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MHTGR — NEW PRODUCTION REACTOR SUMMARY OF EXPERIENCE BASE

**by
PROGRAM STAFF**

MARCH 1988

GENERAL ATOMICS

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ABSTRACT

Worldwide interest in the Modular High-Temperature Gas-Cooled Reactor (MHTGR) stems from the capability of the system to retain the advanced fuel and thermal performance while providing unparalleled levels of safety. The small power level of the MHTGR and its passive systems give it a margin of safety not attained by other concepts being developed for power generation.

The demonstrated capability of the MHTGR to produce tritium, while retaining advanced performance and levels of safety, has made it a prime candidate for the New Production Reactor (NPR). The extensive experience base for the MHTGR is presented in this document. While the concept provides advanced performance and safety features, the basic design concept is evolutionary. In its evolution, full advantage was taken of three decades of development, and the experience derived directly from design, licensing, and operating experience provided by helium-coated HTGRs, including Peach Bottom 1, Fort St. Vrain, AVR,* and THTR.** In addition, use is made of the applicable experience provided by the more than 50 gas-cooled reactor plants operating throughout the world, as well as applicable light-water reactor technology.

This report covers the experience base for the key nuclear system, components, and processes related to the MHTGR-NPR. The specific technologies related to tritium production in a MHTGR-NPR, namely, the tritium target and tritium recovery process are described in a separate document (GA-A18963).

*AVR = Arbeitsgemeinschaft Versuch Reaktor.

**THTR = Thorium Hochtemperature Reaktor.

LIST OF ABBREVIATIONS

AGR	advanced gas-cooled reactor
AVR	Arbeitsgemeinschaft Versuch Reaktor (small German pebble bed reactor prototype)
DOE	Department of Energy
FRG	Federal Republic of Germany
FSV	Fort St. Vrain
FTE	fuel test element
GA	General Atomics (formerly GA Technologies Inc.)
HEU	high-enriched uranium
HEU/Th	high-enriched uranium/thorium
HTGR	high-temperature gas-cooled reactor
LWR	light-water reactor
MHTGR	modular high-temperature gas-cooled reactor
NPR	new production reactor
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
PCRV	prestressed concrete reactor vessel
PWR	pressurized water reactor
RCCS	reactor cavity cooling system

THTR	Thorium Hochtemperature Reaktor [300-MW(e) thorium high-temperature reactor]
TRISO	fuel particle coating with three materials: low-density pyrolytic carbon buffer next to kernel with silicon carbide layer sandwiched between two high-density pyrolytic layers
UCO	uranium oxycarbide
UC ₂	uranium dicarbide
UK	United Kingdom
U.S.	United States

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1. OVERVIEW

1.1. MHTGR-NPR PLANT CONCEPT DESCRIPTION

The MHTGR-NPR plant consists of eight 350-MW(t) reactor modules installed as two 4-module production blocks with the associated auxiliary systems and services necessary to manufacture, handle, irradiate, and process driver fuel and tritium producing targets. Details of the facility are given in Figs. 1-1 and 1-2. Each reactor module is coupled to a turbine generator for electric power production as a by-product. A 100% turbine bypass system ensures that turbine unavailability does not prevent the production plant operation. A simplified system heat balance diagram is shown in Fig. 1-3.

Each 350-MW(t) module consists of an annular reactor core of nuclear fuel and targets in hexagonal graphite blocks, a gas circulator, and a steam generator. They are enclosed in steel pressure vessels, which are located underground in a concrete silo which also serves as part of the containment. The gas circulator transports helium coolant through the core, where the helium absorbs heat generated by nuclear fission in the fuel blocks. Helium is an inert gas that cannot become radioactive or react with the fuel or reactor internals. The hot gas passes through the cross duct vessel to the steam generator. The reactor arrangement is shown in Fig. 1-4. Major features of the plant are given in Table 1-1.

The fuel and target materials are encapsulated in refractory coatings. These coatings, which form the primary fission product and tritium retention boundary, have demonstrated an extraordinarily high level of integrity during both normal and accident condition. The 350-MW(t) annular reactor core geometry, low-power density and below

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Fig. 1-1. MHTGR-NPR plant elevation

Fig. 1-2. MHTGR-NPR production facility plot plan

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Fig. 1-3. MPR system heat balance diagram

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Fig. 1-4. MHTGR-NPR arrangement

TABLE 1-1
MAJOR FEATURES OF MHTGR-NPR PLANT CONCEPT

Item	Parameter
Nuclear system	
Number of standard modules	8
Thermal power per module	350 MW(t)
Plant construction	2 x 4 modules, sequential
Vessel(s) arrangement	Side-by-side
Installation	Below grade silo
Reactor core	
Core arrangement	Annular geometry
Fuel element geometry	Prismatic blocks
Power density	5.91 W/cm ³
Fuel	High-enriched uranium
Fuel form	TRISO-coated fuel particles
Helium inlet temperature	497°F
Helium outlet temperature	1268°F
System pressure	925 psia
Production system	
Target	Lithium aluminate
Target form	TRISO-coated target particles
Tritium recovery	Thermal extraction process
Target compact removal frequency	6 months
Power conversion system	
Prime mover	8 steam turbines
Power cycle	Nonreheat
Electrical output per module	135 MW(e)
Heat rejection	Wet or dry cooling tower
Plant status	
Design	Preliminary
Technology	Demonstrated

grade installation, assure sufficient decay heat removal by passive means (i.e., radiation, conduction, and convection so that integrity of the coated fuel particles is not threatened).

The safety characteristics of the MHTGR are facilitated by the ability to control heat generation and remove core heat by passive means. The elimination of the reliance of dedicated and auxiliary engineering systems and the need for operator actions or powered systems to retain radionuclides have been a major goal of the MHTGR program.

In addition to the main cooling loops, the MHTGR-NPR is designed with a shutdown cooling system to provide decay heat removal during refueling, retargeting, and maintenance to ensure high availability. In the event the primary system and the shutdown system are not available, decay heat is safely dissipated through the uninsulated reactor vessel wall to the reactor cavity cooling system (RCCS). The RCCS consists of water-cooled panels surrounding the reactor vessel which are incorporated into two independent cooling trains. In both active and passive cooling modes, the circulation of water can dissipate enough heat to keep temperatures well within limits to protect equipment and prevent the release of radioactivity from the fuel particles.

1.2. SUMMARY OF TECHNOLOGY BASE(S)

The MHTGR-NPR is based on proven technology and operating plant experience. More than 25 years of engineering effort and over a billion dollars have gone into the design, testing, construction, and operation of systems and components directly applicable to the MHTGR-NPR. As a result, the experience base for its implementation is extensive and reliable.

The MHTGR-NPR draws on proven gas-cooled reactor technology demonstrated by more than 50 carbon-dioxide-cooled (CO₂) and helium-cooled (He) reactors built and operated in Western Europe and the United States since 1956. Five high-temperature gas-cooled helium reactors [the Dragon experiment in England, the 15-MW AVR experimental reactor in Germany, the 40-MW Peach Bottom plant in Pennsylvania, the 300-MW(e) THTR plant in Germany, and the 330-MW(e) Fort St. Vrain (FSV) plant in Colorado] provide significant needed operating plant experience and know-how. A comparison of the major features of the MHTGR-NPR with the two United States built HTGRs and German HTGR plants is given in Table 1-2.

The sources for the extensive technology base for the MHTGR are portrayed in Fig. 1-5. In the large HTGR database, considerable experience was gained from the design, development, testing programs, and licensing activities on major commercial HTGR power projects initiated in the early 1970s, including the 2 x 770-MW(e) Delmarva (Summit) and 2 x 1170-MW(e) Philadelphia Electric (Fulton) reactor projects.

Since the mid-1970s, the database has been further expanded, and continues to be expanded by Department of Energy (DOE) and privately funded programs focused in three main areas: (1) the development of the design and licensing basis of large integrated HTGR plants, and since 1983 the MHTGR; (2) continuing technology interaction with foreign gas-cooled reactor programs, through international agreements at both industrial and government levels; and (3) technology-related research and

TABLE 1-2
MAJOR FEATURES OF HIGH-TEMPERATURE GAS-COOLED REACTORS

	Plant				
	MHTGR-NPR	Peach Bottom 1	FSV	AVR	THTR
Years of service	--	1967 to 1974	1976 to present	1967 to present	1985 to present
Country	U.S.	U.S.	U.S.	FRG	FRG
Plant service	Nuclear material prod/ power	Utility power generation	Utility power generation	Experimental	Power generation
Plant rating, MW(t)	350/module	115	842	46	750
Power output, MW(e)	135/module	40	330	15	300
Fuel element type	Prism	Cylinder	Prism	Pebble	Pebble
Fuel cycle	HEU	HEU/Th	HE/Th	HEU/Th	HEU/Th
Core power density, W/cc	5.9	8.3	6.3	2.5	6.0
Vessel(s)	Steel	Steel	PCRV	Steel	PCRV
Reactor building	Below grade silo containment	Above grade containment	Containment	Containment	Containment
Main circulators	1/module	2 (1/loop)	4 (2/loop)	2	6
Power [unit, kW(e)]	3169	1417	3950	50	2300
Drive type	Electric motor	Electric motor	Steam turbine	Electric motor	Electric motor
Bearing type	Oil (magnetic alternative)	Oil	Water	Oil	Oil
Number of steam generators	1/module	2 (1/loop)	12 (6/loop)	1	6
Steam generator type	Helical nonreheat	U-tube	Helical reheat	Serpentine nonreheat	Helical reheat
Residual heat removal					
Primary	Main loop	2 separate main loops	2 separate main loops	Main loop	2 separate main loops
Secondary	Shutdown cooling system	Main loop with alter- native motive force	Main loop with alternative motive force	Vessel cooling	Main loop with alternative motive force
Tertiary	Reactor cavity cooling (active and passive)	Reactor vessel cooling panels	PCRV liner cooling	None	None
Ultimate	Conduction to earth	Environment	Environment	Environment	Environment

Fig. 1-5. Technology bases for MHTGR plant

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development programs involving the expenditure of several hundred million dollars. The result of these activities has positioned the MHTGR where it can be deployed in the near term without a major need for fundamental research. Design verification and buildup of the database is planned for the MHTGR-NPR to provide assurance that the specific requirements of the production reactor can be met.

The successful performance of the nuclear portion of the MHTGR-NPR is dependent on the reliable and safe operation of the following major systems, processes, and components:

- Reactor fuel, core, and graphite components.
- Nuclear control devices.
- Metallic reactor internals.
- Reactor vessel and support system.
- Circulators.
- Steam generator.
- Fuel/target handling equipment.
- Helium purification system.

The experience base for the design of these systems, processes, and components is discussed in the following sections. The MHTGR-NPR tritium target database and technology are covered in a separate report (GA-A18963).

2. EXPERIENCE BASE

2.1. REACTOR FUEL, CORE, AND GRAPHITE COMPONENTS

An established technology base exists for the reactor core and its major components. Over 200 billion fuel particles have been manufactured by General Atomics (GA), formerly GA Technologies Inc. The performance of the fuel in the FSV power station has been excellent and a very clean primary circuit has been maintained. The MHTGR-NPR graphitic fuel blocks and reflector are almost identical to those in the FSV plant. Based on over 30 years of development, and the operation of five HTGR plants, it can be confidently stated that all of the reactor technology and resources necessary for the MHTGR-NPR are in place.

2.1.1. Fuel

The reference fuel cycle employs high-enriched uranium fuel. This fuel, in the form of dense microspheres of uranium oxycarbide (UCO) or uranium dicarbide (UC₂), is coated in a fluidized bed with multiple layers of pyrolytic carbon and silicon carbide (TRISO* coating). The coated particles are blended together and bonded with a carbonaceous binder into fuel compacts, which are stacked into the fuel holes in the graphite fuel element as shown in Fig. 2-1. The use of coated fuel particles is the key element in HTGR performance and safety. This technology has been advanced over the past 25 years in both the United States and Germany.

*Fuel particle coating with three materials: low-density pyrolytic carbon buffer next to kernel with silicon carbide layer sandwiched between two high-density pyrolytic layers.

(a)

(b)

(c)

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Fig. 2-1. MHTGR fuel technology: (a) fuel particles; (b) fuel compacts; (c) fuel elements

Experience with TRISO UC₂ particles in the U.S. dates back to the late 1960s. The TRISO (U-Th)C₂ developed for FSV in the early 1960s was very similar. In 1973, the TRISO UC₂ design was adopted by GA for commercial HTGR application as part of a separable two particle UC₂/ThO₂ fuel system. The good performance of TRISO UC₂ in the fuel development program supported the technical arguments leading to issuance of a generic license for use of TRISO UC₂ in FSV in 1979.

Fuel compact testing and evaluation have taken place in more than 40 irradiation capsules at Oak Ridge National Laboratory (ORNL) and GA and 33 fuel test elements (FTEs) in the Peach Bottom HTGR (Fig. 2-2). Current testing of TRISO UC₂ in FSV test elements will provide major verification of this fuel for application in the MHTGR-NPR.

The ability of the fuel particles to retain fission products is well understood, and as shown in Fig. 2-3, the integrity is ensured to temperatures of over 2000°C (3632°F), which is several hundred degrees above the maximum fuel temperature postulated for the most serious event (depressurized core conduction cooldown).

Many years of excellent particle fuel performance in U.S. and German He-cooled HTGRs have been experienced. Since 1961, GA has manufactured more than 200 billion TRISO-coated particles for use in power reactors and advanced energy and defense applications. Manufacturing technology and quality control techniques have continually improved.

The performance experience of the highly enriched fuel in the FSV HTGR has been excellent, the primary circuit is very clean, and maintenance personnel exposures have been exceptionally low. Direct hands-on maintenance has been performed on equipment removed from the primary circuit. The low FSV exposure levels are anticipated to remain better than an order of magnitude below those of the U.S. light-water reactors (LWR) (Ref. 2-1). Even higher fuel quality standards are being provided for the MHTGR.

Fig. 2-2. Representative photomicrographs of irradiated TRISO UC_2 particles

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Fig. 2-3. Retention of fission products in refractory-coated fuel particles resulting from proven integrity at high temperature

The TRISO UCO, which is a blend of UC_2 and UO_2 , is the reference fuel for the MHTGR-NPR, with TRISO UC_2 as an alternative. This design has the potential for lower fabrication costs and higher quality but the database is less than that of TRISO UC_2 . Additional fuel and fission product technology development is planned for the commercial MHTGR utilizing low-enriched TRISO UCO fuel. Test programs will be conducted to validate fuel performance and fission product transport models and fuel fabrication processes for the manufacturing of improved quality-coated particle fuel.

2.1.2. Reactor Core

The experience gained in the design, construction, and operation of the core and graphite components in FSV, is used to advantage in the MHTGR-NPR. A comparison of the major reactor core parameters between the FSV and MHTGR-NPR plants is given in Table 2-1.

The MHTGR-NPR reactor core and graphite internals consist of hexagonal fuel/target and reflector elements, permanent reflector blocks, and core support structure. The fuel/target elements, which are almost identical to the FSV fuel/control element design, contain blind holes for fuel compacts and full-length channels for helium coolant flow and target assemblies (Fig. 2-4). The fuel elements are stacked in columns that form an active core annulus, with columns of hexagonal reflector elements located in both the central region and immediately surrounding the active core. A cross sectional view through the reactor is given in Fig. 2-5. The core power density is 5.9 MW/m^3 . The annular active core and outer reflector dimensions allow the maximum power rating of 350 MW(t) while providing a simple heat transport pathway to the vessel wall and the RCCS to ensure acceptable fuel and component temperatures during passive decay heat removal.

The core designs for HTGRs are the result of an evolutionary process that started at GA in 1958 and which has been consistently improved

TABLE 2-1
COMPARISON OF MHTGR-NPR AND FSV REACTOR CORE PARAMETERS

	Plant	
	MHTGR-NPR	FSV
Core thermal rating, MW(t)	350/module	842
Core geometry	Annular	Cylindrical
Fuel element	Hexagonal block	Hexagonal block
Active core diameter, ft	11.5 (o.d.) 5.4 (i.d.)	19.7
Active core height, ft	26.0	15.6
Fuel column height	10 elements	6 elements
Core power density, W/cc	5.9	6.3
Residence time, years	3.0	6.0
Maximum fluence, neutrons per cm ² E > 0.18 MeV	4.2 x 10 ²¹	8 x 10 ²¹
Fuel cycle	HEU	HEU/Th
Control rod type	Gravity; boron carbide in graphite/ clad	Gravity; boron carbide in graphite/clad
Control rod location	Reflector element	Fuel elements
System maximum helium pressure, psia	925	700
Core inlet temperature, °F	497	761
Core exit temperature, °F	1268	1444
Core flow control	Fixed orifice in upper element	7-column regions, orifice valve
Fuel element graphite	H-451	H-327/H-451
Reserve shutdown	Gravity drop/ boronated graphite pellets	Gravity drop/boronated graphite pellets
Reactor core pressure loss, psi	3.9	8.4
Upper plenum element rigidity	Metal, keyed together	Metal, keyed together and to core barrel
Graphite core support	Single-column block, single-post column, no keying	7-column region, single block on two 3-posts, keyed together and to core barrel

Fig. 2-4. MHTGR-NPR fuel element

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Fig. 2-5. MHTGR-NPR section

by technology development programs since then. Testing, licensing, construction, and operational experience confirm the validity of the physics and thermal hydraulic methods used to design the core. Extensive analytical and design studies have been performed for the MHTGR annular reactor core, and details are given in Ref. 2-2.

The design of the fuel elements for FSV started in 1964. A hexagonal fuel element geometry based on a close-packed array yields high-strength characteristics. The same hexagonal fuel element will be utilized in the MHTGR-NPR, and appreciation of the size and major features of the block is given in Fig. 2-6.

2.1.3. Graphite Components

A significant base of graphite technology exists after 30 years of research and development and the construction and operation of about 50 graphite-moderated gas-cooled reactors including five HTGR plants. The data are sufficient to complete the MHTGR-NPR design assuming that all safety and performance requirements will be conservatively met.

Nuclear applications of graphite for moderators and reflectors started with the first atomic pile at the University of Chicago. Graphite was later chosen for the production reactor at the U.S. DOE Hanford site and for a variety of research reactors throughout the world. All gas-cooled reactors in Europe and the United States (Table 2-2), both CO₂ and He-cooled, have been graphite-moderated reactors. Graphite has the following attributes for use as a moderator:

- Low absorption cross section.
- Good moderation and structural properties.
- High-temperature capability.
- Structural and chemical stability.
- High thermal capacitance.

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Fig. 2-6. FSV graphite block

TABLE 2-2
GRAPHITE-MODERATED GAS-COOLED REACTORS

Plant	Advanced		Dragon HTGR	Peach Bottom HTGR	FSV	AVR Plant	THTR Plant
	Magnox CO ₂ -Cooled Reactor	Gas-Cooled Reactor (CO ₂)					
Country	UK	UK	UK	U.S.	U.S.	FRG	FRG
Number of plants	37	11	1	1	1	1	1
Power, MW(t)	130 to 2000	110 to 1500	20	115	842	46	750
Constructor	NNC	NNC	UKAEA	GA	GA	BBC/KRUPP	HRB
Operational dates	1958-	1963-	1966-1975	1967-1974	1976-	1967-	1985-
Pressure vessel	Steel	PCRVR	Steel	Steel	PCRVR	Steel	PCRVR
Fuel element type	Pin in block	Pin in block	Multiple	Cylindrical	Prismatic	Spherical	Spherical
Fuel enrichment, %	Natural	2 to 3	93	93	93	93	93
Reactor outlet temperature, °C	400	675	750	730	840	950	750
Gas pressure, Pa x 10 ⁵	8 to 43	20 to 43	20	24	49	11	39
Current status	Operational	Operational	Decommissioned	Decommissioned	Operational	Operational	Operational

Graphite-related topics addressed in the following sections include: (1) experimental programs, (2) experience from FSV, and (3) MHTGR-NPR design.

2.1.3.1. Graphite Experimental Data/Technology Bases. An extensive database has been developed on various grades of graphite utilized in gas-cooled reactors including H-451, SC 2020, H-327, and PGX. Fundamental material properties were obtained from extensive tests of fuel element graphites to provide a strong basis for MHTGR design analysis predictions.

A spectrum of test results is available, ranging from small-specimen tests to determine material properties to full-size component assembly tests to verify design load capacity and identify failure modes. Fuel element across flat load, dowel shear load, and impact load capacity and core support structure loading, were confirmed in large-size unirradiated component tests (Figs. 2-7 and 2-8). Irradiation tests on simplified elements were used to determine the irradiation conditions that would cause graphite cracking. In general, actual failure stress levels were higher than predicted.

The dynamic behavior of the hexagonal block core assemblies during a seismic event has been determined under an extensive test program, which included shaker tests on a 1/5-scale, full-array core representing an earlier 3000-MW(t) plant design (Fig. 2-9). The test results were used to verify core seismic analysis methods incorporating nonlinear models.

Although it is now recognized that the graphite core did not adversely impact the events at Chernobyl, initial reports of graphite "fire" (rapid, self-supported oxidation) has drawn attention to graphite-moderated cores, and this has fostered special analysis and experiments. Experiments on representative core graphite blocks

(a)

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(b)

(c)

Fig. 2-7. Full-scale fuel element mechanical tests: (a) shear test apparatus; (b) control rod fuel element static load test apparatus; (c) fuel element impact test apparatus.

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Fig. 2-8. Three-quarter-scale FSV core support assembly ultimate load verification test: test apparatus

(a)

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(b)

Fig. 2-9. One-fifth-scale seismic test of the 3000-MW(t) Philadelphia Electric core design

(characterized by a low-surface area-to-volume ratio) have revealed no rational conditions under which rapid oxidation could be induced. This has included prolonged oxyacetylene flame impingement and exposure of the white-hot block to an oxygen stream. Burnoff was mostly confined to the surface at high temperatures, but a runaway condition of graphite "fire" could not be induced. It has been shown (Ref. 2-3) that graphite oxidation by air ingress, even in the postulated case of a catastrophic vessel rupture, is insignificant in the silo containment configuration where only a small volume of air is available. Further information on graphite oxidation in the MHTGR has been documented (Ref. 2-4).

Graphite technology development is an ongoing research program at ORNL. Sponsored by DOE, it aims to refine graphite material behavior models, static failure models, fatigue failure models, and irradiation effects, all of which are in direct support of the graphite structural design criteria formulated by GA with input from the Nuclear Regulatory Commission (NRC), the Franklin Institute, and the Joint ASME/ACI Subgroup for Graphite Core Supports.* The additional test data is expected to allow increases in allowable stress because of reduced uncertainty in the material's property data.

External evaluations and independent reviews of the graphite fuel element and core support design of the FSV reactor and the proposed large commercial HTGRs for Delmarva and Philadelphia Electric were performed by the NRC and its consultants: Franklin Institute, Brookhaven National Laboratory, Battelle Northwest Laboratories, and Los Alamos National Laboratory. In particular, a comprehensive review of the graphite design criteria for the HTGR was performed by the Franklin Institute and sponsored by the Reactor Safety Research Division of NRC.

*ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CE, for gas-cooled reactor design (ACI-359).

2.1.3.2. Graphite Component Experience from FSV. The FSV plant has undergone three refueling operations, and with the removal of 750 blocks, visual inspection has been carried out for signs of damage in the form of cracks or oxidation. The elements have been subject to the same dose as for the MHTGR design. In general, the elements looked as they did when originally loaded into the core. No harmful damage was found, and there were no chips, breaks, gouges, or broken dowel pins. Measured dimensional changes were as predicted.

At the first refueling, it was observed in two adjacent blocks that a hairline vertical crack had developed, extending from the outside face to the nearest cooling hole. Although such cracks have no effect on the safety or operability of the core, a thorough investigation was performed to provide verification.

Prior structural analysis did not account for gap flow between the blocks. Subsequent analysis was performed based on (1) more refined stress analysis, (2) modeling of block gap flow, and (3) reactor operating history. This analysis indicated a local high-stress area in the location where the cracks were observed related to the gap flow between the blocks. Appropriate changes have been made to the core design to ensure that the situation will not occur in the MHTGR-NPR. After two further refueling operations at FSV, no further cracks have been observed in any of the blocks, and the initial occurrence of the two hairline cracks is not viewed as a safety issue.

Temperature fluctuations in the FSV core during initial reactor operation at power levels above about 30% were experienced and where orifice setting established high-pressure drops. These were determined to be a result of thermal-hydraulic mechanisms. Helium flow in the gaps between fuel element regions created lateral pressure forces on regions, and the resultant nonuniform gap cooling caused bowing and this resulted in a cyclic bowing behavior. The cyclic bowing behaviour resulted in localized fluctuations in the core outlet temperature. The problem in

FSV was eliminated by utilizing region constraint devices to hold the tops of the regions to near nominal positions, thus stabilizing the core array at the top.

The experience gained from resolving the problem in FSV has been factored into all subsequent HTGR designs. For the MHTGR-NPR the blocks will be keyed together (to give the functional equivalent of the FSV constraint devices); the core pressure loss is only 3.9 psi compared with 8.4 psi in FSV (hence, less gap flow effect). GA is familiar with core flow and structural testing (Fig. 2-10) and to confirm the design, reduced scale tests will be performed for the 10-block-high MHTGR-NPR core.

2.1.3.3. MHTGR-NPR Graphite Component Design. Improvements in the fuel element and core support designs have been brought about by FSV operating experience, on-going development and test activities, and design reviews. Other improvements have resulted from the need to meet more stringent regulatory requirements. These are summarized as follows:

- Fuel element graphite material was changed to Grade H-451, which has improved isotropy strength properties over H-327. (H-451 is also being used in the reload segments at FSV.)
- A single-post-per-column-support concept with redundant load path was introduced to simplify the design and enable in-service inspection of the lower core plenum and support posts.
- Core support blocks and post materials were replaced with 2020 graphite, which has higher-strength, lower-oxidation-rate properties compared to the PGX and ATJ graphites used in FSV.

(a)

(b)

(c)

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Fig. 2-10. Core flow and structural tests performed by the CEA, France: (a) full-scale single refueling region helium flow test; (b) One-quarter-scale core support block flow visualization test plastic model; (c) one-quarter-scale core support post buckling test apparatus.

- Interkeyed metallic plenum elements are provided within each refueling sector, which eliminates core-column-temperature fluctuations during operation.

2.2. NUCLEAR CONTROL DEVICES

The control rods and the reserve shutdown material are installed in penetrations in the top of the reactor vessel, which also house the drive mechanisms for positioning the control rods, and rod controls, and the reserve shutdown control equipment. These assemblies are periodically removed either to provide access to the core for refueling or for maintenance of the equipment.

The design of the control rod drive system used for reactor control and shutdown is derived directly from very similar cable/drum gravity drop systems that have operated successfully in other gas-cooled reactors including FSV and Dragon. If the control rod drive mechanism should fail, the control rods will fall by gravity into the core to shutdown the reactor. The backup system consists of boronated graphite balls (about the size of marbles) that fall into the core to stop the chain reaction. If the system should fail, the reactor would automatically shut itself down when the core starts to overheat due to its large negative temperature coefficient. This performance characteristic, which is truly fail-safe, has been confirmed by testing at the AVR, Peach Bottom, THTR, and FSV reactors.

The design of the neutron control system equipment for the MHTGR-NPR draws heavily on experience from FSV. The FSV control assembly is shown in Fig. 2-11. The control rod drive mechanisms from FSV are shown in Figs. 2-12 and 2-13.

Moveable control rods located in channels in the inner ring of the hexagonal side reflector and in the outer ring of the central reflector control the rate of fission energy generation. The 24 rods in the side reflector are used for normal control and reactor trip. The six rods in the central reflector are only inserted for cold shutdown. These rods, when fully inserted into the core, assure an adequate shutdown margin when the reactor is not operating. The MHTGR-NPR control rod assembly is shown in Fig. 2-14.

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Fig. 2-11. FSV control and orificing assembly

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Fig. 2-12. Overall view of FSV control rod drive

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Fig. 2-13. FSV control rod drive assembly

Fig. 2-14. Control rod assembly: (a) drive mechanism and guide tubes; (b) control rod element

The neutron control system for the MHTGR-NPR is based on proven equipment, and a comparison with the equipment in FSV is given in Table 2-3. An inherent part of the development testing will be verification of the equipment over the full spectrum of reactor operation, for both normal and abnormal events.

TABLE 2-3
COMPARISON OF FSV AND MHTGR CONTROL ROD DRIVE FEATURES

Feature	FSV	MHTGR
Insertion direction	Top entry	Top entry
Method of rod support	Cable	Cable
Rod length, ft	15.6	26
Rod weight, lb	120	206
Total stroke, ft	15.9	30.5
Rods/drive motor	2	1
Regulating speed, in./s	1.05	1.2
Motor type	AC induction motor with electromagnetic brake	Brushless dc torque motor
Scram insertion method	Gravity with dynamic braking	Gravity with dynamic braking
Scram time, maximum, s	152	25

2.3. METALLIC REACTOR INTERNALS

The major metallic reactor internals include the core support structure below the core, the core lateral restraint structure located between the outer side reflector and the reactor vessel, an upper plenum shroud attached to the core lateral restraint structure in order to reduce the thermal load on the reactor vessel head, and a hot gas duct located within the cross duct to link the core outlet plenum with the steam generator inlet. All components are designed from Alloy 800H material. The location of these components is shown in Fig. 2-15.

The technology for the metallic materials is well established and a sufficient database exists from the two MHTGRs operated in the U.S. Designs similar to the MHTGR-NPR in the areas of core support structures (Fig. 2-16), shrouds, thermal barriers, and ducts have all performed satisfactorily. The major criteria for the internals (i.e., pressure, temperature, loading, seismic, radiological) are bounded by operating experience.

The technology for the hot gas duct (which is used to facilitate thermal energy transfer from the reactor to the steam generator) has been established by similar systems in other gas-cooled reactors and component tests on the large MHTGR (Fig. 2-17). An important consideration is that the principle of the coaxial hot gas duct was incorporated in the Peach Bottom 1 HTGR reactor. The integrity of the duct assembly was demonstrated during the 7-1/2 years of successful operation (Ref. 2-5). Since the MHTGR-NPR hot duct (Fig. 2-18) is designed from similar but higher grade materials, and has approximately the same size and operating environment (i.e., temperature, gas velocity, acoustics, pressure, etc.), the existing technology base is directly applicable.

Fig. 2-15. Reactor internals metallic components

Fig. 2-16. MHTGR metallic core support structure - isometric and elevation view

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Fig. 2-17. Hot gas duct technology

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Fig. 2-18. Hot duct assembly - elevation view

2.4. VESSELS AND SUPPORT SYSTEM

The nuclear steam supply system, as illustrated simply in Fig. 2-19, is characterized by three major vessels: (1) reactor vessel, (2) steam generator vessel, and (3) the interconnecting cross vessel. The short horizontal vessel is essentially an extension of the two nozzles from the vertical assemblies and is subject to the same criteria and design philosophy as the vessel containing the core and steam generator.

Steel vessels for nuclear reactors have been manufactured for over 30 years and have performed exceptionally well in over 400 commercial reactors as well as in numerous nuclear submarines and aircraft carriers. Furthermore, the helium-cooled Dragon, Peach Bottom, and AVR HTGRs all have operated safely and successfully with steel vessels. Their experience base is extensive and proven.

Details of the three vessels are given in Table 2-4. The vessels are designed using LWR technology and are comparable in size and weight to current large, LWR vessels as illustrated in Fig. 2-20. The steam generator and cross vessel are insulated in the exterior surfaces to minimize heat loss during operation. The reactor vessel is intentionally noninsulated so that in the event all forced core cooling is lost, the core residual and decay heat from the shutdown core can be passively radiated from the vessel walls to the RCCS.

Although extensive property data exists for the SA-533 material used in the MHTGR and LWR vessels, additional tests to augment the data for nil ductility transition temperature shifts are planned for at the lower design basis temperatures for the MHTGR-NPR vessel.

The reactor vessel is supported by three flexible columns anchored to the vessel at the cross duct level. Keys are provided at both the top of the reactor vessel and at the support elevation to accommodate vertical and radial thermal expansion while providing lateral seismic

Fig. 2-19. Isometric view of reactor system for steam cycle MHTGR concept

TABLE 2-4
STEEL VESSELS FOR MHTGR-NPR PLANT

Vessel	LWR For Comparison			
	Reactor Vessel	Steam Generator Vessel	Interconnecting Cross Vessel	Reactor Vessel Steam Generator Vessel
Vessel orientation	Vertical	Vertical	Horizontal	Vertical
Material	Mn-Mo; SA-533, Grade B	Mn-Mo; SA-533, Grade B	Mn-Mo; SA-533, Grade B	Mn-Mo; SA-533, Grade B
Operating temperature, °F	400	490	500	650
Operating pressure, psia	925	925	925	2250
Nominal diameter, ft	22.35	13.70	6.33	17.0
Nominal thickness, in.	5.25	3.5	2.5	11.2
Overall length, ft	95	87	14	48.5
Weight, ton	650/300	613	--	615
Support location	Bottom plane	At cross vessel	--	Bottom plane
Vessel insulated	No	External	External	External
Fabrication	Factory	Factory	Factory	Factory
Transportation to site	Use of existing and proven equipment	Use of existing and proven equipment	Use of existing and proven equipment	Many sites in U.S. via barge or C-E railcar
Field weld	No	No	Weld to complete vessel coupling	No
ASME Code classification	Section III Class 1	Section III Class 1	Section III Class 1	Section III Class 1
Technology base	Combustion Engineering System 80 PWR	Combustion Engineering System 80 PWR	Combustion Engineering System 80 PWR	Combustion Engineering System 80 PWR

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Fig. 2-20. Comparison of MHTGR and LWR vessels

restraint. The radial center of the reactor core is maintained stationary at all times. The steam generator vessel is supported by two sliding bases, allowing thermal expansion relative to the reactor vessel. A pair of keys and snubbers accommodate radial and vertical translation along the axis of the cross duct, and provide seismic restraint.

The support design is based on the principals of Combustion Engineering System 80 for commercial pressurized water reactors (PWRs). Similar sliding supports and hardware have been operating successfully in a much more severe environment compared to that of the MHTGR-NPR. A large database exists on all materials employed for the design.

All major nuclear system components would be factory-fabricated and transported to the site. In the design of the MHTGR-NPR, full advantage is being taken of existing technology from the LWR industry, and this includes vessel design to ensure that the weights and envelopes are consistent with proven transportation capability to reactor site (Fig. 2-21).

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Fig. 2-21. Overland transportation of large factory-fabricated components

2.5. HELIUM CIRCULATOR

An established technology base exists for the MHTGR-NPR circulator. Over 200 circulators with oil-lubricated bearings and electric motor drives have operated successfully in the U.S. and Europe. The choice of oil-lubricated bearings (Table 2-5) for the MHTGR-NPR plant circulator was based on the goal of utilizing proven technology. The MHTGR vertical circulator assembly is physically smaller, has a lower power requirement, and is simpler than the vertical machines (with submerged electric motor drives) used in the British AGR plants.

The design of the circulator for the MHTGR-NPR is derived directly from circulators being successfully operated in gas-cooled reactors. The FSV circulators (Fig. 2-22) have met all design specifications, however, the bearings seals and support systems for the water-lubricated bearing have caused many problems. Further, the circulators employed a steam turbine drive that adds complexity to system operations. These unique design features resulted in water ingress to the core, the primary reason for poor plant availability.

The main helium circulator is vertically mounted on the top of the steam generator vessel, and thus is readily accessible from the upper region of the silo. The circulator consists of a single-stage axial flow compressor driven by a 4250 hp submerged electric motor. The machine arrangement is shown in Fig. 2-23, and the major features are compared with contemporary machines in Table 2-6. Full details of the machine design for the MHTGR utility plant have been described previously (Ref. 2-6).

The most significant circulator technology base is that for the carbon-dioxide-cooled AGR plants in the UK. An example of the vertical circulators for the Hartlepool AGR is shown in Fig. 2-24. These circulators have submerged electric motor drivers and oil-lubricated bearings. Rated at 6600 hp, the machines are more powerful, and physically

TABLE 2-5
CIRCULATOR EXPERIENCE WITH OIL-LUBRICATED BEARINGS

- Peach Bottom 1 HTGR
 - Externally mounted circulators
 - Machines performed well for 7-1/2-year plant life
- Magnox and AGR stations (UK)
 - Over 200 oil-bearing circulators operational
 - Virtually zero oil ingress (2 grams/day, 0.2 ppm in CO₂)
 - Several million operating hours
 - High circulator availability (99.4% at Hinkley Point B)
- AVR 1 (experimental HTGR in Germany)
 - 20 years of operation
 - 133,000 h trouble-free operation on each circulator with oil-lubricated bearings
 - One machine maintenance free since 1966
- THTR (commercial HTR in Germany)
 - Circulators in operation since 1984
 - Oil-lubricated bearing machines (6/plant)
 - Trouble-free circulator operation to date (over 100,000 h)

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Fig. 2-22. Helium circulator for FSV plant

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Fig. 2-23. MHTGR main circulator reactor installation

TABLE 2-6
CIRCULATOR DESIGN FEATURES COMPARISON

Plant	HTCR	FSV	770 MW(e) HTCR	AVR 1	THTR 300	Insulation Test Loop	Current AGR
Country	U.S.	U.S.	U.S.	Germany	Germany	Germany	UK
Gas	Helium	Helium	Helium	Helium	Helium	Helium	Carbon dioxide
Circulators/reactor	1/module	4	4	2	6	1	8
Circulator orientation	Vertical	Vertical	Vertical	Horizontal	Horizontal	Horizontal	Horizontal and vertical
Compressor type	Single-stage axial flow	Single-stage axial flow	Single-stage axial flow	Centrifugal	Centrifugal	Single-stage axial flow	Centrifugal
Compressor tip diameter, mm (in.)	889 (35)	711 (28)	1,067 (42)	710 (28)	900 (35.4)	508 (20)	813 (32)
Tip speed, m/s (ft/s)	289 (947)	356 (1167)	377 (1238)	149 (490)	264 (866)	96 (315)	126 (415)
Gas inlet pressure, MPa (psia)	6.29 (912)	4.73 (686)	4.85 (704)	1.1 (158)	3.8 (551)	6.0 (870)	3.95 (573)
Pressure rise, kPa (psia)	91 (13.2)	96.5 (14)	145 (21)	0.83 (11.2)	124 (18)	1.0 (0.15)	290 (42)
Inlet temperature, °C (°F)	255 (491)	394 (742)	311 (592)	200 (392)	250 (482)	950 (1,742)	295 (563)
Rotational speed, rpm	6,200	9,550	6,750	4,000	5,600	3,600	2,970
Power, kW(e) (hp)	3,210 (4,302)	3,954 (5,300)	10,670 (14,300)	220 (165)	2,300 (3,083)	20 (27)	4,924 (6,600)
Flow control	Variable speed	Variable speed	Variable speed	Variable speed	Variable speed	Variable speed	Variable inlet guide vanes
Drive type	Submerged electric motor	Steam turbine	Steam turbine	Submerged electric motor	Submerged electric motor	Integrated electric motor	Submerged electric motor
Bearing type	Oil (alternating magnetic)	Water lubricated	Water lubricated	Oil	Oil	Active magnetic	Oil
Machine status	Preliminary design	Plant decommissioned	Operational machine	Over 131,000 h operation	Operational machine	Operational machine	Over 1,000,000 h operation

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- 116 CIRCULATORS FABRICATED FOR AGR PLANTS BY JAMES HOWDEN AND COMPANY (U.K.)
- VERTICAL AND HORIZONTAL MACHINES BUILT
- RADIAL FLOW COMPRESSOR
- OIL-LUBRICATED BEARINGS
- SUBMERGED ELECTRIC MOTOR DRIVE
- OVER ONE MILLION OPERATING HOURS
- LEAD MACHINE OVER 70,000 HOURS (MAINTENANCE FREE)
- CIRCULATOR AVAILABILITY 99.4%

Fig. 2-24. Hartlepool AGR CO₂ circulator

bigger than those for the MHTGR-NPR, although many of the features would be common. Circulators for the AGR have operated for over a million hours with high availabilities (99.4% Hartlepool). Oil ingress to the reactor system is minimal and is not a problem.

The ultimate bearing system for closed circuits can be realized by an emerging tribology approach - magnetic bearings (Ref. 2-7). With this approach, the circulator rotor would be magnetically suspended, and there would be no contact between the shaft and bearings. Active magnetic bearings have been introduced for industrial machinery, and a total of 372 machines are operational in Europe, Japan, and the U.S. It was the development of modern electronic equipment that made magnetic bearings a reality since the positioning and control of the levitated rotor is done by a computer system involving real-time monitoring of the shaft. To date, over 45,000 h of magnetic bearing operation has been accumulated. In Germany, a small helium circulator with active magnetic bearings (Fig. 2-25) has operated for over 15,000 h, trouble-free in a high-temperature insulation test rig.

Magnetic bearing development efforts in support of the MHTGR circulator are underway. As this development advances, and maturity is reached in other industries, a lubricant-free circulator will be introduced for the MHTGR. Both the oil and magnetic bearing variants have identical vessel interfaces and the oil-bearing machines could be retrofitted with a magnetic bearing machine as it becomes available. An example of this type of machine is given in Fig. 2-26.

As the circulator in the MHTGR is mounted on the top of the steam generator, it is very accessible. The complete circulator assembly can be removed and replaced in an estimated time of 3 days, and plant downtime would be minimized in the event of a circulator malfunction.

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MAJOR PARAMETERS

- BBC/HRB HELIUM CIRCULATOR
- OPERATING IN THERMAL INSULATION TEST LOOP AT HRB, JULICH
- SHAFT DIAMETER 3.15 IN.
- ROTOR WEIGHT 570 LB
- SPEED 3600 RPM (MAX SPEED)
- POWER 20 kW(e)
- SINGLE STAGE AXIAL COMPRESSOR
- HORIZONTAL INSTALLATION
- BEARING SPAN 20 IN.
- BALL BEARING CATCHER SYSTEM
- OPERATION INITIATED IN SEPTEMBER 1982
- SERVICE OPERATION - INTERMITTENT
- NORMAL STARTS AND SHUTDOWNS - 20/YR
- APPROXIMATELY 15,000 HOURS OF OPERATION
- NO FAILURES OF BEARING COILS OR SENSORS
- CATCHER BEARINGS HAVE NOT BEEN REQUIRED TO DATE

Fig. 2-25. Operational helium circulator with magnetic bearings

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Fig. 2-26. MHTGR circulator concept

2.6. STEAM GENERATOR

An established technology base exists for the MHTGR-NPR steam generator. Thirty-four helical steam generators have been fabricated and successfully operated in nuclear power plants, including FSV (Fig. 2-27), the AGR (Fig. 2-28), and the THTR (Fig. 2-29). The performance history of these systems has been outstanding with tube failure rates significantly lower than for water reactor systems. To date, the FSV steam generator has only experienced two pin hole tube leaks, which have been plugged by remote equipment. The MHTGR-NPR steam generator is larger than the ones in the THTR reactor and at FSV but smaller than those designed for the Delmarva and Philadelphia Electric plants for which fabrication work had been started before project cancellation.

The steam generator is a once-through, helically wound, tubular type heat changer located in the steam generator vessel. It has a thermal rating of 350 MW(t). As shown in Fig. 2-30, the unit consists of an economizer, an evaporator, and a first- and second-stage superheater. A comparison with contemporary steam generators is given in Table 2-7. Details of the steam generator design are given in Ref. 2-8.

The MHTGR-NPR steam generator design has been supported by an intensive materials development program which was established in the mid-1970s for the commercial HTGR program. It included long-term tests on materials such as time-dependent creep-rupture and creep-fatigue of structural materials and weldments, bimetallic weld cyclic and static load tests, destructive specimen testing of a large size forging of steam outlet tubesheet, and sliding and fretting wear tests on steam generator tube material samples and actual tubes.

Other technology development programs conducted for the commercial HTGR projects have also been of direct benefit. Tests that have yielded significantly useful data include a 1/4-scale air flow test of the MK II-B straight tube superheater to helical bundle transition section,

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Fig. 2-27. FSV steam generator

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Fig. 2-28. AGR Hartlepool helical coil steam generator

Fig. 2-29. THTR steam generator

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Fig. 2-30. Helical bundle steam generator configuration for MHTGR

TABLE 2-7
COMPARISON OF HTGR/AGR STEAM GENERATORS

Unit	MHTGR	FSV (U.S.)	THTR (FRG)	Hartlepool AGR	C-E System 80 PWR Steam Generator
Number per reactor	1/module	12	6	8	2
Unit thermal rating, MW	350	70	128	190	1,900
Gas reheat	No	Yes	Yes	Yes	N/A
Bundle geometry	Helical	Helical	Helical	Helical	U-tubes
Helium inlet temperature, °C (°F)	686 (1266)	775 (1427)	750 (1382)	650 (1202)	327 (621)(a)
Helium pressure, MPa (psia)	6.38 (925)	483 (700)	3.85 (558)	4.13 (600)	15.5 (2250)
Steam outlet temperature, °C (°F)	541 (1005)	541 (1005)	550 (1022)	543 (1009)	289 (553)(b)
Steam pressure, MPa (psia)	17.3 (2515)	17.3 (2512)	18.6 (2700)	17.1 (2480)	7.4 (1070)
Number of tube circuits	439	18/54	40/80	285	11,012
Tube outer diameter, in.	0.875	1.0/1.125	0.98/1.06	1.125/1.50	0.75
Tube wall thickness, in.	0.13	0.125-0.205/0.14	0.10-0.20/0.16	0.128/0.160	0.042
Active tube surface area, ft ²	43,341	3,330/400	12,770/1,500	19,650/2,680	128,800
Materials	Alloy 800H/ 2-1/4 Cr - 1 Mo	Alloy 800H/ 2-1/4 Cr - 1 Mo	Alloy 800H/ 2-1/4 Cr - 1 Mo	9 Cr - 1 Mo/ 316H stainless	Inconel 163
Overall height, m (ft)	26.5 (87)	17 (55)	19 (61)	19.5 (64)	9.7 (32)(c)
Overall diameter, m (ft)	4.7 (15.4)	1.7 (5.6)	2 (6.6)	2.5 (8.1)	4.5 (14.8)
Approximately dry weight, metric tonnes	291	23	65	--	659
Installation	In steel vessel	In prestressed concrete vessel	In prestressed concrete vessel	In prestressed concrete vessel	In steel vessel
Boiling flow direction	Upflow	Upflow	Downflow	Upflow	Upflow
Tube plugging access	Manual	Manual (reheat-remote)	Manual	Manual	Manual
Access for removal/reinstallation	Good	Difficult	Easy	Good	Difficult
Drainability	Good	Good	Poor	Good	Good
Status	Preliminary design	Operational	Operational	Operational	Operational

(a) Water conditions at the primary side (inside tubes).

(b) Water conditions at shell side.

(c) To the top of U-tubes.

single tube boiling stability test performed for the FSV steam generator orificing design, flow-induced cross flow tube vibration model test, and large helical bundle tube support fabrication test.

Seismic shaker tests (Fig. 2-31) were performed on a section of the tube bundle for the 3000-MW(t) Philadelphia Electric steam generator to confirm tube bundle dynamic response and determine structural damping.

The steam generator database will be further augmented by planned development programs such as material property tests for weld materials (2-1/4 Cr - 1 Mo and Alloy 800H) at elevated temperatures.

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Fig. 2-31. Seismic shaker test large HTGR steam generator with bundle

2.7. FUEL/TARGET HANDLING EQUIPMENT

The fuel handling systems for HTGRs have evolved for nearly 25 years and have been refined from experience of similar systems servicing the Peach Bottom and FSV reactors as well as British gas-cooled reactors. During this period, a strong technical foundation for fuel handling system design has been established in terms of acceptable materials, lubricants, mechanical and electrical components, operating speeds, analytical methods, and interface requirements with the core and other reactor components. The comparable hardware and software at FSV, including the fuel handling equipment positioner, the fuel transfer cask, and the fuel handling machine have performed well. An overall view of the FSV fuel handling machine is shown in Fig. 2-32, and details of the all-important machine grapple are given in Fig. 2-33.

The major fuel/target handling equipment at the reactor includes the fuel handling machine and transfer casks. "In-vessel" core refueling equipment is used to perform the periodic remote replacement of fuel and reflector elements and is shown schematically in Fig. 2-34. The refueling equipment, which is operated in the upper core plenum during refueling, is installed and removed through one of the refueling penetrations in the reactor vessel top head.

The refueling/retargeting occurs after the reactor has been shut-down and depressurized. The exchange of elements and target assemblies occurs in one-sixth of the core area at a time. A digital computer is used to control all operations and to monitor related activities. Time to perform refueling/retargeting operations including the time to cool-down and to return to full power is less than 12 days.

At FSV, the concept of a computerized control system was introduced and provided valuable experience relative to the mating of hardware to the computer and the interaction of the operating personnel with a computer-controlled system.

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Fig. 2-32. FSV fuel handling machine

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Fig. 2-33. FSV fuel handling machine grapple

Fig. 2-34. In-vessel fuel handling equipment

Although fuel handling equipment is different for the MHTGR-NPR, FSV time and motion data provide an actual basis for the estimated 12-day time for refueling/retargeting.

2.8. HELIUM PURIFICATION SYSTEM

The helium purification system consists of standard chemical and physical operations for the removal of impurities such as moisture, dust, radioactive gases, and contaminants from the process helium. Clean helium is returned to the primary coolant loop as a purge gas for selected vessel penetrations and for the main and shutdown circulators.

This system, which is essentially the same as that used successfully in the Peach Bottom and FSV reactors, is one of the factors contributing to the extremely low level of radiation exposure at FSV. The size of the purification system for the MHTGR-NPR takes into account the experience at FSV.

Although there is a loss of pure, clean helium at FSV due to helium leakage through flanges and valves, the THTR has a welded system and improved valves and has experienced very little leakage. The MHTGR-NPR will also utilize a welded system to minimize helium leakage.

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- 2-8. McDonald, C. F., and D. D. DeFur, "Steam Generator Design Considerations for Modular HTGR Plant," Proceedings of 21st IECEC, 1, pp. 281-286, August 1986.

3. CONCLUSIONS

The experience base for the nuclear portion of the MHTGR-NPR has been accumulated over several decades and constitutes more than a billion dollars of investment. It provides a solid base for constructing and operating a NPR. The operational experience from FSV (Ref. 3-1) is a vital part of this experience base. Additionally, the basic HTGR technology and designs have in the past received extensive review by the MHTGR program participants including General Atomics, Combustion Engineering, General Electric, Oak Ridge National Laboratory, Stone and Webster, and representatives of the nuclear utilities who are members of Gas-Cooled Reactor Associates. These reviews confirm that the technology is available to provide an MHTGR-NPR which satisfies the technical, schedule, and performance requirements for a New Production Reactor.

Following the construction of five HTGR plants and over three decades of research and development, the MHTGR-NPR is based on demonstrated technology. Deployment of the MHTGR-NPR to meet the requirements can be confidently projected based on the following: (1) established analytical tools and methods (i.e., computer codes) are already in place; (2) established designs, standards, codes, and procedures; (3) no need for basic research; and (4) design verification needs have been fully identified and are limited.

The NPR effort will benefit from the extensive and detailed design and design verification program developed as part of the DOE MHTGR program, which will provide additional assurance of successful system operation.

3.1. REFERENCE

- 3-1. Kantor, M. E., and R. A. Moore, "The MHTGR Experience Base at Fort St. Vrain," Proceedings of 18th IECEC, 1, pp. 42-47, 1983.

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