General Atomics' Prismatic Modular High Temperature Gas Cooled Reactor

Full name	General Atomics' Prismatic Modular High Temperature Gas Cooled Reactor
Acronym	Prismatic HTR
Reactor Type	Gas-cooled Reactor
Coolant	Helium
Moderator	Graphite
Neutron Spectrum	Thermal Neutrons
Thermal Capacity	350 MW(t)
Gross Electrical Capacity	150 MW (e)
Design Status	Conceptual Design
Designer	General Atomics

Contents

1. Introduction

1.1 Development History

Gas-cooled reactors have a long history and are in use today. The earliest development of gas-cooled reactors began in the United Kingdom (UK) and France. The original gas-cooled reactors in the UK were designed with natural uranium fuel cladded in a magnesium alloy, hence the name Magnox. The UK's Magnox reactor was then followed by the Advanced Gas-cooled Reactor (AGR). The AGRs utilized 2 -4% enriched uranium oxide fuel. Compared to the Magnox reactor, the AGR had a higher power density, enriched fuel, higher burnup rate and was 10% more efficient. The second generation of gas cooled reactors, termed high temperature reactors (HTRs), used graphite as a moderator and helium for its coolant. In addition, these early plants also demonstrated coated particle fuel, a fuel form that employs ceramic coatings for containment of fission products at high temperature, a key feature of HTGRs.

Dragon was the first experimental HTR and served as a demonstration plant. It was built in the UK with a power of 20 MWth and went critical in 1964. Following the experience of the Dragon reactor, more HTR designs were introduced in Europe, the United States and Japan. The Peach Bottom Unit 1 was built in the U.S. with a power of 40 MWe and went critical in 1966 followed by the Fort Saint-Vrain (FSV) which had a power of 330 MWe. Both of these reactors had cores consisting of prismatic graphite blocks cooled by helium and served as an experimental platform that provided valuable experience in the operation and design of gas-cooled reactors. Meanwhile, Germany began building its first HTR in 1961, the AVR, which was a 15 MWe reactor using the pebble bed concept. Following the experience of the AVR, Germany then built a second HTR named Thorium High-Temperature Reactor (THTR-300) which went critical in 1983.

More recently, two additional HTGR test reactors have been constructed and are successfully being operated, the 30 MWt High Temperature Test Reactor (HTTR) in Japan and the 10 MWt High Temperature Reactor (HTR-10) in China (Table 1), with design outlet temperatures of 950ºC and 900ºC respectively. Table 1 provides a listing of the seven HTGR plants built and operated to-date.

The HTGR technology that the General Atomics' Prismatic Modular HTGR is based upon has been developed over the past 40 years. The technology basis includes design, licensing, construction and operation of seven HTGR plants. The HTGR concept evolved from early air-cooled and CO2-cooled reactors. Two key characteristics of HTGRs are the use of helium coolant and graphite moderator. The use of helium in lieu of CO_2 as the coolant in combination with a graphite moderator offers enhanced neutronic and thermal efficiencies. The combination of helium cooling and graphite moderator makes possible production of high temperature nuclear heat, and hence the name, High Temperature Gas-cooled Reactor (HTGR).

In the US, General Atomics used the HTGR technology from the early HTGR plants to design several large, 2000 – 3000 MWt, HTGR plants and orders were received from US electric utilities for 10 of these large HTGR plants. The large HTGR plant orders were canceled, along with the cancellation of orders for a large number of other nuclear power plants in the US following the oil embargo in the early 1970s and the ensuing energy conservation measures that dramatically reduced energy demand and the need for new electricity generation capacity.

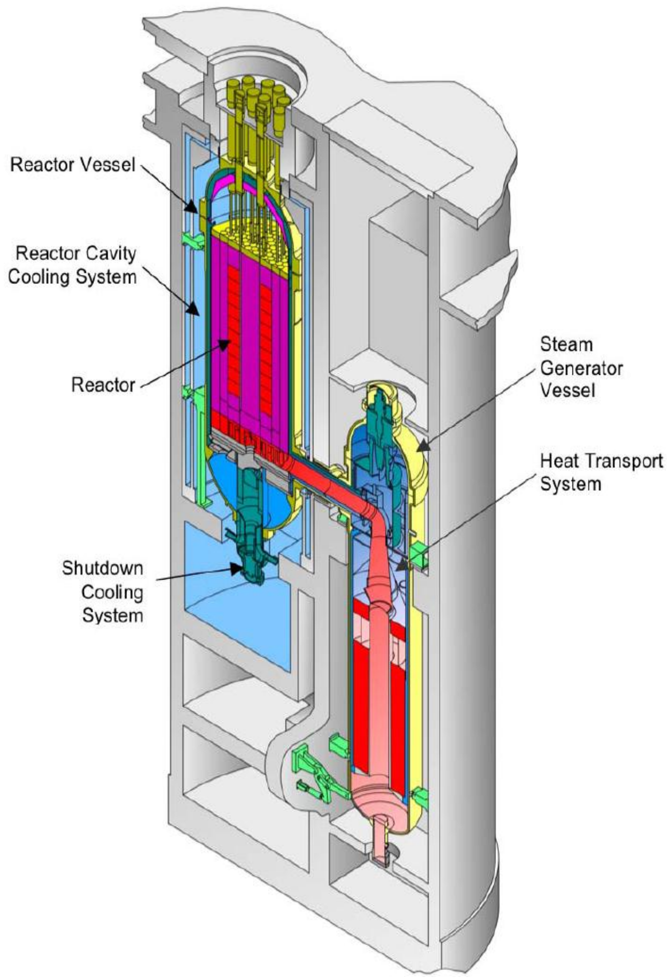

Figure 1: Isometric View of GA Proposed NGNP Reactor Module in Reactor Building Silo

During 2001, the Generation IV International Forum, which includes the U.S. Department of Energy (U.S.DOE) and some other countries which are developing advanced reactors, formulated a roadmap which identified, as one of six advanced nuclear technologies [termed Generation-IV technologies], the Very High –Temperature Reactor (VHTR) "as a system with potential for economical near-term development that is compatible with advanced electricity and hydrogen production, and high-temperature process-heat applications." Subsequently, the U.S.DOE established the Next Generation Nuclear Plant (NGNP) project with goals that include reducing greenhouse gas emissions and enhance energy security. The NGNP project's overall development goals include commercialization of HTGRs for use in the U.S. and abroad. Subsequent HTGR designs were proposed and among them was General Atomics' prismatic modular HTGR (design details are presented in this document).

Table 1: HTGR Plants Constructed and Operated

Feature	**Dragon**	**Peach Bottom**	**AVR**	**Fort St. Vrain**	**THTR**	**HTTR**	**HTR-10**
Location	UK	USA	Germany	USA	Germany	Japan	China
Power (MWt/MWe)	20/ -	115/40	46/15	842/330	750/300	30/ -	10/ -
Fuel Elements	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
He Temp (In/Out°C)	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He Press (Bar)	20	22.5	11	48	40	40	20
Power Density (MW/m^3)	14	8.3	2.3	6.3	6	2.5	2
Fuel Coating	TRISO[(a)]	BISO[(b)]	BISO[(b)]	TRISO[(a)]	BISO[(b)]	TRISO[(a)]	TRISO[(a)]
Fuel Kernel	Carbide	Carbide	Oxide	Carbide	Oxide	Oxide	Oxide
Fuel Enrichment	LEU[(c)]/ HEU[(d)]	HEU[(d)]	HEU[(d)]	HEU[(d)]	HEU[(d)]	LEU[(c)]	LEU[(c)]
Reactor Vessel	Steel	Steel	Steel	PCRV[(e)]	PCRV[(e)]	Steel	Steel
Operation Years	1965-1975	1967-1974	1968-1988	1979-1989	1985-1989	1998 -	1998 -

(a) TRISO refers to a fuel coating system that uses three types of coatings, low density pyrolytic carbon, high density pyrolytic carbon and silicon carbide
(b) BISO refers to a fuel coating system that uses two types of coatings, low density pyrolytic carbon and high density pyrolytic carbon
(c) LEU means low enriched uranium (<20% U^{235})
(d) HEU means high enriched uranium (>20% U^{235})
(e) PCRV means Prestressed Concrete Reactor Vessel

1.2 NGNP Research and Development

For the NGNP, research and development is divided into the following major technical categories: (a) Fuel Development and Qualification, (b) Graphite Qualification, (c) High Temperature Materials Qualification, and (d) Design and Safety Methods Validation. The U.S. DOE's R&D program is structured to address the licensing needs identified in the phenomenon identification ranking table for NGNP as well design data needs.

1.3 Fuel Development and Qualification

The NGNP concept is based on coated particle fuels. Such fuels have been extensively studied over the past four decades. Layers of carbon and silicon carbide surround a uranium kernel to form a tri-isotropic (TRISO) coated fuel particle of approximately 1 millimeter in diameter. The NGNP will contain billions of TRISO coated-particles that are pressed into compacts. The compacts are shaped as either small cylindrical fuel rods for the prismatic design or tennis-ball-sized spheres for the pebble bed design. Rigorous control is applied at every step during the fabrication process to produce high-quality, very low-defect, fuel. Defect levels are typically on the order of one defect per 100,000 particles.

The NGNP Project had to re-establish the capability to fabricate and characterize TRISO-coated particle fuel in the United States. This was a significant effort that required further advancement of the fabrication processes and characterization approaches used in historical TRISO-coated fuel made in the 1970s and 1980s. In 2010, the first fuel irradiation experiment, called AGR-1, completed approximately three years of radiation exposure at the high temperatures expected under normal operation in a HTGR. About 300,000 TRISO fuel particles have been tested to a very high level of uranium fuel utilization, called peak burnup, of 19 percent *without a measureable indication of a single particle failure*. These results are critical in demonstrating the superior performance capability of TRISO fuel and ultimately the HTGR concept. Work has also been underway to establish the capability to perform high temperature testing of this fuel at accident conditions (higher temperatures) to confirm robust safety performance of the fuel under highly unlikely but possible conditions.

1.4 Graphite Qualification

Graphite has been effectively used in the past as structural material for high temperature reactor cores and reflectors. Graphite properties undergo changes during irradiation that are functionally dependent on the grade of graphite and feed stock used to manufacture the graphite. Historical grades of well-characterized graphite and the supply of raw feed stocks used in prior gas reactors no longer exist. The objective of the NGNP Graphite Qualification R&D is to demonstrate that modern grades of nuclear graphite made with current feedstock materials will perform at least as well as historical grades did. The Project is seeking a science-based understanding of the fundamental mechanisms of irradiation behavior of graphite in order to predict how new types and grades will behave in the future. In the longer term, the Project plans to evaluate the influence of fabrication processes and different feedstock materials on graphite behavior so that extensive qualification efforts are not needed when new feed stocks or improved fabrication methods are used to make graphite for future HTGRs after NGNP.

1.5 High Temperature Materials Qualification

The high outlet temperature of an HTGR (750°C or higher depending on the application need) requires the development of high performance metallic alloys to transfer heat from the reactor to the process application. Because these alloys will operate at high temperatures and pressures, stringent requirements are imposed to ensure that metallic structures and equipment will maintain their integrity.

Production grade quantities of candidate high temperature alloys have been procured. State-of-the-art mechanical and environmental testing of the candidate high temperature metallic alloys is underway to understand its mechanical behavior at high temperatures and ensure that they do not degrade after long term exposure to low levels of moisture and other impurities in the helium coolant environment at the high temperatures expected in an HTGR. These tests will allow for examination, characterization, and selection of heat transfer component materials of construction after testing at the temperatures, pressures, and environmental conditions similar to those experienced in the reactor.

2. Description of Nuclear Systems

The US modular HTGR concept design effort began in 1984 when the US Congress challenged the HTGR industry to investigate the potential for using HTGR technology to develop a "simpler, safer" nuclear power plant design[2]. The goal was to develop a passively safe HTGR plant that was also economically competitive. Like most nuclear power plants up to that time, HTGR plants had been designed with reactor core length-to-diameter (L/D) ratios of about 1 for neutron economy. Detailed evaluations showed that low power density HTGR cores with L/Ds of 2 or 3, or more, were effective for rejecting decay heat passively. In the long slender, low power density HTGR cores, it was found that decay heat could be transferred passively by natural means (conduction, convection and thermal radiation) to a steel reactor vessel wall and then thermally radiated (passively) from the vessel wall to surrounding reactor cavity walls for conduction to a naturally circulating cooling system or to ground itself.

The General Atomics' Prismatic Modular HTGR module (Figure) consists primarily of a reactor in one vessel and a steam generator in an adjacent vessel interconnected by a cross vessel. The reactor vessel is similar in size to a large boiling water reactor (BWR) vessel. The single steam generator vessel houses a helically coiled steam generator bundle as well as an electric motor-driven circulator. The pressure-retaining components are constructed of steel and designed using existing technology. The reactor vessel is un-insulated to provide for decay heat removal under accident conditions.

Figure is a flow diagram illustrating how reactor heat is transferred during normal operation by the Heat Transport System within the reactor module primary system. Helium coolant flows from the helium circulator to the reactor vessel in the outer annulus formed between a hot duct concentrically located in the cross vessel, flows up an annulus between the reactor vessel and a core barrel surrounding the core, down through the core, through the center region of the hot duct located in the cross vessel, down through the steam-generator bundle, then up the annular region between the steam-generator bundle and steam-generator vessel back to the inlet of the helium circulator. On the secondary coolant side, feedwater enters the steam-generator vessel at the bottom, flows up through the helical coil tube bundle, exiting as superheated steam.

When the reactor is shut down for maintenance or refueling, decay heat can be removed from the core by the normal Heat Transport System described above, or by the independent Shutdown Cooling System (SCS). The SCS consists of a motor-driven circulator coupled with a compact heat exchanger mounted below the reactor core within the reactor vessel. The shutdown heat exchanger is water-cooled. The SCS is not safety-related.

A third means of removing decay heat is provided by a safety-related Reactor Cavity Cooling System (RCCS). A schematic of the RCCS is shown in Figure 4. The RCCS removes heat radiated from the un-insulated reactor vessel by natural circulation of outside air through enclosed cooling panels along the reactor cavity walls. To minimize the release of argon-41 from the reactor cavity, cooling air does not mix with air inside the cavity. Because air naturally circulates through the RCCS continuously, it is always available to remove decay heat under accident conditions without reliance on active components, power supplies, or operator action. The RCCS provides cooling of the reactor cavity concrete during normal operation.

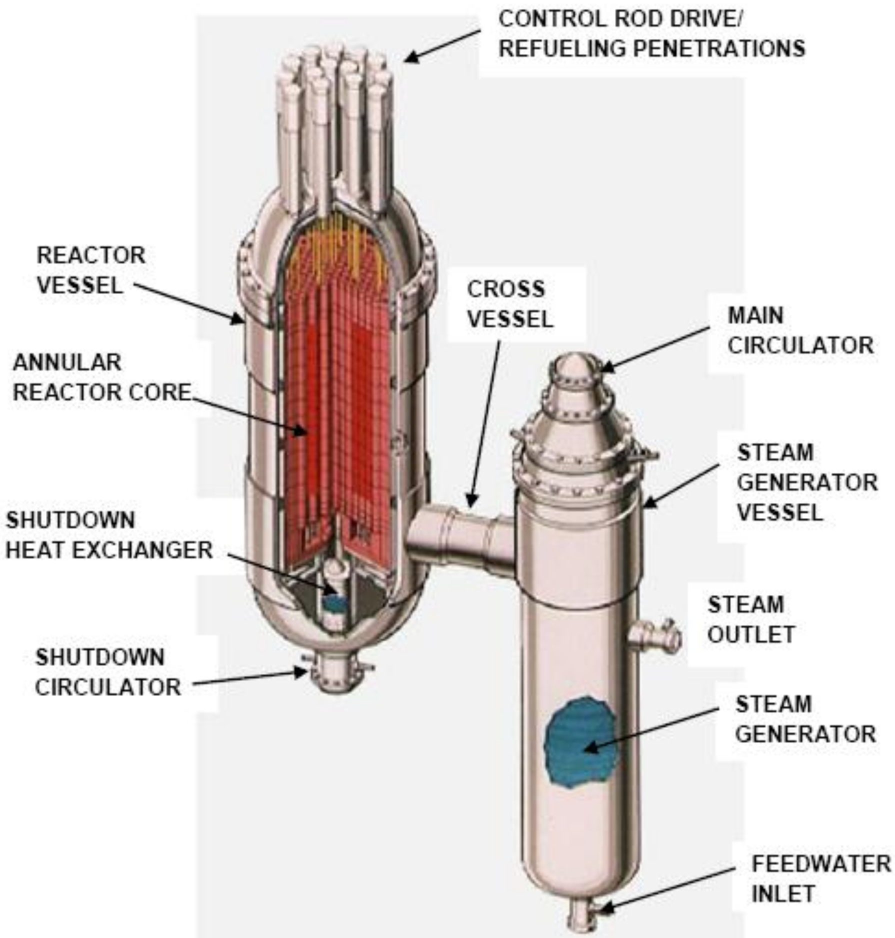

Figure 2: GA Proposed NGNP Reactor Module.

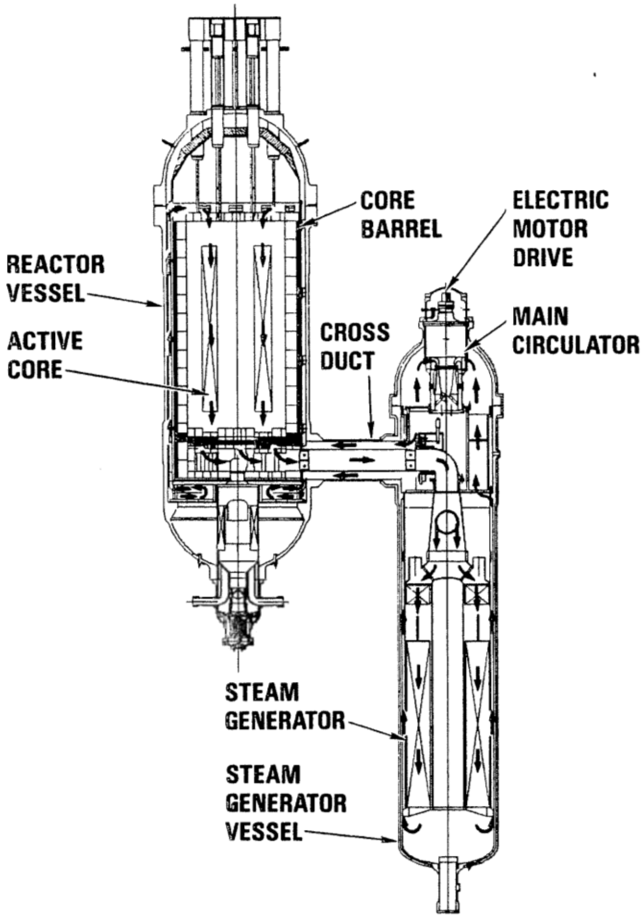

Figure 3: GA proposed NGNP Primary System Flow Diagram.

2.1 Reactor System

The General Atomics' concept for a Prismatic Modular HTGR System, housed in the reactor vessel, consists of a Reactor Core Subsystem, a Neutron Control Subsystem, and a Reactor Internals Subsystem.

2.1.1 **Reactor Core Subsystem**

The Reactor Core Subsystem consists of hexagonal graphite fuel and reflector elements, plenum elements, startup sources, and reactivity control material. The active core consists of hexagonal graphite fuel elements, shown in Figure5, that contain 210 blind holes for fuel compacts and 102 full-length channels for helium coolant flow.

The fuel elements are stacked to form columns 10 fuel elements high. The fuel columns are arranged in an annulus with columns of hexagonal graphite reflector elements in the central and outer regions as shown in Figure . Six central reflector elements and 24 side reflector elements contain channels for control rods.

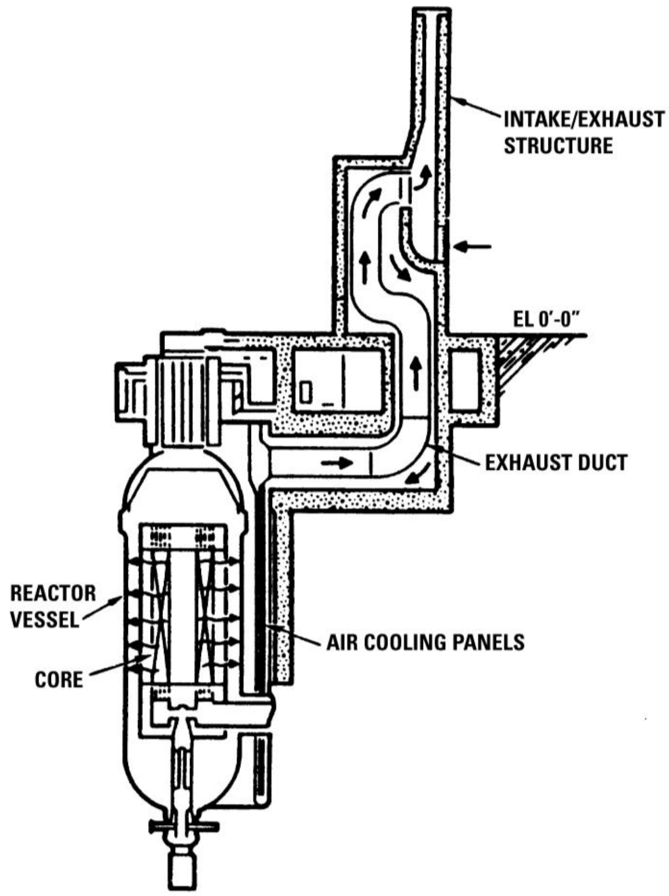

Figure 4: Reactor Cavity Cooling System

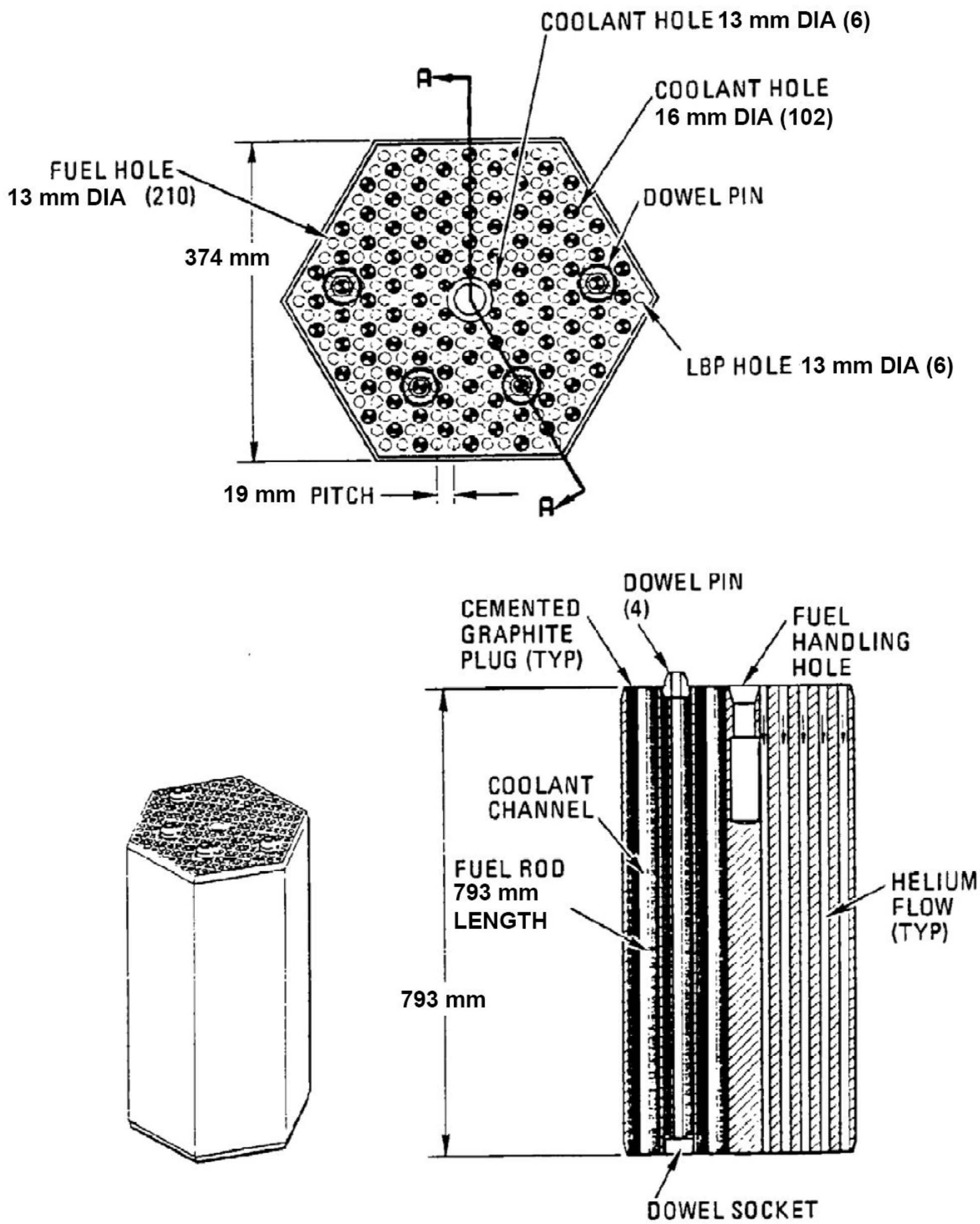

Figure 5: Standard Fuel Element.

The annular core configuration was selected, in combination with the power density (5.9 MW/m3), to achieve maximum power rating and still permit passive core decay heat removal while maintaining the maximum fuel temperature below 1,600°C during a conduction cooldown event caused by either a loss of coolant flow or loss of coolant pressure. The active core outer diameter is 3.5 m, sized to maintain a minimum reflector thickness of 1.0 m within the reactor vessel of 6.55 m inner diameter. These dimensions allow for a lateral restraint structure (core barrel) between the reflector and vessel. An inner core diameter of 1.65 m is selected on the basis of studies on the reactivity worth of control rods with annular cores. To meet the projected reactivity control requirements, using reflector control rods (inner and outer), the annular width of the core can be no greater than ~1 m. The core height is 7.9 m. This core height assures

The fuel compacts consist of refractory coated fuel particles, identified as TRISO coated fuel particles, bonded together with a carbonaceous matrix into cylindrical fuel compacts nominally 12.5 mm OD x 50 mm long. The TRISO fuel particles consist of a spherical kernel of either fissile (LEU) fuel encapsulated in multiple layers of refractory coatings (**Figure**7**)**.

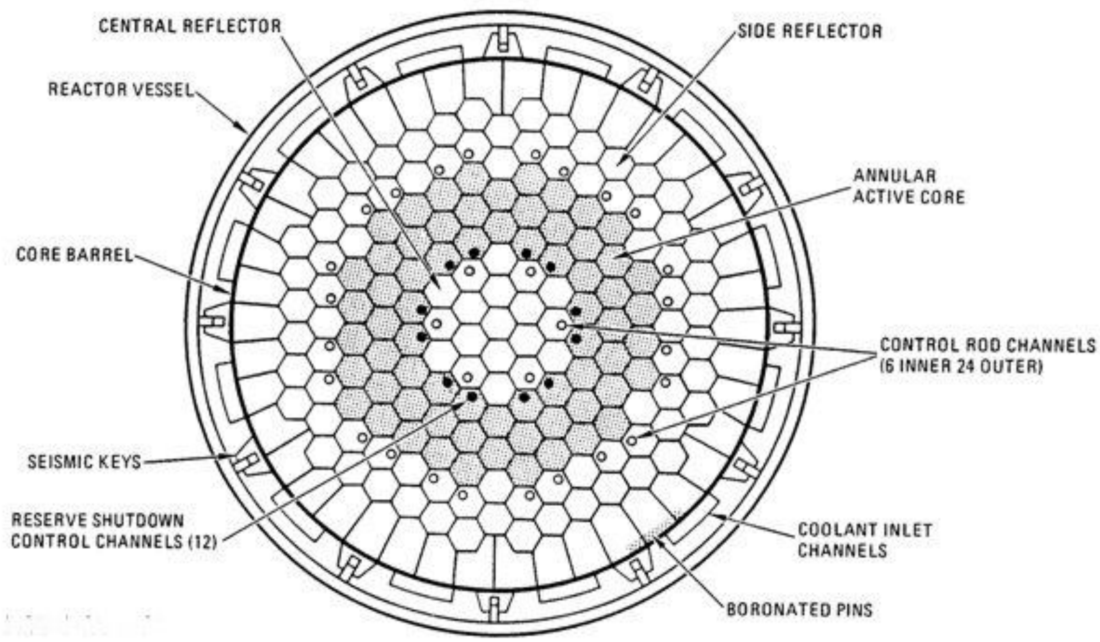

Figure 6: GA proposed NGNP Reactor Core Cross Section

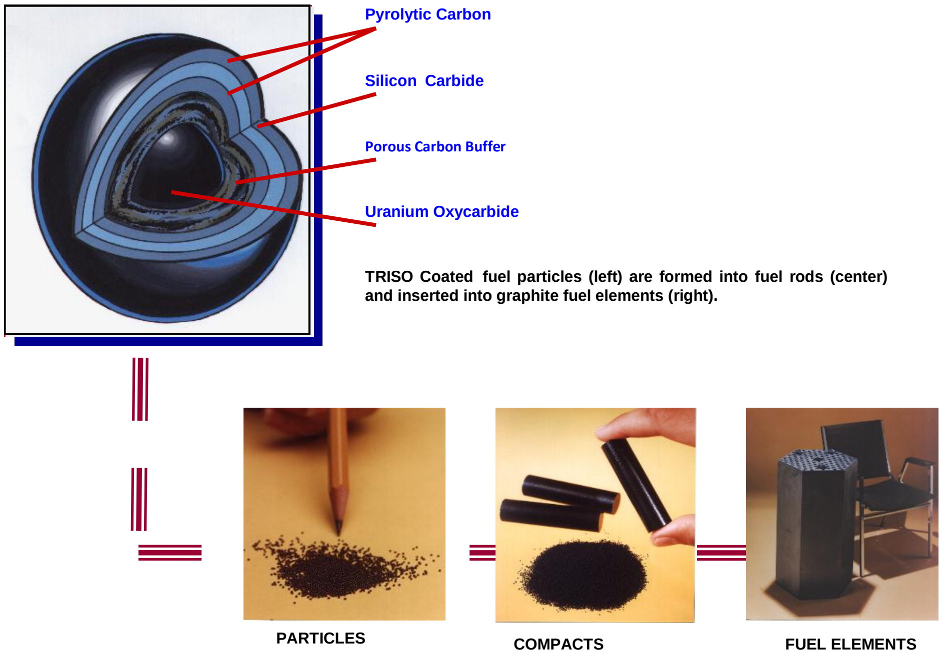

Figure 7: TRISO Coated Particle Fuel.

The multiple coating layers form a miniature, highly corrosion resistant pressure vessel and an essentially impermeable barrier to the release of gaseous and metallic fission products. The overall diameter of standard TRISO-coated particles varies from about 650 microns to about 850 microns.

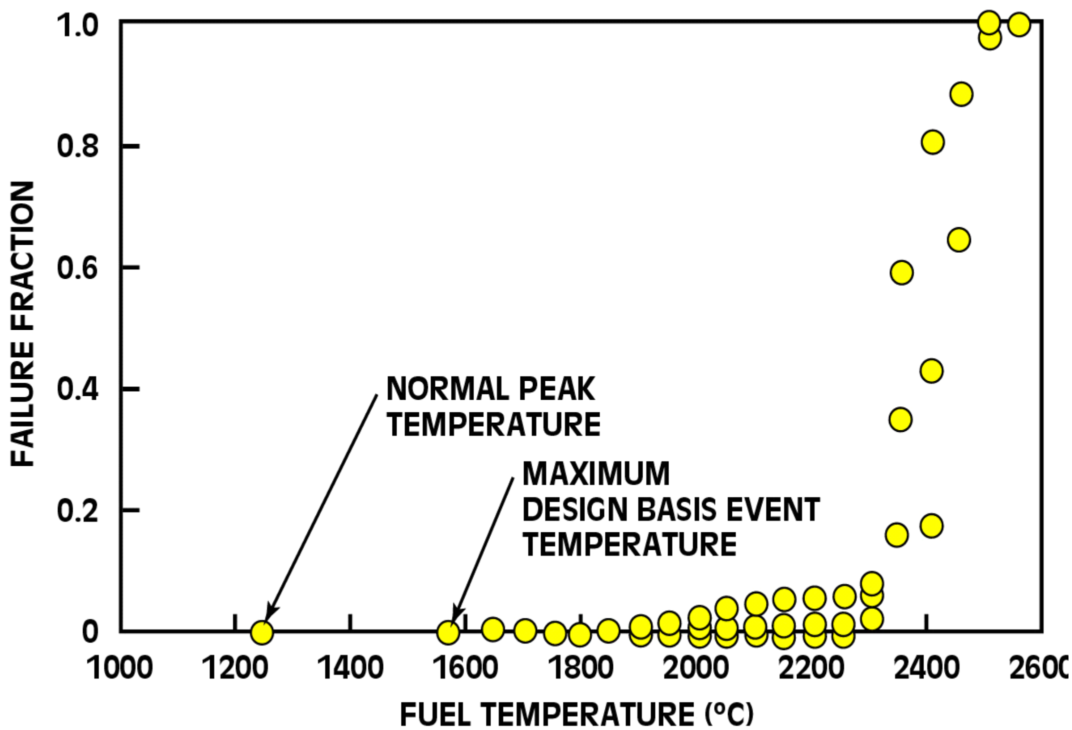

Figure 8: TRISO coated particle fuel temperature capability

The TRISO coatings provide a high temperature, high integrity structure for retention of fission products to very high burnups. The coatings do not start to thermally degrade until temperatures approaching 2000°C are reached (Figure). Normal operating temperatures do not exceed about 1250°C and worst case accident temperatures are maintained below 1600°C. Extensive tests in the United States, Europe, and Japan have demonstrated the performance potential of this fuel.

As listed in Table 2, the fuel for the proposed NGNP reactor is UCO with 15.5% enriched uranium. The UCO fuel was used instead of UO_2 due to the latter's limitations on issues such as burnup and power density. The UCO fuel, for example, prevents free oxygen from being released during fission due to the fuels mixed carbide and oxide components. This allows the suppression of any carbon monoxide to be generated during irradiation.

Table 2: General Atomics' Prismatic Modular HTGR Module Design Parameters

Parameter	Value
Heat transport system	
Module Power	350 MWt
Coolant and pressure at rated power	Helium at 6.39 MPa at circulator discharge
Cold helium temperature	322°C at circulator discharge
Hot helium temperature	750°C at core exit
Core helium flow rate	157.1 kg/s (1,246,000 lb/h)
Core helium pressure drop	34.5 kPa
Feedwater temperature/pressure	193°C/21.0 MPa
Steam temperature/pressure	541°C/17.3 MPa
Vessel material	Low alloy steel, manganese-molybdenum SA533 Grade B, Class 1
Reactor vessel overall height	22 m
Reactor vessel outside diameter	6.8 m
Plant design lifetime	60 yr
Design basis operation	90 percent capacity factor
Number of components per module	
Steam generators	1
Main circulators	1, electric motor-driven
Shutdown cooling heat exchangers	1
Shutdown circulators	1, electric motor-driven
Control rods	30 (6 inner, 24 outer reflector rods)
Reserve shutdown channels	12 (inner row of core fuel elements)
Core and fuel cycle	
Fuel element	Prismatic hex-block, 360 mm across flats x 793 mm height
Active core configuration	66-column annulus, 10-blocks high
Fissile material	Uranium oxycarbide (UCO)
Power density	5.9 W/cm^3
Coolant volume fraction	0.19
Average enrichment	15.5 percent U-235
Power peak/average axial ratio	1.4:1
Initial core loading, kg: U	4,183
Equilibrium reload, kg: U	1,786
Equilibrium burnup, MW-d/MT	117,360

2.1.2 Neutron Control Subsystem

The core reactivity is controlled by a combination of fixed lumped burnable poison, movable poison, and a negative temperature coefficient. The fixed poison is in the form of lumped burnable poison rods, similar in shape to the fuel compacts; the movable poison is in the form of metal-clad control rods. In the event that the control rods become inoperable, a reserve shutdown control capability is provided in the form of boronated pellets that may be released into channels in the active core.

The control rods are fabricated from natural boron in annular graphite compacts with metal cladding for structural support. The rods are located in channels in the outer ring of the central reflector elements and in the inner ring of the side reflector (Figure). The control rods enter the core through top reactor vessel penetrations in which the control rod drives are housed. The twenty-four (24) control rods located in the side reflector are used for normal control and for trip from high power. The six (6) control rods in the central reflector are inserted only for cold shutdown.

2.1.3 Reactor Internals Subsystem

The reactor internals consist of a metallic lateral restraint (core barrel), permanent graphite side reflector, graphite core support structure, metallic core support structure, upper plenum shroud, and the hot duct. The core lateral restraint and the permanent reflector surround the core; the graphite core support structure and metallic core support structure are located below the core; the upper plenum shroud is located above the core; and the hot duct is located within the cross vessel between the reactor vessel and the steam generator vessel.

2.2 Heat Transport System

The Heat Transport System (HTS) consists of the Steam-Generator Subsystem and the Main Circulator Subsystem.

2.2.1 Steam Generator Subsystem

The steam-generator is housed in the steam-generator vessel, with its thermal center located below that of the reactor core as shown in Figure. The steam-generator (see Figure) is a vertically oriented, upflow boiling, cross-counterflow, once-through shell-and-tube heat exchanger. It utilizes a multitude, helically wound tube bundle with a heat duty of 352 MWt.

The steam generator design provides access for tube leak detection and plugging from both ends of each tube. In addition, the design makes possible the removal and replacement of the steam-generator tube bundle through the removable upper vessel head even though the unit is designed with a service life equal to that of the plant.

2.2.2 **Main Circulator Subsystem**

The main circulator is installed in the top of the steam generator vessel (see Figure and Figure). The circulator is a single stage axial flow electric motor driven compressor unit. The electric motor is submerged in the helium coolant and forms an integral part of the compressor rotor. The rotating assembly is fully floating on a set of active magnetic bearings, backed up with a set of antifriction catcher bearings.

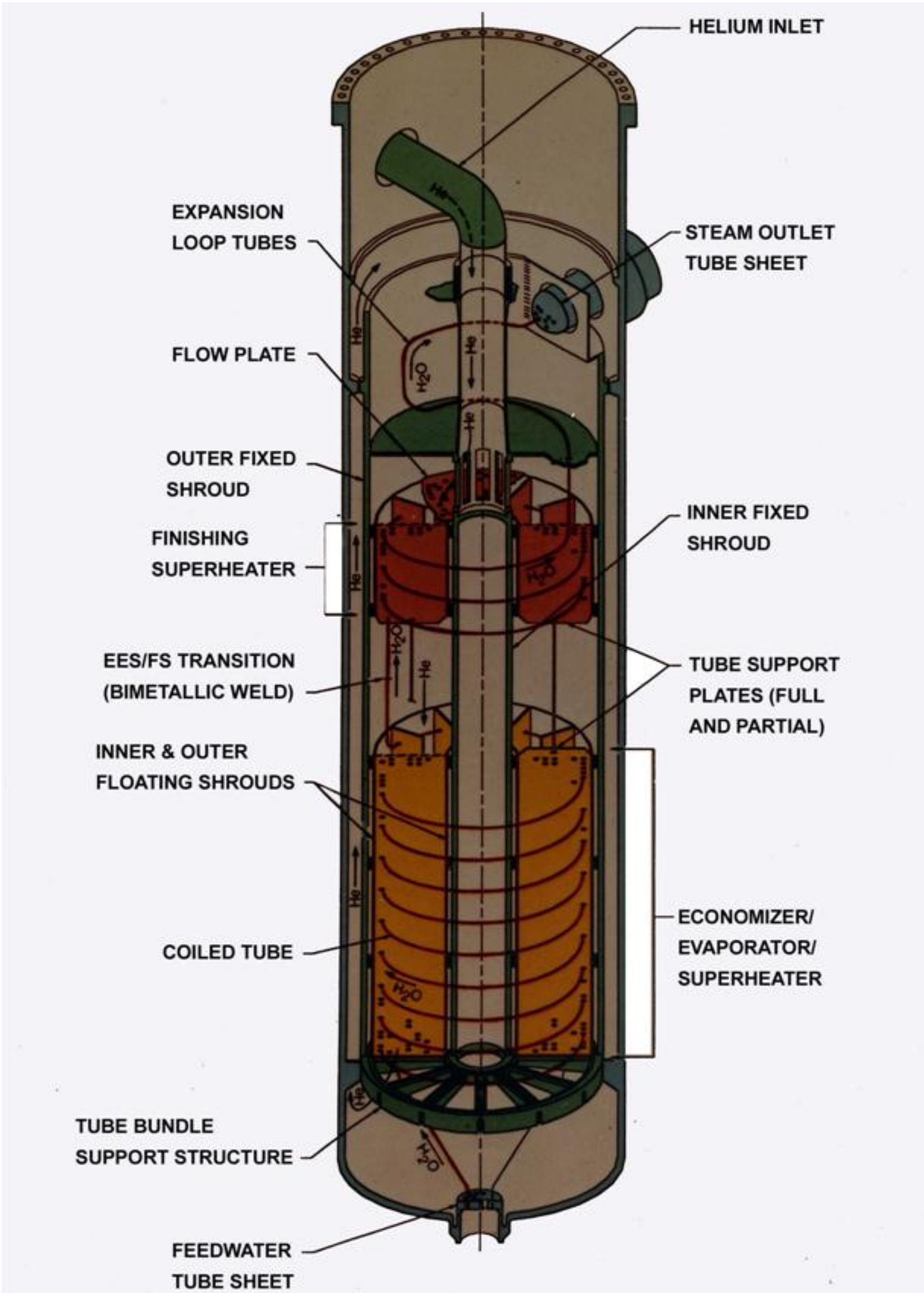

Figure 9: Steam Generator Arrangement.

Primary flow control is provided by use of a variable-speed motor. Downstream of the compressor impeller, a flow-actuated check valve is incorporated in the cylindrical outlet duct. This valve isolates the heat transport system from the reactor vessel when the circulator is not operational (i.e., prevents natural circulation flow of hot gas from entering the steam-generator).

Closure of the valve is affected by counterweights and the pressure forces generated across the core by the operating shutdown cooling circulator.

2.3 Shutdown Cooling System

The Shutdown Cooling System (SCS) provides reactor cooling when the HTS is non-operational. The SCS consists of a shutdown circulator and shutoff valve, a water-cooled shutdown heat exchanger, and shutdown cooling control.

The SCS heat exchanger, arranged in series with the shutdown circulator and shutdown loop shutoff valve assembly, are located at the bottom of the reactor vessel (see Figures 3 and 6). Hot helium from the core outlet plenum flows downward through multiple parallel openings (pipes) in the center of the core support structure and into the water-cooled shutdown heat exchanger. Once cooled, the helium continues downward through the shutdown loop shutoff valve to the shutdown circulator where it is compressed and discharged into the reactor vessel bottom head cavity. The cool helium then flows through the internal passage formed by the core support structure and up channels attached to the core barrel to the core inlet plenum. Heat is rejected from the shutdown cooling water to the plant service water system.

2.4 Reactor Cavity Cooling System

The Reactor Cavity Cooling System (RCCS) is a safety-related system that provides a passive means of removing core residual heat during accident conditions when neither the HTS nor the SCS is available. The RCCS is a completely passive design that has no pumps, circulators, valves, or other active components. The RCCS receives heat transferred from the un-insulated reactor vessel by thermal radiation and natural convection. RCCS components include cooling panels that surround the reactor vessel, inlet/outlet structures that are located above grade and a concentric duct system with the annular, outer flow path acting as the cold leg and the inner flow path acting as the hot leg. Natural convection airflow is established through the RCCS circuit through a balance of buoyancy and gravitational forces.

2.5 Fuel Handling System

The Fuel Handling System consists of a fuel handling machine, fuel transfer casks, an auxiliary transfer cask, a fuel handling equipment positioner, fuel handling equipment support structure, and local used fuel storage and handling facilities. Two or three large, portable, isolation gate valves are also included in the fuel handling system equipment inventory. These valves are placed over the spent fuel storage wells, or the spent fuel sealing and inspection facility whenever elements are moved in or out of these locations. All operations and movements of the machines and the associated fuel and reflector elements are automatically monitored and recorded to maintain full accountability. Each fuel and reflector element is uniquely identified as necessary to support this accountability requirement. A fuel sealing and inspection facility is included in the system to provide for receipt and inspection of new fuel, and for packaging of used fuel that is to be transported for storage or disposition either within the plant area or off-site. Refueling takes place on a specific schedule, and involves the entire fuel element inventory in the reactor core, plus certain replaceable reflector elements as may be required.

- The design proposed by General Atomics (GA) for the Next Generation Nuclear Plant (NGNP) is a High Temperature Gas Reactor (HTGR) that has been evolved from the Modular High Temperature Gas Reactor (MHTGR) [1] concept. .

3 Description of Safety Concept

The General Atomics' Prismatic Modular HTGR safety design objective is achieved through a combination of inherent safety characteristics and design selections that take maximum

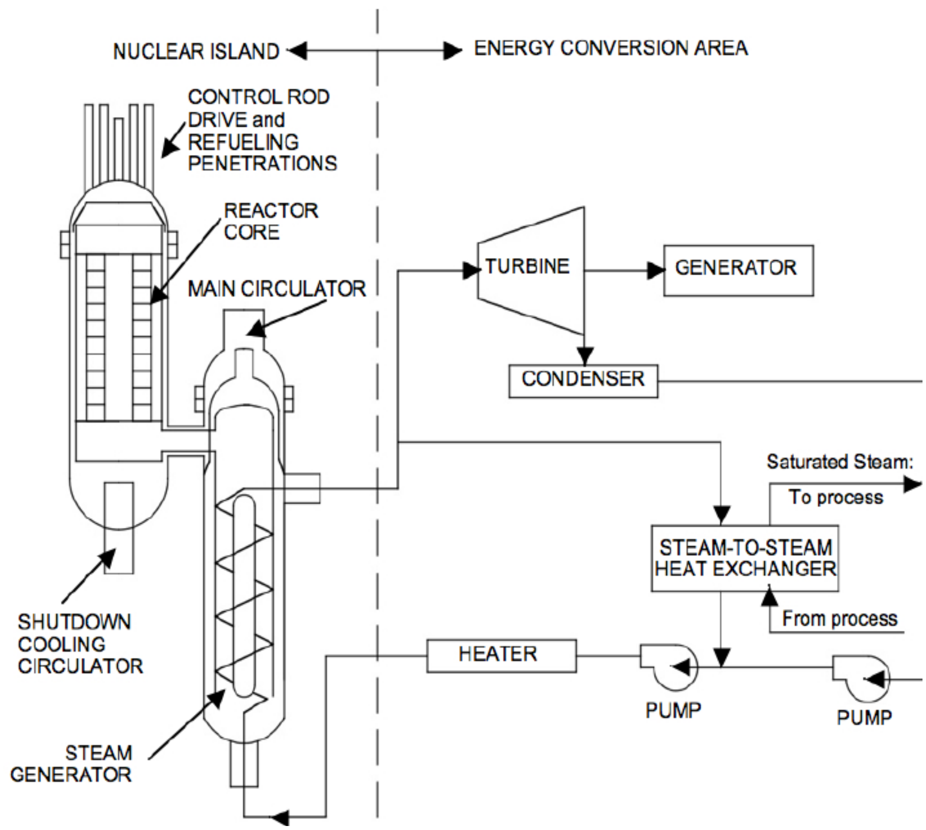

Figure 2: Overall GA Proposed NGNP System.

advantage of the inherent characteristics. The inherent characteristics and design selections include:

- Helium coolant, which is single phase, inert, has only minute reactivity effects and does not become radioactive;
- Graphite core, which provides high heat capacity, slow thermal response, and structural stability to very high temperatures;
- Refractory coated particle fuel, which retains fission products at temperatures much higher than normal operation and postulated accident conditions;
- Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures; and
- An annular, low power density core (5.9 watts/cm^3) in an un-insulated steel reactor vessel surrounded by a natural circulation reactor cavity cooling system (RCCS).

The General Atomics' Prismatic Modular HTGR has two active, diverse active heat removal systems, the HTS and SCS that can be used for the removal of decay heat. In the event that neither of these active systems is available, an independent passive means is provided for the removal of core decay heat. This is the reactor cavity cooling system (RCCS) that surrounds the reactor vessel (Figure).

For passive removal of decay heat, the core power density and the annular core configuration have been designed such that the decay heat can be removed by conduction to the pressure vessel and transferred by radiation from the vessel to the natural circulation RCCS without exceeding the fuel particle temperature limit (Figure 12).

Even if the RCCS is assumed to fail (beyond design basis accident), passive heat conduction from the core, thermal radiation from the vessel, and conduction into the silo walls and surrounding earth is sufficient to maintain peak core temperatures to below the design limit. As a result, radionuclides are retained within the refractory coated fuel particles without the need for

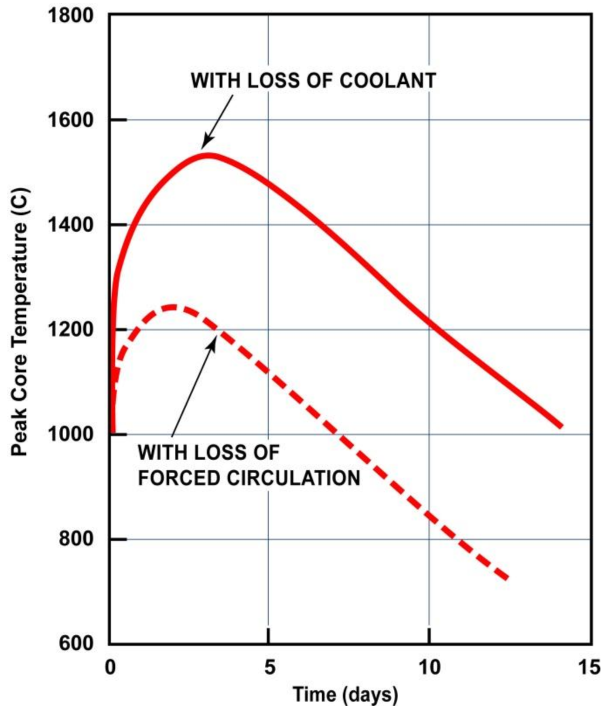

Figure 12: Core heat-up temperatures with passive after heat rejection

active systems or operator action. These safety characteristics and design features result in a reactor that can withstand loss of coolant circulation, or even loss of coolant inventory, and maintain fuel temperatures below damage limits (i.e., the system is meltdown proof).

4 Proliferation Resistance

The General Atomics' Prismatic Modular HTGR has very high proliferation resistance due to low fissile fuel volume fractions and the refractory characteristics of the TRISO fuel particle coating system that forms a containment from which it is difficult to retrieve fissile materials.

Both fresh fuel and used fuel have high resistance to diversion and proliferation. The fresh fuel is very diluted by the quantity of fuel element graphite (low fuel volume fraction). The used fuel has self-protecting high radiation fields and the following proliferation resistance characteristics:

- The quantity of fissile material (plutonium and uranium) per used fuel element is low (50 times more volume of used General Atomics' Prismatic Modular HTGR fuel elements would have to be diverted than used LWR fuel elements to obtain the same quantity of Plutonium-239).

- The used fuel plutonium content, the material of most proliferation concern, is exceedingly low in both quantity per fuel block and quality because of high fuel burnup. The discharged plutonium isotopic mixture is degraded well beyond LWR used fuel making it particularly unattractive for use in weapons.
- No process has yet been developed to separate the residual fissionable material from used TRISO fuel particles. While development of such a process is entirely feasible (and potentially desirable sometime in the future) there is no existing, readily available process technology such as for used LWR fuel. Until such time as when the technology becomes readily available, the lack of the technology provides proliferation resistance.

The TRISO fuel particle coating system, which provides containment of fission products under reactor operating conditions, also provides an excellent barrier for containment of the radionuclides for storage of used fuel.

5 Safety and Security (physical protection)

The General Atomics' Prismatic Modular HTGR is passively safe. The expected high level of safety is achieved through a combination of inherent safety characteristics and design selections consisting of: (1) helium coolant, which is single phase, inert, and has only minute reactivity effects; (2) graphite core, which provides high heat capacity, slow thermal response, and structural stability at very high temperatures; (3) refractory coated particle fuel, which allows extremely high burnup and retains fission products at temperatures higher than normal operation; (4) negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures; and (5) an annular, 350 MWt low power density core in an un-insulated steel reactor vessel surrounded by a reactor cavity cooling system.

The General Atomics' Prismatic Modular HTGR module is located in a below-grade concrete silo (). The below-grade silo arrangement provides high resistance to and protection against natural external threats as well as man-made saboteur threats.

6 Description of turbine-generator Systems

The General Atomics' Prismatic Modular HTGRreactor module is designed to produce steam at 541°C and 17.3 MPa that can be used to generate electricity only or, to co-generate process steam and electricity (the reference NGNP power conversion system). The specific quantities of each of these co-generated products can vary over a large range from electricity only to process steam only. For the purposes of this section, electricity only generation is addressed.

For electricity generation, the turbine plant systems receive steam from the heat transport system and convert steam energy to electrical energy. The main systems needed to support the power generation cycle are:

- Turbine generator and accessories
- Feedwater and condensate
- Main stream and turbine bypass
- Steam and water dump
- Heat rejection

The conceptual design of these systems is provided in the following sections.

6.1 Turbine Generator and Accessories

The turbine generator is nominally rated at 150 MW(e), gross output. The throttle steam conditions are 16.5 MPa (2,400 psig)/538(C (1,000(F). The turbine selected for the MHTGR is of a 3,600 rpm non-reheat design with single-flow HP and IP components and a double-flow LP component. A somewhat improved turbine design is under consideration for the General Atomics' Prismatic Modular HTGR that is based on using a steam-to-steam reheat system such as commonly used in other nuclear power plants.

6.2 Feedwater and Condensate Systems

The condenser associated with the 150 MW(e) turbine generator unit operates at 8.5 kPa (2.5 in. Hg abs). The condenser is a single-shell, double-pass tube configuration and is designed to handle the heat rejection loads associated with the following modes of plant operation:

- Plant startup and shutdown (approximately 25 percent of steam generator thermal rating)
- Normal steady-state plant operation (reactor module power and turbine generator load ranges of 25 to 100 percent)
- Plant transients (heat rejected is limited to approximately one-half of the maximum steam flow rate provided to the turbine under steady-state plant operation)

Four one-third capacity mechanical vacuum pumps are supplied for removing non-condensible gases from the condenser. The vacuum pumps are motor-driven rotary two-stage units. Three 50 percent capacity condensate pumps take suction from the condenser unit. The pumps discharge the condensate through three half-size condensate polisher vessels, and then through four stages of feedwater heaters. The heaters are placed in series and operate under the increased pressure of various stages of extraction steam from the turbine. The second, third, and fourth point heaters are full-sized horizontal units with U-tube arrangement. The first point heater is a direct contact deaerating heater which maintains a constant outlet temperature of 193°C.

The deaerator supplies feedwater to two variable-speed feedwater pumps. In the piping leading to the steam-generator, there are feedwater control valves and two hydraulically operated isolation valves mounted in series. The isolation valves serve to isolate the steam-generator on detection of high moisture concentration in the primary coolant as a result of a steam generator tube leak.

Throughout the feedwater and condensate systems, suitable branch connections are provided for chemical addition, sampling and support of startup, shutdown, and transient modes of operation.

6.3 Main Steam and Turbine Bypass Systems

The main steam and turbine bypass system provides a continuous steam supply to the turbine generator at 16.6 MPa / 538°C . In the event of turbine generator load rejection or during startup and shutdown, the turbine bypass is equipped with pressure reducing and desuperheating stations to automatically condition the main steam before dumping into the condenser.

Two hydraulically operated main steam isolation valves, mounted in series, are provided at the outlet of the steam-generator. These isolation valves serve to isolate the steam-generator on detection of high moisture concentration in the primary coolant, as a result of a steam generator tube leak.

6.4 Steam and Water Dump System

The Steam and Water Dump System serves to eliminate further ingress of water into the primary coolant in the event of a steam-generator tube leak or rupture. It does so by dumping the steam/water inventory of the steam-generator into a dump tank. This action minimizes possible damage to the reactor core by limiting the amount of water made available for fuel hydrolysis and graphite oxidation.

The system bottles up the steam-generator steam/water inventory, including any in-leakage from the primary coolant, for subsequent disposal through the Gaseous and Liquid Radioactive Waste Management systems. This ensures no direct primary coolant release to the environment.

6.5 **Heat Rejection System**

The Heat Rejection System consists principally of a Circulating Water System and a Service

Water System.

<u>Circulating Water System</u>.

The Circulating Water System is an open-loop system flow path associated with the turbine generator and includes the cooling tower and circulating water pumps. Its main function is to remove the heat load rejected to the condensers and convey it to and dissipate it in the cooling tower.

Two 50-percent-capacity circulating water pumps are provided to deliver water to the condenser. The pumps are of the vertical mixed-flow type.

One mechanical draft wet cooling tower is provided. The tower is sized to meet the cooling requirements for the Circulating Water System and the Service Water System.

Chlorination is provided for the circulating water system to protect the main condenser and cooling tower fill from biofouling. There are two evaporators (one operating and one spare) and two chlorinators (one operating and one spare). Chlorine is injected into the pump bays immediately upstream of the circulating water and service water pumps.

6.6 Service Water System.

The Service Water System is an open-loop system that removes waste heat from the Station Chilled Water System, the Reactor Plant Cooling Water System, the Used Fuel Storage Cooling System, and the Shutdown Cooling Water System (SCWS). Two flow paths are provided for heat removal capability during normal and shutdown modes of plant operation.

The normal portion of the system consists of two 100-percent-capacity vertical-type pumps which take suction from the cooling, tower basin. The shutdown portion of the system consists of three 50-percent-capacity vertical-type pumps which supply water to the SCWS heat exchanger during the cooldown and shutdown modes of any reactor module that uses the SCWS.

7 Electrical and I&C Systems

7.1 Plant Electrical Systems

The plant main electrical systems are:

- Offsite and Main Generator Transmission System
- Ac Distribution System
- Uninterruptible Power Supply System

- Essential Uninterruptible Power Supply System
- Dc Power System
- Essential Dc Power System

The conceptual design of these systems is provided in the following sections.

7.1.1 Offsite Power and Main Generator Transmission System

The Offsite Power Transmission System consists of two physically separated and independent transmission circuits interconnecting (1) the utility transmission network and the station switchyard and (2) the unit transformers and the interconnecting circuitry between the switchyard and the high-voltage bushings of the unit and startup auxiliary transformers.

7.1.2 AC Distribution System

The AC Distribution System provides electric power at 13.8 kV and 4.16 kV, three-phase, and 480 V or less, three-phase and single-phase, 60 Hz to electrical switchgear associated with the unit generator to feed the plant's auxiliaries. The system is normally fed from the unit generator through two unit auxiliary transformers. For plant or unit startup, the unit's 13.8 kV and 4.16 kV buses are fed from the grid through two startup auxiliary transformers when the main generator is not running. The system includes transformers, switchgear, unit substations, motor control centers, distribution panels, and backup generators.

7.1.3 Uninterruptible Power Supply (UPS) System

The UPS System provides 120 Vac, single-phase, 60 Hz electric power to the plant's control and instrumentation loads connected to the 120 V UPS bus.

7.1.4 Essential Uninterruptible Power Supply (UPS) System

The Essential UPS System provides reliable and regulated 120 Vac, single-phase, 60 Hz electric power to the plant "safety-related" control and instrumentation loads connected to the four redundant and independent vital buses.

7.1.5 DC Power System

The DC Power System supplies 125 Vdc electric power to the plant's control and instrumentation loads connected to the 125 Vdc bus. The system includes storage batteries, battery chargers, and distribution switchgear.

7.1.6 Essential DC Power System

The Essential DC Power System supplies reliable 125 Vdc electric power to the plant "safety-related" dc loads connected to the four redundant and independent 125 V essential dc buses. The DC distribution system includes storage batteries, battery chargers, distribution switchgear, and distribution panels. Power is distributed from the four dc buses that constitute the plant's four "safety-related" dc control and instrument channels A, B, C, and D.

7.2 Plant Instrumentation and Control (I&C) Systems

In the General Atomics' Prismatic Modular HTGR, the plant I&C system functions are performed by the Plant Protection and Instrumentation System (PPIS) and the Plant Control, Data, and Instrumentation System (PCDIS). The PPIS monitors plant parameters for protection of the public health and safety and to protect plant systems and equipment to protect plant investment. The system consists of the following subsystems:

- Investment Protection System
- Safety Protection System
- Special Nuclear Area Instrumentation Subsystem

The Investment Protection and Safety Protection Subsystems include the sensors and actuation features up to the interface with the actuated equipment. The Special Nuclear Area Instrumentation includes preventive features (interlocks) and various information and display systems. The latter monitors protection system status and plant status under normal and accident conditions Only the Safety Protection Subsystem is needed for the protection of the health and safety of the public.

The scope of the PPIS includes process sensors and electronic logic in a two-out-of-four voting configuration, local equipment, and operator stations that communicate via protection system data highways. The actuation equipment and some system-specific process sensors are within the scope of other systems. The PPIS is functionally independent of the plant control systems, electrically isolated from plant control, and not affected by plant control system failures.

8 Spent Fuel and Waste Management

The General Atomics' Prismatic Modular HTGR has significant spent fuel and waste management (environmental impact) differences relative to light water reactor plants. The thermal discharge (waste heat) from the General Atomics' Prismatic Modular HTGR per kWh is less than a LWR plant because of its higher thermal efficiency. The lesser waste heat makes heat rejection directly to the atmosphere more practical using air cooled heat rejection systems that require no water coolant resources. Because of this capability, the potential for using the

General Atomics' Prismatic Modular HTGR in regions with quite limited water resources is enhanced.

The General Atomics' Prismatic Modular HTGR produces less heavy metal radioactive waste per unit energy produced because of the plant's high thermal efficiency and high fuel burnup. Similarly, the General Atomics' Prismatic Modular HTGR produces less total plutonium and Pu239 (materials of proliferation concern) per unit of energy produced.

The deep-burn capability and high radionuclide containment integrity of TRISO particles offer potential for improvements in nuclear spent fuel management. A high degree of degradation of plutonium and other long-life fissile actinides can be achieved. Nuclear design analyses of the deep-burn capability indicate that, in one pass through the reactor, virtually complete destruction can be accomplished of weapons-usable materials (Plutonium-239), and up to 90% of all transuranic waste, including near total destruction of Neptunium-237 (the most mobile actinide in a repository environment) and its precursor, Americium-241. The resultant particles contain significantly reduced quantities of long-life radionuclides and very degraded fissile materials that can then be placed in a geologic repository with high assurance the residual products have insufficient interest for intentional retrieval and will not migrate into the biosphere by natural processes before decay renders them benign.

Spent fuel is expected to be disposed as coated particle fuel both for long-term interim storage and permanent geologic disposal. The refractory coatings are predicted to retain their integrity in a repository environment for hundreds of thousands of years. As such, they provide defense-in-depth to ensure that the spent fuel radionuclides are contained for geologic time frames.

9 Plant Layout

The General Atomics' Prismatic Modular HTGR Plant layout (Figure 13) is separated into two major areas: a Nuclear Island (NI), containing four reactor modules; and an Energy Conversion Area (ECA) containing the power conversion systems (turbine generators, process steam production re-boilers and support systems). All safety-related structures, systems, and components are contained within the NI. Separation of the plant into distinct physical areas permits procurement and construction of the ECA to conventional standards.

All four modules and the balance-of-plant systems are controlled from a single non-safety-related control room. Systems containing radionuclides and safety-related systems are minimized and contained within the NI which is separated physically and functionally from the remainder of the facility.

Within the NI, each reactor module is housed in adjacent, but separate, reinforced-concrete Reactor Building located below grade (as shown in Figure 3). This below-grade location provides significant design benefits, such as reducing the seismic amplifications typical of above-grade structures. Safety systems for each module are independent of other modules and are located within the individual concrete structures. These dedicated safety systems include safety-related protection and decay heat removal systems. Support functions that are not safety-related but that are located within the NI include cooling water, ventilation, helium processing, radioactive waste processing, and fuel handling.

Other buildings and structures contained in the reference plant layout include:

- Turbine building in the ECA.
- An operations center located adjacent to the nuclear island personnel services building; a double plant security boundary uses part of the operations center as part of the security boundary.
- A standby power supply facility, fire protection facility, makeup water treatment and auxiliary boiler/water treatment facility located in the ECA.
- Separate NI and ECA warehouses.

9.1 Reactor Building and Vented Low Pressure Containment

The General Atomics' Prismatic Modular HTGR Building is a multi-celled, embedded structure constructed of cast-in-place reinforced concrete. The degree of embedment was selected to serve a number of objectives, including reduced cost and complexity of construction, ease of operation, minimization of shielding, and good seismic performance. The operating floor is set at site grade, with a common maintenance enclosure covering the operating area traversed by shared refueling equipment. Two floors below grade with a rectangular footprint are used to house mechanical, electrical, and instrumentation systems dedicated to each reactor. A number of additional mechanical and electrical systems which do not require radiation shielding or protection from external hazards are designed to be delivered to the site as prefabricated modules and located at grade outside the maintenance enclosure. The Reactor Building from ~9m below grade is configured as a cylinder to enable it to resist soil and groundwater pressure. Access to and from the cylindrical portion of the building for piping, electrical services, personnel, and the concentric RCCS ducting is made from the rectangular portion of the building between elevations -9m and grade. Access for refueling and for major maintenance activities is from the operating floor. There are two extensions of the reinforced concrete Reactor Building above grade. On the east side (see Figure 13) of the Reactor Building, the reinforced concrete portion of the building extends to elevation +29m to serve as the Reactor Cavity Cooling System elevated inlet-outlet structure. On the west side, the reinforced concrete extension above grade contains cells that provide a confinement vent path. These cells and most of the cells in the cylindrical portion of the building have been designed to form a closed, interconnected space which is normally isolated from the environment. This space is designed to have a leak rate of no greater than 1 volume per day at an internal pressurization of 1 psid, and to vent whenever the internal pressure exceeds 1 psid.

In the event of a large primary coolant leak within the closed portion of the Reactor Building that results in an internal pressure greater than 1 psid, the confinement gases are able to flow out vent path relief valves (or dampers) to the atmosphere. The vent dampers are maintained in a closed position by gravity, and the weight of the damper determines the relief setpoint pressure, which is the internal pressure needed to open the damper. The relief setpoint pressure affects both the nominal reactor building leak rate and the building pressure transients following a large primary coolant leak. The building relief setpoint pressure and vent opening area can both be adjusted if needed to obtain satisfactory performance during a pressure transient.

The gases vented to the atmosphere via the Reactor Building vent are considered to be ground level releases. Radionuclides released are assumed to travel a minimum of 425 m to the site exclusion area boundary (EAB) before they result in the exposure of a member of the public.

10 Plant Performance

10.1 Plant Operation

The General Atomics' Prismatic Modular HTGR plant is designed for the integrated control and operation of four reactor modules by the operators from a central control room. The plant control and protection systems are designed with a high level of automation to aid the operator in controlling the plant. The operator will supervise all automatic control actions and will have the means to control actions during normal, off-normal, and emergency events. Since the General Atomics' Prismatic Modular HTGR uses inherent characteristics in a configuration which provides assured, simple, and passive safety features: (1) operator actions or active engineered safeguards equipment are not required to protect public health and safety, and (2) operator errors will not negate these safety features and capabilities.

10.2 Reliability

The reliability of the General Atomics' Prismatic Modular HTGR has been evaluated by assessments made of the MHTGR forced outage rate. Forced outages are due to an unplanned component failure or other condition which requires that a module or the plant be removed from service immediately, or up to and including the next period of lower capacity needs (e.g., a weekend).

The MHTGR plant forced outage assessment was based on individual estimates of forced outage for each of the plant's systems. The major contributors to forced outage are the turbine generator, feedwater and condensate, heat transport, and vessel systems. These systems were assessed to contribute a total outage rate of approximately 5 percent/year.

10.3 Availability Targets

Total plant availability is a combination of scheduled outages and forced outages. The specific availability target for a 4-module General Atomics' Prismatic Modular HTGR is a design capacity factor of 90%. The primary driver for scheduled outage is the time required for the plant to be shut down for refueling. To achieve the design capacity factor with a forced outage rate of ~5%, the following plant refueling targets have been set:

- Fuel cycle length of 18 months
- Single module refueling outage of ~15 days

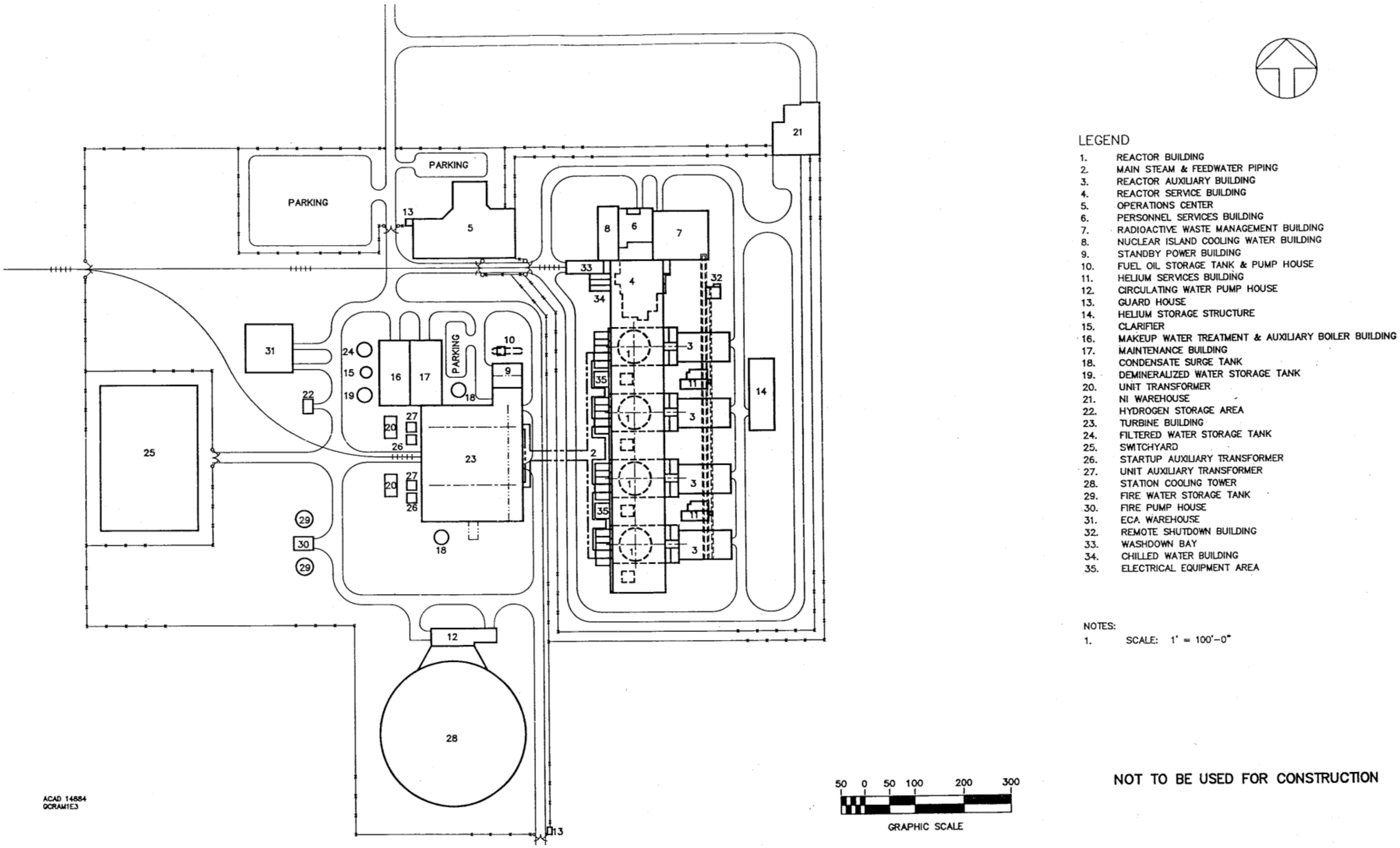

Figure 13: Reference 4-Module Plant Layout

Evaluations of refueling outages indicate the design capacity factor can be achieved with these plant refueling targets.

10.4 Provision for Reduced Capital and Construction Costs

The General Atomics' Prismatic Modular HTGR has been designed to reduce the potential for escalation of capital and construction costs through the use of passive safety systems, standardization, modularization, factory construction and licensing approach[1].

The potential for construction delays resulting from objections of the public in hearings after the start of construction has been reduced in the US by the adoption of a "one-step" regulatory approach that involves obtaining a combined construction and operating license (COL). The COL approach requires resolution of all safety issues, including public hearings, before the start of construction. Once all safety issues have been resolved and the COL is issued, there is reduced potential for construction delays because no additional public hearings required.

The bases for reduced capital and construction cost due to passive safety, standardization, modularization, and factory construction are as follows:

- The use of a passive safety concept minimizes the quantities of safety related equipment required which reduces both construction cost and schedule.
- Standardization enables the realization of the economies of serial production.
- Modularization reduces the amount of field construction labor required and enhances the potential for factory fabrication.
- Factory fabrication provides for more optimum working conditions, more skilled craftsmen, and better quality control, all of which results in more efficient production that reduces cost and schedule.

10.5 Construction Schedule

The provisions identified above for reduced capital and construction costs provide for a more optimum construction schedule.

10.6 Provision for Low Fuel Reload Costs

Provision is made for low fuel costs primarily by high thermodynamic efficiency and high fuel burnup. The TRISO fuel has the capability for very high burnup, up to ~120, 000 MWd/tonneU. On balance, the General Atomics' Prismatic Modular HTGR fuel cycle cost is projected to be

[1] As mandated by The EPAct, the NGNP Project has developed and begun to implement a licensing strategy for the Next Generation Nuclear Plant. The centerpiece of the NGNP licensing strategy is the development of a combined license application (COLA) pursuant to 10 CFR 52.

higher than that for a comparable LWR but the total production cost (fuel plus O&M) is comparable due to lesser and simpler equipment required by the NGNP passive safety systems.

11 Development Status of Technologies Relevant to the NPP

11.1 Technology Development for the General Atomics' Prismatic Modular HTGR

The R&D requirements for the GA proposed NGNP are incorporated in the program described in Reference [3].

12 Deployment Status and planned schedule

12.1 Development Status

The path forward for deployment of the General Atomics' Prismatic Modular HTGR technology is necessarily a demonstration project because of the relatively large cost, schedule and performance risks associated with a first-of-a- kind (FOAK) plant. The plant characteristics that particularly need demonstration include items such as the safety design approach, and fuel operating conditions (burnup, fluence, temperature). The risk uncertainties associated with these FOAK characteristics makes attempts to obtain project financing by private industry extremely difficult.

The potential benefits of the General Atomics' Prismatic Modular HTGR for the generation of electricity (passive safety, economics, reduced environmental impact, and high proliferation resistance) coupled with the capability to produce high temperature process heat are considered by GA to provide significant incentives for a government-sponsored demonstration program such as the proposed NGNP demonstration project. The NGNP demonstration project is a first step toward the development of the next generation of nuclear power resulting from the extensive, multi-national evaluation of advanced nuclear power generation options conducted in the Gen-IV program. The demonstration project objectives are currently being planned to:

- Demonstrate a full-scale prototype gas-cooled reactor to produce high temperature steam for efficient electricity production and high temperature process steam for process industries.
- Demonstrate by test the exceptional safety capabilities of advanced gas cooled reactors
- Obtain an NRC License, under 10CFR Part 52, to construct and operate

The General Atomics' Prismatic Modular HTGR is well suited for these objectives. Or, conversely, these objectives well suit GA proposed NGNP demonstration requirements.

Demonstrating that the fuel satisfies performance requirements will necessarily have to be an integral part of the demonstration project. Test fuel will have to be fabricated, irradiated and accident tested to provide the performance data needed for licensing the plant by the NRC.

12.2 Companies/Institutions Involved in R&D and Design

The industrial companies involved in the design of the General Atomics' Prismatic Modular HTGR are:

- General Atomics (Team Lead)
- URS Washington Division
- Electric Boat

The Idaho National Laboratory is the lead R&D institution.

13 References

1. Gas Cooled Reactor Design and Safety, Technical Reports Series No. 312, Chapter 5.2, pp 91-116, IAEA, Vienna, 1990, ISSN 0074-1914.

2. Letter to Harold Agnew, President GA Technologies Inc, from U.S. House of Representatives, Committee on Science and Technology, November 16, 1983.

3. Next Generation Nuclear Plant Project Technology Development Roadmaps: The Technical Path Forward for 750 - 800°C Reactor Outlet Temperature, INL/EEXT-09-16598, Rev 0, August 2009.