste (LLW).	Discharge of low heat materials will be considerable higher than an LWR due to the large volume of graphite.
e) R&D needs to facilitate transmutation or other waste management goals.	Deep Burn technology has the potential of reducing plutonium concentrations in the spent fuel. Graphite recycling technology can reduce the amount of carbonaceous material going into a repository.
Category IV. Operational capabilities and aspects such as control strategies, operating modes (e.g., base load versus load following capability), maintenance and inspection requirements, refueling interval, etc.	
1) Flexibility in electricity generation including the proposed ARC capability for load following (the capability to adjust generation as demand for electricity fluctuates). Limitations, if any, on such operation arising from considerations of fuel	The large thermal inertia of the smHTGR core that provides confers many safety benefits also precludes rapid (on the order of minutes) response to changes in load. Slower fluctuations in demand can be accommodated. Many smHTGR concepts are designed to produce multiple energy products (electricity, process heat, hydrogen, etc.) which may allow for rapid shifting between parallel loads. An OECD transient benchmark project focused on the PBMR-400 (MWt) modular design specified a 100%-40%-100% load follow operation with the power-down and power-up ramps each occurring over a 6-minute interval. A similar project for the MHTGR specifies a %100-%50 power ramp occurring over 5 minutes to stimulate a

Criteria	Small Modular High Temperature Reactor (smHTR)
performance, reactivity limitations, mechanical and thermal stress in materials and components should be addressed.	xenon oscillation. In the 15% to 100% power range, power output is adjusted by varying the position of the helium bypass valves over the short term (0 to 20 min) and by varying helium inventory over the long term (0 to 60 min). Temperature is maintained constant throughout these operations due to the inherent effects of the negative temperature coefficient of reactivity and by small adjustments of the control rod position. The systems, structures, and components in each of the reference concepts are designed for such operation.
2) Other features allowing utilization beyond base-load electricity production, for example process heat generation, high temperature operation for hydrogen production via chemical splitting, etc.	Both of the reference concepts are well-suited for applications 'beyond the grid'. The high outlet temperature is uniquely suited to driving a number of process applications such as hydrogen production (chemical or electrolytic), steam reformation of methane, and coal-to-liquids conversion. The extremely low fission product release rate is amenable to collocation of the process heat facility; the facility is close enough for efficient heat transport but still outside of the emergency planning zone.
3) Features expected to improve availability in operation as estimated from the system's capacity factor, frequency of outages for refueling, and other planned outages (compared with those for an LWR). Include in the analysis of operational availability/dependability, elements from the information requested under Category V that follows, which includes concept maturity and operating experience (if available) associated with the proposed ARC.	The 80% capacity factor quoted for the MHTGR is based upon the industry average in the 1980's when the design was submitted to the NRC. Modern operational procedures and operating experience are likely to increase availability. Refueling and maintenance operations are not considerably different than an LWR. The HTR Modul features online refueling and thus only needs to be shut down for maintenance and repairs. Capacity factors well above 90% would be attainable.
4) Maintenance and operation – Are there features that will allow easier maintenance to reduce the duration and frequency of outages? Are there special requirements for maintenance and inspection that are different from current LWRs (simpler or more complex)?	Helium is transparent and would allow for in-service inspection using remotely operated instruments. Requirements would be similar to those of LWRs. The relatively low fission product release rates, however, will likely result in reduced worker does during such operations.
5) Describe design efforts to provide reliable equipment in the BOP (or safety-system independence from BOP) to reduce the number of challenges to safety systems.	Some small HTGR concepts have been designed with an intermediate gas cooling loop as an additional buffer between process and the reactor but this choice is mainly one of economics. Dynamically, the reactor core is loosely coupled to the secondary and thus a severe BOP transient has minimal effect on core safety. With a Rankine cycle, reliability is expected to follow LWR practice although no specific reliability studies are cited.
Category V. Concept maturity, operating experience, unknowns and assumptions (e.g., availability of advanced materials, fuels, etc. currently under development).	
1) Description of the general level of ARC design maturity (pre-conceptual, conceptual, or detailed). Identify the proposed schedule for completion and initial operation of the proposed ARC.	Both the MHTGR and HTR Modul designs are in the preliminary design stage. Specific systems are based upon decades of technological maturity and experience.
2) Description of Technology	See the discussion on Technical Readiness in the previous section.

Criteria	Small Modular High Temperature Reactor (smHTR)
Readiness Levels (based on DOE TRL definition in DOE G 413.3-4, U.S. Department of Energy Technology Readiness Assessment Guide) of major technologies and systems and their relation to previous operating reactors. Identify the overall TRL of the proposed ARC, which should be based on the TRL of the least ready major technology or system.	
3) Applicable experience from other reactor systems (test, research, demonstration reactors, naval reactors, foreign reactors) such as design elements, component testing and demonstration).	The lessons learned from high temperature reactor programs in the U.S. and worldwide and the operating experience of seven demonstration and commercial reactors have been incorporated into the design of the smHTR to the extent possible. International collaboration under Gen IV International Forum, and bi-lateral agreements has enhanced the transfer of HTGR/VHTR fuel and material technologies.
4) Status of applicable design and analysis tools.	Codes developed for the 1st generation HTGR programs (MICROX/DIF3D for prismatic reactors and VSOP for PBRs) are available but are being supplanted by modern tools including high fidelity neutron transport and computational fluid dynamics (CFD). Although the old codes were used successfully to license the old plants and the newer codes show great promise in terms of accuracy and resolution, any code used for licensing an smHTGR will require formal validation and verification.
5) Discussion of the assumptions made regarding the expected ARC performance (associated with unique or unproven aspects of the design) and the basis of those assumptions, including identification of uncertainties.	Design features of the MHTGR and HTR Modul were chosen originally based upon tradeoff studies and analyses by the vendors, often in conjunction with national laboratories in the US and Germany and greatly influenced by the operating experiences of the early demonstration and commercial plants. The latest releases of "industry standard" codes like RELAP-5 and SCALE have already been modified to incorporate features that are unique to HTGR designs (double heterogeneity, helium and graphite material properties, etc.).
6) Identification of major technology issues, R&D needs to address design and operational uncertainties, and technology gaps.	Design Data Needs, issued in reports by the vendors, and reconciled with NGNP research and development objectives were commissioned by the NGNP Project.
7) Estimated time frame to develop the needed information identified in Item 6 above.	The remaining R&D efforts could be completed in the decade required to build an initial smHTGR plant.
Category VI. Fuel Cycle Considerations	
1) Ore mining and conversion requirements (qualitatively compared to the once-through LWR cycle).	Comparable. The smHTGR requires a higher enrichment than an LWR but also achieves a higher burnup of the fuel, particularly the recirculating pebble bed version.
2) Fuel fabrication (compared with LWR fuel).	Fuel fabrication is somewhat more complicated than an LWR. A sol-gel process is used to manufacture the UCO kernels followed by a multistep particle coating deposition process and compaction in a graphite matrix. Although TRISO-based fuel was manufactured for the 1st generation plants, production-scale capability in the US was lost. Under the NGNP, a fuel vendor (B&W) participated in the development and qualification of a new fabrication technique.
3) Fuel form experience base (as needed for licensing/ certification) for fuel forms different from current UO2 fuels.	High quality UO_2 and UCO TRISO fuel was manufactured for the German HTGR program and, with somewhat less success, for the early US program. Data from those efforts is still marginally useful today but the licensing of a new reactor will rely heavily on the testing that began in 2004 under the AGR (NGNP) program.
4) Are the systems currently used	To some extent. Once- through smHTGR spent fuel can be stored safely in qualified

Criteria	Small Modular High Temperature Reactor (smHTR)
for managing used fuel/ waste in LWRs applicable?	transport and storage casks. The lower decay heat power density of the large volume HTGR fuel along with the robust fuel form allows the fuel to be transferred more quickly from water-cooled storage to air-cooled containers.
5) Is a reprocessing capability required? If yes, what type of technology is needed, has it been proven/demonstrated, and what are the waste forms?	Reprocessing is not required. HTGR fuel can be reprocessed but this requires additional steps to remove the graphite element, break the coatings, and leach out the kernel materials for subsequent reprocessing.
6) Discuss any unique features/aspects of processing/storage/transportation of used fuel, high level waste (HLW), or LLW.	The particle coatings and matrix also provide an additional barrier to FP release. The volume of spent fuel, per MWe generated, however, is much higher owing to the graphite matrix that forms the bulk of the fuel element. The decay heat density is, however, comparably lower. Depending upon the limits and costs of spent fuel management, it may be desirable for the operator to have the coated particles separated from the graphite element to reduce HLW volume. The graphite may be decontaminated and re-used (research is underway particularly in Europe).
7) Describe how the ARC is compatible with IAEA safeguards guidance of reactor and associated fuel cycle facilities.	Current guidance is applicable. Safeguards approaches for LWR fuel can be applied to the prismatic HTGR [11]. The HTR Modul, being fueled online with small pebbles, has unique features with regard to safeguards [12]. • The pebble fuel HTGR is an on-load refueled reactor, and in this respect is more akin to the Canadian Deuterium/Uranium (CANDU) reactor • The pebble fuel assemblies are more small and numerous, and consequently are more challenging to count due to the greater number in the core (several hundred thousand) • Startup of the pebble fuel HTGR core requires an initial loading of graphite moderator pebbles, which are drawn down as the reactor reaches nuclear equilibrium • Verifying pebble fuel and distinguishing it from pebble moderator is a new safeguards issue • After the pebble fuel HTGR reaches nuclear equilibrium, spent fuel is continuously removed and fresh fuel transferred to the core to take its place • The flow of fuel to and from the pebble fuel HTGR core follows a more elaborate flow scheme and requires greater attention in monitoring fuel transfers, because the pebble fuel is smaller and the movements concealed below the reactor building floor • Because of the greater number of inventory and flow key measurement points (KMP), the pebble fuel HTGR requires containment and surveillance measures at more locations (relative to an LWR).
Category VII. Assessment of market attractiveness (e.g., efficiency, initial capital costs, application beyond electricity generation, etc.)	
1) Energy products of the ARC (e.g. electricity production, desalination, process heat, hydrogen production, etc.) and its power (thermal, electric) output and/or product output.	The high outlet temperature of the smHTGR is well-suited for providing a wide range of process heat-driven applications. The technical and economic feasibility of integrating the HTGR into various industrial processes was assessed and documented in[13]. These applications included: seawater desalination, ex situ and in situ oil shale retort, oil sands recovery, coal-to-liquids and gas to liquids conversion. The economic viability of the nuclear-driven process varies with the application and none are currently competitive with fossil fuels for providing heat, in some cases such as coal-to-liquids conversion the HTGR-driven process can compete with natural gas at historic levels.
2) Expected thermal to electric conversion efficiency, and overall multi-use plant efficiency.	The thermal-to-electric efficiency is ~40% when driving a conventional steam cycle.
3) Estimated capital and operating costs (compared with current LWRs).	A cost model was developed with input from reactor vendors as part of the NGNP project. Capital, operating, and decommissioning costs were estimated for an 850C HTGR with either a 350 MWt or 600 MWt output and single or '4-pack' deployment.

Criteria	Small Modular High Temperature Reactor (smHTR)
	These were documented in[14] and shown in Table 4.
4) Estimated siting requirements (e.g., less water usage or accident consequences may favorably impact siting requirements).	The HTGR is suitable for deployment at a wide variety of sites. The source term will be lower than that of an LWR as will cooling requirements on a per MWt basis. The reactor vessel will be seismically isolated. Dry cooling has been evaluated as an option for HTGR deployment in arid regions. This reduces the plant output. As originally designed, the MHTGR can withstand a maximum free-field acceleration of 0.15g during operation and 0.30g for a Safe Shutdown Earthquake9. As originally designed, the HTR Modul can withstand a maximum free-field acceleration of 0.15g during operation and 0.30g for a Safe Shutdown Earthquake. A scoping evaluation of siting an HTGR cogeneration plant at an existing US nuclear plant site was performed an documented in [15]. Similar evaluations were performed for siting an HTGR at other sites such as an existing petrochemical plant.
5) Environmental impacts under normal/abnormal conditions, including severe accident conditions, and from spent fuel arrangements (as compared with current LWRs).	The smHTGR will meet siting regulations specified in NRC's 10CFR100 and section 8.c.7 of DOE order 5480.30. It is expected that the small reactor source term will be much smaller, resulting in reduced environmental impact (but comparable to an LWR on a per MWt basis).
6) Competitiveness on international markets/export potential. Specifically, what ARC features could make it desirable to a foreign customer?	The smaller output would make the smHTGR more attractive for nations with lower capacity grids and expected growth. Co-generation of electricity and process heat may be desirable for remote industrial operations such as oil sand recovery in northern Canada. Water desalination options using smHTGR designs are particularly attractive to arid Gulf states. The smaller emergency planning zone enables collocation of industrial operations.
7) Derived technologies arising from ARC development.	Hydrogen production was a primary goal of the early NGNP project and various hydrogen production technologies are still being developed with HTR projects around the world. Hybrid energy systems (electricity + process heat) developed in conjunction with HTGRs continue to attract research funding. Accident-tolerant TRISO fuel is being explored for other advanced reactor concepts.
8) Unique features.	High outlet temperature, low power density core with high thermal inertia, melt-proof fuel, very low fission product release rate enabling a reduced emergency planning zone suitable for collocation of industrial operations.
9) Expected time frame of introducing the ARC to the market.	The schedule for deployment of the first NGNP was most recently updated in 2011 and specified a span of 17 years between the commencement of R&D and licensing activities and full operation of the first HTGR, with R&D activities continuing through to the end particularly in fuel qualification. Although the expected level of funding for the project did not materialize, considerable progress in R&D and licensing has occurred. Under the schedule projected from 2013, fuel qualification will be largely completed by 2021. About 5 years are needed for the fuel vendor to set up the production line and fabricate the first core. Assuming that R&D, licensing, construction and fuel production activities are conducted in parallel, the first plant could operate in 8-10 years.
Category VIII. Economics (including construction, manufacturing, and operating costs, uncertainties)	
1) The extent and ability to use pre-fabricated modular construction for plant structures and systems. The materials and features of proposed modules that would improve the ARC economics.	Modular construction techniques will be used to the extent possible. The vessel for a 350 MWt HTGR is still large (comparable to an LWR) as are some of the other components so complete factory fabrication is unlikely. Helium circuit components are smaller than comparable LWR pumps and turbines.
2) ARC improved economics by simplified design compared with LWRs (e.g. length of piping, electrical cables, valves, number of loops, pool design, etc.).	The primary advantage arises from the elimination of active safety systems. Further savings can be realized in that the balance of plant components are smaller than those used in a large LWR.

Criteria	Small Modular High Temperature Reactor (smHTR)
3) Cost of nuclear fuel.	The table under Estimated Capital and Operating costs (from [14]) included values for the fuel.
4) Cost of major components. Need for special materials and/or construction methods (how many large vessels and pipes have to be fabricated, and how large would they have to be?).	Reference [14] includes estimates of itemized costs for major components. Except for the graphite which is readily available, LWR materials and construction methods can be employed.
5) Estimated construction schedule (as compared with LWRs).	For the original MHTGR design, a 33 month construction schedule was assumed.
6) Special skill sets and/or procedures required for construction and their availability.	The first smHTGR will require special skilled labor during construction since this is first-of-a-kind facility. Plant systems and components related to the helium coolant, instrumentation and control (I&C), special core design, shielding, special fuel transfer and handling, etc., are substantially different than LWRs. The labor force will have to be trained. The technology and skills to manufacture the fuel is being developed in cooperation with a commercial fuel vendor.
7) Estimated overnight capital cost.	See the table listed under Estimated Capital and Operating costs (from [14]).
8) Estimated yearly operational cost (accounting for decommissioning and waste management).	See the table listed under Estimated Capital and Operating costs (from [14]).
9) Estimated cost of electricity.	The HTGR cost model developed under the NGNP project estimates the cost to produce electricity with an HTGR to be between 70 and 80 $/MWhe [16]. The estimated cost to produce steam with an HTGR is 10-15 $/1000 lbs.
Category IX. Potential regulatory licensing environment (advantages and uncertainties/risks)	
1) A description of the licensing approach envisioned for the proposed ARC. This would include the general applicability of current regulatory requirements (10 Code of Federal Regulations (CFR) 50, 52) and guidance documents (e.g., NUREG-0800 and Regulatory Guides) to ARC design, construction, and operating licensing.	The overall licensing approach for the NGNP was developed by a joint DOE and NRC working group, to establish a viable licensing path that would support the deployment schedule directed by EPAct 2005. That agreed upon licensing strategy and approach was summarized in a report to Congress [17] in 2008. The NGNP project team then developed a more specific and detailed plan for implementing the licensing strategy. This NGNP Licensing Plan [18] from 2009 identified and prioritized the plan for addressing the key HTGR licensing policy and technical issues, and established an approach for developing the required license application content. It has provided the basis for the NGNP licensing activities and NRC interactions implemented during the past five years. As a part of its implementation of the Plan, the NGNP team performed a comprehensive regulatory gap analysis, which included a review of over 2600 regulations and associated NRC regulatory guidance to identify areas requiring revision and/or adaptation to support NGNP licensing. The results of that analysis [19] concluded that a relatively small number of regulations would require revision/update to support modular HTGR licensing.
2) ARC design/operational features that positively impact licensing requirements (reduced radioactive inventory, enhanced passive safety, low-pressure operation, etc.).	The design is consistent with the NRC's Advanced Reactor Policy Statement (the incorporation of inherent or passive means for reactor shutdown and heat removal, longer time constants, simplified safety systems which reduce required operator actions, etc.) Specifically, the HTGR designs for NGNP utilize the following inherent material properties: • Helium coolant – neutronically transparent, chemically inert, low heat capacity, single phase • Ceramic coated fuel – high temp capability, high radionuclide retention • Graphite moderator – high temp stability, large heat capacity, long response times In addition, the modular HTGR was developed as a simple modular reactor design with passive safety, including the following characteristics:

Criteria	Small Modular High Temperature Reactor (smHTR)
	• Radionuclides are retained at their source within the fuel • The reactor is sized and configured for passive core heat removal from reactor vessel with or without forced or natural circulation of pressurized or depressurized helium primary coolant • There is a large negative temperature coefficient for intrinsic reactor shutdown • There is no reliance on AC-power • There is no reliance on operator action and the design is insensitive to incorrect operator actions Additional information regarding the safety basis for the modular HTGR can be found in [10].
3) ARC design/operational features that have not been subject to the licensing process for the current fleet of LWRs, or if the proposed ARC design/operational features do not include features typically found in LWRs.	There are a number of design/operational features of the modular HTGR that have not been previously licensed in association with the current fleet of LWRs. These features and the path to successful licensing were the primary subject of the joint NRC-DOE Licensing Strategy (Ref. A), which established an approach for adapting the existing regulatory requirements to address the modular HTGR. In particular, the proposed approach, which has been implemented by NGNP, addresses the following key items that are different from LWR fleet licensing: • Establishment of a design and licensing basis event identification process that is based on a risk informed performance based approach • Definition and implementation of a mechanistic source term concept for establishing the types and levels of radionuclide releases that must be considered from licensing basis events One feature found in all LWRs that is not included in the modular HTGR design and licensing concept is a pressure retaining primary containment structure. Protection of public health and safety is instead provided by a functional containment that relies on the radionuclide retention capabilities of the TRISO fuel (kernels, coatings, and graphite element) in conjunction with additional radionuclide release barriers that include the helium pressure boundary and the reactor building. This approach and the specific NGNP proposals in the above areas have been extensively reviewed by the NRC staff, and they have developed a set of draft assessment reports that conclude that the NGNP proposals are generally reasonable. The NRC’s formal issuance of these reports is expected in the near term.
4) Applicability of current codes and standards and possible development required.	For elevated temperature metallic pressure boundary components and core support structures, the applicable structural design code is the ASME Boiler and Pressure Vessel Code, Section III, Division 5, Subsections HB, HC and HG and applicable code cases. Currently, rules in Division 1, Subsections NB, NG and NH are referenced in these Division 5 subsections. There is plan to directly incorporate Subsection NH into Division 5 and simultaneously delete Subsection NH. The applicable design rules for the short term elevated temperature excursions during loss of flow and loss of coolant accident scenarios of the actively cooled RPV made of low alloy steels in the reference design are provided in Division 5 Subsection HB, Subpart B appendix. As described in previous criteria sections, Alloy 800H and 2¼ Cr-1 Mo steel (solution annealed) are currently qualified in Subsection NH, and hence by reference, in Division 5 for up to 300,000 hours and 760^{o}C and 593^{o}C, respectively. Extension to beyond 300,000 hours for these two materials is required to support design lifetimes that are greater than 40 years. Recently, concerns have been raised by ASME code committees on the applicability of existing simplified design methods near the upper temperature limits of Subsection NH materials (including Alloy 800H and solution-annealed 2¼ Cr-1 Mo steel). R&D on development and qualification of appropriate simplified design methodologies for these two materials is required. There is also a lack of inelastic analysis methods for these two materials; hence, R&D on unified constitutive model development and finite element implementation is required. Alloy 617 is being qualified for up to 950^{o}C and 100,000 hours for Subsection NH

Criteria	Small Modular High Temperature Reactor (smHTR)
	applications to support a 950°C outlet ARC. It is anticipated that the code committee approval process will be completed by FY17. R&D on design rules and methods for post-construction examination and in-service inspections for compact heat exchanger design is needed. R&D on design rules and analysis method for dissimilar metal weld joints to support steam generator design is required.
5) Applicability of current analysis tools and data (new R&D needed).	Long term creep, aging, environmental and crack growth data are needed for Alloy 617, Alloy 800H and solution-annealed 2¼ Cr-1 Mo steel, and their weldments (including dissimilar metal weld joints), to support NRC licensing. The actively cooled RPV is designed to operate at temperatures below the ASME code temperature boundary of 371°C but higher than the operating temperatures of light water reactors, hence these data are also needed to support NRC licensing. Long term emissivity data for reactor pressure vessel materials in impure helium and neutron irradiation environment are needed to support the adequacy of passive decay heat removal during accident scenarios. Tools developed for LWR safety analysis have been modified for HTGR use. Performance against available benchmarks and experiments indicates that these tools can be successfully deployed. It is likely, however, that the regulator will require additional validation and verification of the tools before they will accept the results as part of a license application. Some validation experiments began under the NGNP program and continue at a modest level. Certain phenomena and scenarios, such as natural circulation heating of the upper vessel head during a pressurized loss of forced cooling, cannot be modeled adequately with traditional system codes. These require computational fluid dynamics codes which have not been accepted by the NRC for HTGR safety analysis. Without such analyses and associated standards, the extra conservatism will need to be built into the design. NRC has sponsored the development of a tool set that can assess the coupled neutronics and thermal fluids behavior of both prismatic and pebble bed HTGR systems at the University of Michigan. These tools were applied in both international code-to-code benchmarks that were conducted under the auspices of the OECD for pebble bed (2005-2010) and prismatic designs (2011-2015).
6) Knowledge base and skills of NRC staff to address ARC design and licensing.	Experimental validation of safety analysis codes for loss of forced cooling events and ex-core heat removal. A consortium of universities and Argonne National Laboratory are completing experiments that re-create the performance of the Reactor Cavity Cooling System. The university work on air-and water-cooled RCCS code validation will be completed by 2016. ANL is completing experiments on air-cooled RCCS after which the facility will be reconfigured to validate water-cooled simulations. Tests using the High Temperature Test Facility at Oregon State University will begin later in 2014. These tests are funded mostly by the NRC and will re-create the conditions following various types of breaks in the primary system Loosely-related experiments in bypass flow, air ingress, core heat transfer, and plenum-to-plenum heat transfer are underway or are planned. Complementary analyses with CFD and the RELAP5-3D system code are being performed at INL and ANL. These two efforts will provide much of the data needed to validate smHTR simulations as long as they are completed. The NRC may require further experiments to validate system and CFD codes for specific phenomena or sceanrios. The codes used in a license application will require this data for validation and further work for verification.
7) Estimated validation and verification effort (tests and computer codes).	HTTF, NSTF methods budget
8) Identification of any additional regulatory activities or products, such as previous NRC reviews or research efforts, that could enhance the licensability of the ARC.	The NRC has previously engaged in a number of efforts that enhance the capability to license smHTRs. Those NRC efforts have included: • Licensing and operational oversight of the Fort St. Vrain facility (late 70s through early 80s) • Interactions with DOE regarding the proposed New Production Reactor

Criteria	Small Modular High Temperature Reactor (smHTR)
	• Review of the GA Modular High Temperature Gas Cooled Reactor (MHTGR) license application material, which resulted in issuance of a draft Safety Evaluation Report (documented in NUREG-1338, issued in 1989 and updated in 1995) • Engagement in Exelon pre-licensing efforts in 1999 – 2001 associated with licensing a pebble bed modular reactor • Engagement with Pebble Bed Modular Reactor (PBMR) Ltd. pre-licensing activities in 2005-2006 • Joint development, with DOE, of the NGNP Licensing Strategy – A Report To Congress (August 2008) • Development and implementation of the NRC's Advanced Reactor Research Plan (March 2011) – focused primarily on HTGRs and reviewed by NRC's Advisory Committee on Reactor Safeguards • Significant interactions with the NGNP project (2008 – 2013) to address and resolve the key licensing and policy issues associated with modular HTGRs. It is noted that NRC plans to issue a series of assessment reports in the near term documenting the results of its reviews of the NGNP project pre-licensing proposals. Those reports will provide key inputs and insights regarding advanced (non-light water) reactor licensing that will likely be very useful in the development of both the smHTR and other advanced reactor design types.
9) The effect of unique fuel configurations on the licensing requirements for storage of spent nuclear fuel. In addition to the relevant regulatory requirements in 10 CFR Parts 20, 50, and 52, the applicant should address any unique issues of how the requirements of 10 CFR Part 72 would impact long-term storage of spent nuclear fuel.	Currently, There are no unique fuel configurations for smHTRs that would create issues associated with the requirements of 10 CFR Part 72. It is noted that coated particle fuel waste characteristics are different than those for LWR spent fuel. NRC regulations 10 CFR 51.51, Table S-3, "Uranium Fuel Cycle Environmental Data" and 10 CFR 5 1.52, Table S-4, "Environmental Effects of Transportation of Fuel and Waste" address LWRs but not HTGRs.
Category X – Nonproliferation	
1) Characteristics of the fresh and spent/used fuel.	An UCO-fueled HTR Modul fuel element is a 6cm (o.d.) mostly graphite pebbles (184.5 cm^3 effective volume and 205 g each) containing 19,000 coated particles of LEU. Each fresh pebble contains 7.1 g of 14% enriched uranium (~1 g U-235). To get a significant quantity of U-235 (75 kg) one requires about 75,000 pebbles. A single reactor contains < 400,000 pebbles during operation or about 5x an SQ of U-235. With 15% enriched fuel, the HTR Modul would consume approximately 180 fresh pebbles per day and discharges the same amount of spent pebbles to its Used Fuel Container. Each pebble discharged from the core (~166 MWD/kghm) contains roughly the following: U-235: 48 mg Pu-238: 7 mg Pu-239: 42 mg Pu-240: 28 mg Pu-241: 19 mg Pu-242: 28 mg Pu-total: 0.172 g U-238 and Mixed Fission Products: ~6.88g All of the pebbles discharged from the core will have a high radiation dose and can be expected to remain on site in the Used and Spent Fuel Containers.
2) Other design characteristics that impact the materials control and accounting system (and whether significant development of a	The HTR Modul and the MHTGR are in an early stage of engineering development; as the designs progress there will be more detail on how appropriate materials, control, and accounting methodology will be established to ensure that fuel is protected from theft or diversion.

Criteria	Small Modular High Temperature Reactor (smHTR)
materials control and accounting methodology will be needed).	Conceptually, verification of the nuclear content of fresh pebble fuel casks can be accomplished using NDA verification systems similar to those used for LEU fuel and MOX materials. The gross and net weight of pebble fuel casks may be accomplished using load cells or operator weighing systems. A primary means of spent fuel verification could be established using a pebble fuel flow monitor, this would be used to distinguish spent fuel from damaged, irradiated, and fresh pebble fuel and graphite pebble moderator. Spent fuel could then be verified to detect the substation of damaged, irradiated, or spent fuel with a non-fuel object (such as graphite pebble moderator). As the design of the HTGR reactors progresses, the materials, control and accounting methodology will be developed along with details of the operational and safety aspects of the plant[11,12]. This will ensure that the SNM will be more than adequately protected.
3) Operational concept for the design as may impact proliferation risk.	A government which plans to divert nuclear material may find the continual loading and unloading of pebbles in the HTR Modul to be an attractive source of material. If the plant were operated by the state, the unloaded pebbles could potentially be diverted and replaced with a different material. One way of decreasing the likelihood of diversion success is to ensure that the regulatory agency that monitors nuclear materials (IAEA) has a way of verifying that the particular fuel pebble that is supposed to be in the used fuel storage area is actually there. One approach would be to develop a way to uniquely mark each pebble so that it can be identified while in storage, i.e., by making each fuel assembly "inspectable." For large HTRs, however, this may be difficult. Another way would be to use a 'item counting' protocol in which each pebble is counted as it leaves a major structure (core, used fuel vessel, spent fuel vessel, etc.) Frequent refueling can also allow a government frequent access to the interior of the pool near the reactor core. This could potentially provide easy access for a government to irradiate material for future use as a weapon. This is most effectively considered via the use of a safety, security, and safeguards by design (3S by Design) program. 3S by Design advocates developing a team of professionals with expertise in each of the three areas of concern to work together from the conceptual design phase to ensure that a globally optimized 3S program is developed via the use of Systems Engineering principles. The use of SBD to aid in developing these fuel handling procedures and to ensure that these operations do not cause undue stress on other operational aspects of the plant would seem to be a necessary step.
4) Relevant integral nonproliferation and security perspectives (e.g. material attractiveness of fuel considered in the context of anticipated security features/operational concept).	One issue of concern here is the potential to remove HTR Modul pebbles by the state in order to obtain a significant quantity of nuclear material for development of a weapon. The material accountability aspects of online refueling should be considered in detail and in concert with safety, security, and operations design personnel.
Category XI – Research and Development Needs	
1) A description of the R&D needs that could reasonably be supported by a national laboratory	<u>TRISO Fuels Qualification</u> – the TRISO fuel being qualified under the DOE's AGR fuel program underpins all HTR commercialization efforts. The AGR-1 and -2 irradiations are complete and demonstrated the performance of the fuel particles fabricated using laboratory-scale and production scale coaters. AGR-3 and AGR-4 will demonstrated the behavior of fission products (and their release from kernels) under normal irradiation conditions and within various graphitic materials. AGR-5 and AGR-6 will be used to qualify the fuel for the first core while AGR-7 and AGR-8 will provide additional data for validating fuel performance codes. Samples from these tests are also subjected to heating in a furnace to measure performance under accident conditions. The fuel targets used in many of these tests are of a 'compact' geometry, suitable for prismatic cores. For use in pebble fuel, an additional irradiation

Criteria	Small Modular High Temperature Reactor (smHTR)
	of a spherical target will need to be performed. Graphite Qualification – The non-irradiated baseline characterization program focuses on developing a statistically valid material database for each of the graphite types selected for irradiation testing. This will establish baseline values for material properties that can be used to determine the quantitative changes during irradiation. Determination of grain size, morphology/anisotropy, and pore size/distribution within the graphite is critical to determining macroscopic physical, thermal and chemical (oxidation) behavior. And mechanical properties. In addition, a key technological deficiency is the inability to determine microstructural features within a graphite material, specifically with non-destructive techniques. This is important for determining not only the evolution in test specimen microstructures as a function of irradiation but also for determining defects within the large graphite billets. A large irradiated specimen population will then be exposed to the expected HTR reactor environment (temperature and dose) in a series of (AGC) irradiations. The resulting graphite materials property database will include values for thermal and mechanical properties for a number of grades at anticipated low, medium, and high doses. High Temperature Alloys – Testing is needed to qualify selected alloys and weldments for extended HTR duty at various temperatures. *Pressure Vessel* - For outlet temperature below 850C, current designs can use of LWR steel (SA508/533) for the pressure vessel, perhaps with minor modifications to the coolant pathways. Above 850C, alloys (Grade 91 or 2.25Cr-1Mo-V would need to undergo significant testing. For Grade 91– in parts of code up to 649°C, but for NH pressure vessels up to 371°C, the program must - Define insignificant creep. - Address heavy section products. - Determine weld reduction factors. - Define allowable excursion conditions. - Investigate weldability of thick sections - Identify a manufacturer and maximum ingot size - Characterize environmental and aging effects For SA508 – in code case N-499-1 for general operating temperatures below 371°C , the program must define allowable excursion conditions. For 2.25Cr-1Mo-V – not code qualified at all with Vanadium (needed for thick sections), the program must: - Define insignificant creep. - Address heavy section products. - Determine weld reduction factors. - Define allowable excursion conditions - Confirm vendor weldability processes. For all alloys, inspection requirements must be established. Emissivities must be confirmed. *High Temperature Metallic Applications* – the following materials must be codified for extended use in steam generator tubes, control rod guide tubes, and intermediate heat exchangers (IHX). For Inconel 617 (for the IHX) – A draft code case exists with allowable temperatures up to 982C and 100,000 hours. The draft case is being developed further in preparation for inclusion in the code. The code case will go in for ballot at the end of FY15. For Alloy 800H (Control Rods, Internals), there is an on-going effort to extend temperature range above 760C. The operating temperature definition may include short time, very high temperature excursions.

Criteria	Small Modular High Temperature Reactor (smHTR)
	The program must also: - Develop the equivalent to DOE NE F9-5T for guidance on inelastic design analysis methods for nuclear components - Develop creep-fatigue life prediction methodology for base and weld metals and overall weldments corresponding to the selected joining processes for IHX fabrication, considering environmental effects. - Conduct analysis for guidance and design of test parameters, geometry, loading and test facility requirements to develop a simplified methodology and produce relevant data for verification of design criteria for the complex geometry IHX. - Develop analysis and design methods for complex 3-dimensional structures like PCHE. - Expand to other components (e.g. pumps, valves, piping) as needed - Determine joining and fabrication methods addressing various product forms (plate, sheet, foil) and physical design - Characterize environmental and aging effects - Develop inspection requirements and processes specifically: pre-service and In-service inspection requirements and the need for defect-tolerance and flaw assessment data, perhaps fracture mechanics as well. - Investigate irradiation-induced helium embrittlement and irradiation creep in control rods Finally, the program must develop a High Temperature Design Methodology to - Establish alternative primary load design methods - Establish simplified methods and criteria - Identify failure modes - Develop inelastic design analysis methods - Perform confirmatory structural analyses and methods - Conduct safety and reliability assessment - Resolve any identified shortcomings and regulatory concerns. *Instrumentation* – Most of the instrumentation needs are associated with the fuel and graphite testing programs. Precise measurements of fluences and temperatures (<2000C) are needed to properly characterize experimental conditions. *Modeling and Simulation* – As with most reactor concepts, the ability to reliably simulate plant phenomena and operating conditions is an essential component of advanced reactor research and development. The purpose of modeling and simulation R&D is to raise the technical readiness of chosen concepts to a level to support commercial deployment. Much of the simulation work needed to support that effort can be effectively provided by concept-optimized system analysis tools. The system analysis codes run much faster than higher-fidelity simulation tools that rely on super-computers, and usually can be executed on desktop servers and work-stations. The systems analyses can capture integral transient effects for the entire plant, as opposed to high resolution analyses focusing only on specific components or local phenomena. Therefore, in a systems analysis, the major physics of the plants and integral effects are captured, albeit with some uncertainties if the geometry is only coarsely modeled or if the correlations have not been validated for the operating conditions. For design optimizations and sensitivity studies, system codes coupled with appropriate subgrid physics, correlations, or higher-fidelity tools can provide the information that is needed for an advanced concept in the conceptual and preliminary design phases. Even though the physics may be common (neutron transport, incompressible fluid flow), the method implementations need to be optimized for specific concepts and operating modes (e.g. fluid flow around pebbles or through prismatic fuel blocks, resonance treatment in doubly heterogeneous fuel, natural circulation after a loss of

Criteria	Small Modular High Temperature Reactor (smHTR)
	forced circulation). International code-to-code benchmark projects have been conducted or are underway to evaluate the ability of system and core analysis codes to model HTR behavior under steady state and transient conditions. Although not code validation exercises, these projects are essential for developing and maintaining expertise in core physics, thermal fluids, and plant dynamics. Modern uncertainty analysis methods are being applied to characterize uncertainties in key core safety parameters relative to input uncertainties and model assumptions, an essential component of any licensing evaluation. Some phenomena that occur during off-normal scenarios are not easily captured with existing tools and thus a fair degree of uncertainty is associated with them. These were largely captured in [20] but are summarized here. - Air ingress (leakage of air into the primary system and core after a break and depressurization). Unmitigated, this may lead to excessive oxidation of the lower graphite core structure. - Core Heat Transfer – detailed knowledge of conjugate heat transfer in the core which may effect peak fuel temperatures and fission product release. - Plenum-to-Plenum heat transfer – After a blower trip, plumes of hot helium driven by natural circulation may rise and impinge on, and overheat, the upper vessel head and control rod drive tubes, - Steam ingress – after a steam generator tube rupture. Water may reacto with graphite and facilitate the release of fission products otherwise captured in the graphite matrix. - Lower Plenum – hot helium exiting the core most turn and flow around graphite support columns. Hot streaking of helium flow may result in local overheating of structures. Depending on the specific core design, these phenomena may or may not significantly impact the structural integrity of the plant. A combination of integral, separate effects, mixed effects, and single physics tests have been planned to generate validation data per NRC Regulatory guide 1.203. Advances in computing power and algorithms for solving complex systems of equations are enabling high fidelity simulations of advanced reactors which promise a greater understanding and predictability of phenomena and scenarios. Much development work is needed, however, before these codes can be used to their full potential and on a routine basis. Furthermore, validating high performance multiphysics, multiscale simulations remains a somewhat unresolved challenge. Development and testing of these tools is best conducted in concert with development, testing, and validation of the existing system and core analysis codes that will continue to be an essential part of R&D activities. For estimating the safety margins in very high temperature reactors (ROT>850), these high performance codes will need to be developed further and validated using the data from a wide variety of experiments. Brayton Cycle Power Conversion – If a Brayton (gas turbine) cycle is preferred for the HTR, the material performance of the intermediate heat exchanger under transient conditions will have to be demonstrated. At design pressures (~7MPA primary/>19 MPa secondary), a sudden loss or pressure or change in temperature of one side may stress the IHX boundary. A focused effort to test materials, designs, and joining technique will need to be completed to raise the TRL of the IHX above its current value of 3. Other balance-of-plant components (recuperators, intercoolers, gas turbines) will also require design and testing as current Brayton cycle components have been designed for natural gas-driven systems running at different pressures, temperatures). The current reference concepts would use a steam PCS and thus these are not immediately required.

Criteria	Small Modular High Temperature Reactor (smHTR)
2) Identification of the general costs for the R&D	Through FY2013, approximately $330M was spent on HTGR/VHTR research and development in the areas listed above. Another $225M is estimated to complete the Fuel and Material (graphite and alloys) qualification and Methods benchmarking and validation for a system with an upper limit of 850C on the reactor outlet temperature and driving a steam cycle. For a Brayton cycle system, another $10M would be needed to balance-of-plant components (IHX, circulators, turbines).
3) Identification of the time frame in which the R&D is needed	With planned funding levels, the qualification of fuels and materials and the experimental validation of codes and methods will be complete by 2021.

4) Relative prioritization of the potential R&D activities	Fuel and graphite qualifications are the highest priority as no TRISO fuel types or existing graphite grades are currently qualified. Alloys 617 and 800H are currently qualified for use in high temperature (<950C) structures (vessels, steam generator tubes, the IHX, and control rod guide tubes) but only up to about 10 years. A power plant could be built under the current code case but many of the expensive components would need replacing well before the planned lifetime of the plant. Qualification of Alloy 800H is needed to extend the ASME code from 40 years at <760C to 60 years, making it suitable for the reactor pressure vessel. Qualification of Alloy 617 is needed to extend the ASME code to 100000 hours at up to <950. This would make it suitable for an IHX that would need to be replaced about every 10 years. Experimental validation of methods and codes is needed to ensure the core conditions and behavior are reasonably well known under all anticipated conditions. With only partial validation, it is likely that the regulator would allow operation only under reduced power density and temperature conditions.

5. References

1. "NGNP Program 2013 Status and Path Forward", INL/EXT-14-31035, March 2014.
2. Safety Analysis Report for the High Temperature Reactor Module Power Plant, April 1987.
3. Lohnert, G. H., "Technical Design Features and Essential Safety-Related Properties of the HTR-Module", *Nuclear Engineering and Design* 121, pp. 259-275, 1990.
4. Strydom, G. and Gougar, H., "Preliminary Reactor Physics Assessment of the HTR Module with 14% enriched USO fuel", *Nuclear Engineering and Design* 256, pp. 304-321, 2013.
5. Personal Correspondence between Chris Ellis (General Atomics) and Javier Ortensi (INL), 18 March, 2014.
6. INL "NGNP System Requirements Manual," INL/EXT-07-12999, Revision 3, September 2009.
7. INL, *Research and Development Technology Development Roadmaps for the Next Generation Nuclear Plant Project*," INL/EXT-11-22512, July 2011
8. INL, *Reactor User Interface Technology Development Roadmaps for a High Temperature Gas-cooled Reactor Outlet Temperature of 750°C*, INL/EXT-10-20460, December 2010.
9. MHTGR PSID – DOE-HTGR-86-024.
10. Modular High Temperature Gas-Cooled Reactor Safety Basis and Approach, INL/EXT-13-30872, January 2014.
11. Durst, P. C., Safeguards-by-Design: Guidance for High Temperature Reactors (HTGRs) with Prismatic Fuel, INL/EXT-12-26130, August 2012.
12. Durst, P. C. Safeguards-by-Design: Guidance for High Temperature Reactors (HTGRs) with Pebble Fuel, INL/EXT-12-26561, August 2012.
13. Nelson, L., "Integration of High Temperature Gas-Cooled Reactors into Industrial Process Applications, INL/EXT-09-16942, Rev. 3, September 2011.
14. Gandrik, A. and Wallace, B., " Assessment of High Temperature Gas-Cooled Reactor (HTGR) Capital and Operating Costs, TEV-1196, Idaho National Laboratory report, January 2012.
15. Next Generation Nuclear Plant Evaluation of Siting an HTGR Co-generation Plant on an Operating Commercial Nuclear Plant Site, INL/EXT-11-23282, Rev. 1, October 2011.
16. High Temperature Gas-Cooled Reactor Projected Markets and Preliminary Economics, INL/EXT-10-19037, August 2011.
17. Next Generation Nuclear Plant Licensing Strategy – A Report to Congress, August 2008.
18. NGNP Licensing Plan, PLN-3202, Rev. 0, 6/26/09.
19. NGNP Project Regulatory Gap Analysis for Modular HTGRs, INL/EXT-11-23216, September 2011.
20. Next Generation Nuclear Plant Phenomena identification and Rankings tables (PIRTs), NUREG/CR-6944, United State Nuclear Regulatory Commission, March 2008.