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MEDIUM-SIZE HIGH-TEMPERATURE GAS-COOLED REACTOR

by

C. O. PEINADO and S. L. KOUTZ

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ABSTRACT

This report summarizes high-temperature gas-cooled reactor (HTGR) experience for the 40-MW(e) Peach Bottom Nuclear Generating Station of Philadelphia Electric Company and the 330-MW(e) Fort St. Vrain Nuclear Generating Station of the Public Service Company of Colorado. Both reactors are graphite moderated and helium cooled, operating at ~760°C (1400°F) and using the uranium/thorium fuel cycle. The plants have demonstrated the inherent safety characteristics, the low activation of components, and the high efficiency associated with the HTGR concept. This experience has been translated into the conceptual design of a medium-sized 1170-MW(t) HTGR for generation of 450 MW of electric power. The concept incorporates inherent HTGR safety characteristics [a multiply redundant prestressed concrete reactor vessel (PCRV), a graphite core, and an inert single-phase coolant] and engineered safety features (core auxiliary cooling, relief valve, and steam generator dump systems). Enhanced safety options are being considered which include improved feedwater turbines, natural helium convection, and afterheat removal PCRV helium blowdown.

1. INTRODUCTION

Based on experience from CO_2-cooled Magnox reactors in the United Kingdom, initial high-temperature gas-cooled reactor (HTGR) development was directed toward higher coolant temperatures. Advantages foreseen were more efficient steam-powered electrical power plants and the potential for use in high temperature process heat applications and closed-cycle gas turbine power trains.

The goal of higher temperatures led to the choice of high pressure helium as the primary coolant and graphite as the moderator. The availability of highly enriched uranium in the United States resulted in the selection of a semihomogenous fuel element design optimized for the uranium-thorium fuel cycle. Reprocessing and recycle of U-233 were planned.

Recently, concerns have been expressed regarding the risks of potential proliferation/diversion of fissionable material inherent in various nuclear reactor fuel cycles. This concern has led to some modifications in the fuel cycle for HTGR. European gas-cooled reactors have operated on different fuel cycles and, although fuel cycle economics may have been somewhat downgraded, there were no technical feasibility questions regarding the use of alternative fuel cycles.

All of the HTGR concepts presently being developed and considered for potential commercialization in the United States contemplate the use of prestressed concrete reactor vessels (PCRVs) and the integration within the vessel of basically all of the components associated with heat removal from the helium coolant.

Modularization of the HTGR nuclear steam supply into 585-MW(t) steam generation loops allows a choice in total reactor power. An HTGR containing two loops would have a total thermal power of 1170 MW(t) and would generate 450 MW(e). This plant size results in certain inherent enhanced safety features.

2. HTGR EXPERIENCE

2.1. PHILADELPHIA ELECTRIC, PEACH BOTTOM

The first HTGR in the United States was the 40-MW(e) Peach Bottom Unit 1 plant. This plant, owned and operated by the Philadelphia Electric Company, was built as a prototype to demonstrate a high-performance helium-cooled nuclear power plant. The nuclear steam supply system (NSSS) was designed and supplied by General Atomic Company. The engineer/constructor was Bechtel Corporation. A group of 53 utility companies, comprising High Temperature Reactor Development Associates, Inc., supported construction of the Peach Bottom plant.

The Peach Bottom reactor was highly successful from its initial commercial operation in June 1967 to its scheduled shutdown for decommissioning on October 31, 1974. An overall plant availability of 88% was achieved during this period, with an accumulated operational total of 1349 full-power days, using two reactor cores, for a gross generated total of 1,385,919 MWh.

The NSSS was a helium-cooled, graphite-moderated, 115-MW(t) reactor with cylindrical rod elements enclosed in a steel pressure vessel and with two external helium-circulator, steam-generator loops. The fuel cycle concept was based on batch operation of the reactor core with the complete core of 804 spent fuel elements being replaced at core end of life.

The overall fuel and plant performance of the Peach Bottom HTGR was particularly gratifying. There were no major equipment failures, and the steam generator had no tube leaks throughout the plant operating life. During Peach Bottom operation, excellent agreement was found between predicted and actual core physics characteristics, verifying the methods used and providing a reference data base for application to larger HTGR plants.

2.2. PUBLIC SERVICE COMPANY OF COLORADO, FORT ST. VRAIN

The Fort St. Vrain Nuclear Generating Station, located 56 km (35 miles) north of Denver, Colorado, USA, is the first commercial HTGR. The 330-MW(e) plant is owned and operated by the Public Service Company of Colorado; General Atomic Company was the prime contractor; Sargent & Lundy Engineers was the architect/engineer; and Ebasco Services and Stearns-Roger were the constructor.

The entire primary coolant system (reactor core, steam generators, and helium circulators) are contained within a 32.3-m (106-ft) high prestressed concrete reactor vessel (PCRV). A single reheat steam cycle operates at 16.5 MPa/538°C (2393 psi/1000°F) and uses a standard 3600 rpm tandem-compound steam turbine generator. The net thermal efficiency is projected at 39.4%.

The Fort St. Vrain startup program has had a number of unplanned delays in its initial operation period. The plant has operated to 68% of rated power following initial rise to power in December 1976. Fuel performance and integrity have been excellent. Net plant efficiency at 60% power was 37%. The plant is currently limited to 70% power until regulatory approval for full-power operation is obtained. This approval is contingent on successful demonstration that the core temperature fluctuations can be mitigated by the mechanical devices installed in the core.

3. CONCEPTUAL DESIGN OF THE 1170-MW(t) HTGR

This section describes the 1170-MW(t) HTGR reference plant design for a steam-turbine cycle.

The HTGR reactor core is cooled with pressurized helium, moderated and reflected with graphite, and fueled with a uranium/thorium mixture. It is constructed of prismatic hexagonal graphite blocks with vertical holes for coolant channels, fuel rods, and control rods. The entire reactor core and other major primary system components are contained in a multicavity PCRV. Helium coolant flows from two electric-motor-driven circulators through the core, through two steam generators (each located in separate cavities in the PCRV wall), and back to the circulators. Superheated steam [17.2 MPa, 541°C (2500 psi, 1005°F)] produced in the once-through steam generators is expanded through a tandem compound turbine generator. Steam is condensed in a water-cooled condenser, and waste heat is rejected to the atmosphere in a wet cooling tower. In addition to the two primary coolant loops, three core auxiliary cooling system (CACS) loops are provided. Each CACS loop consists of a gas/water heat exchanger and auxiliary electric-motor-driven circulators located in cavities in the PCRV. If the main loops are not available, coolant gas is circulated from the reactor core through the heat exchangers, where heat is transferred to the auxiliary cooling water system (CACWS) for rejection from cooling towers to the atmosphere. The component and systems described for the NSSS are shown in an isometric view of the PCRV in Fig. 1. Figure 2 shows a simplified schematic diagram of the primary and secondary coolant systems. Table 1 shows expected performance parameters.

The PCRV and ancillary systems are housed inside a reactor containment building which is a conventional steel-lined reinforced secondary

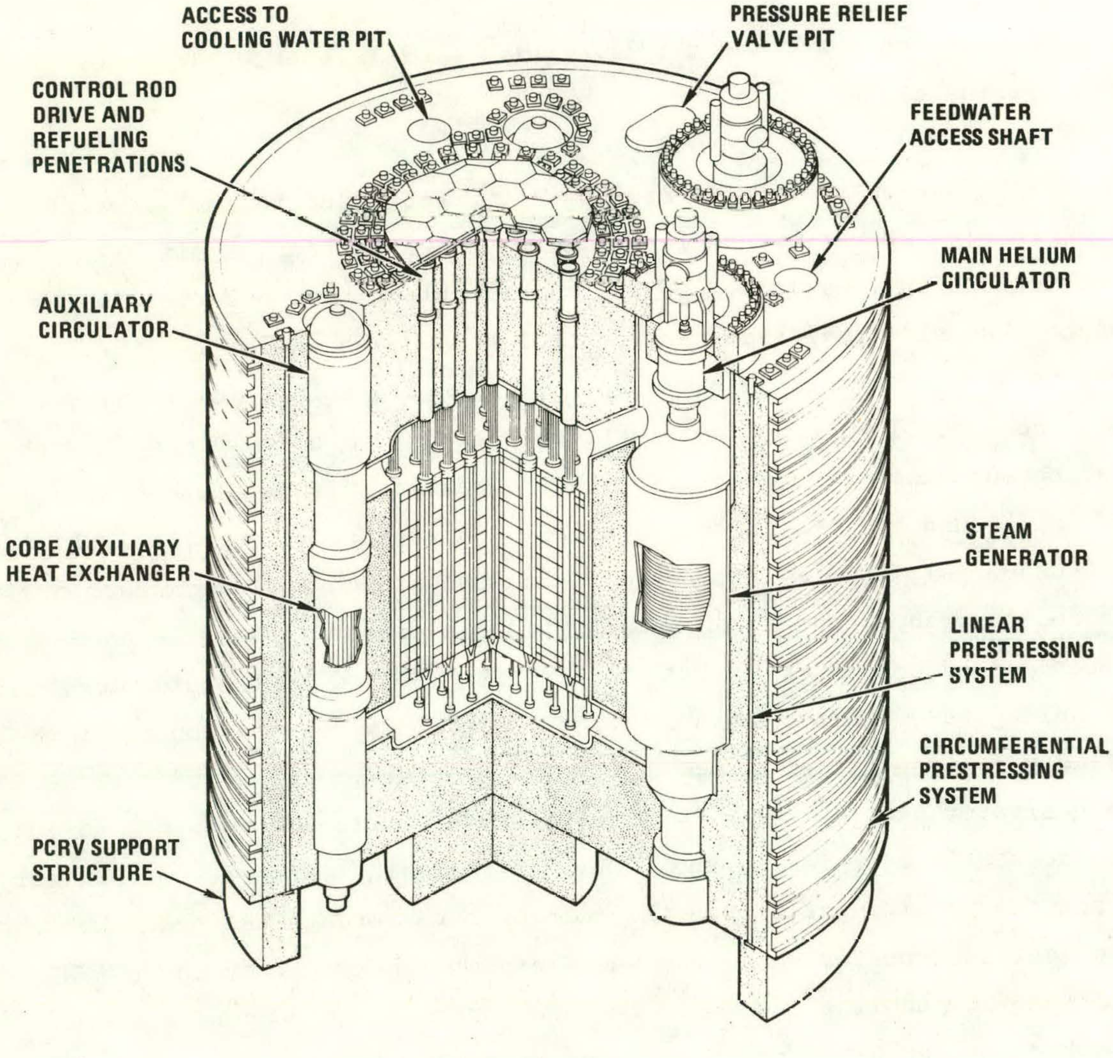

Fig. 1. HTGR nuclear steam supply system

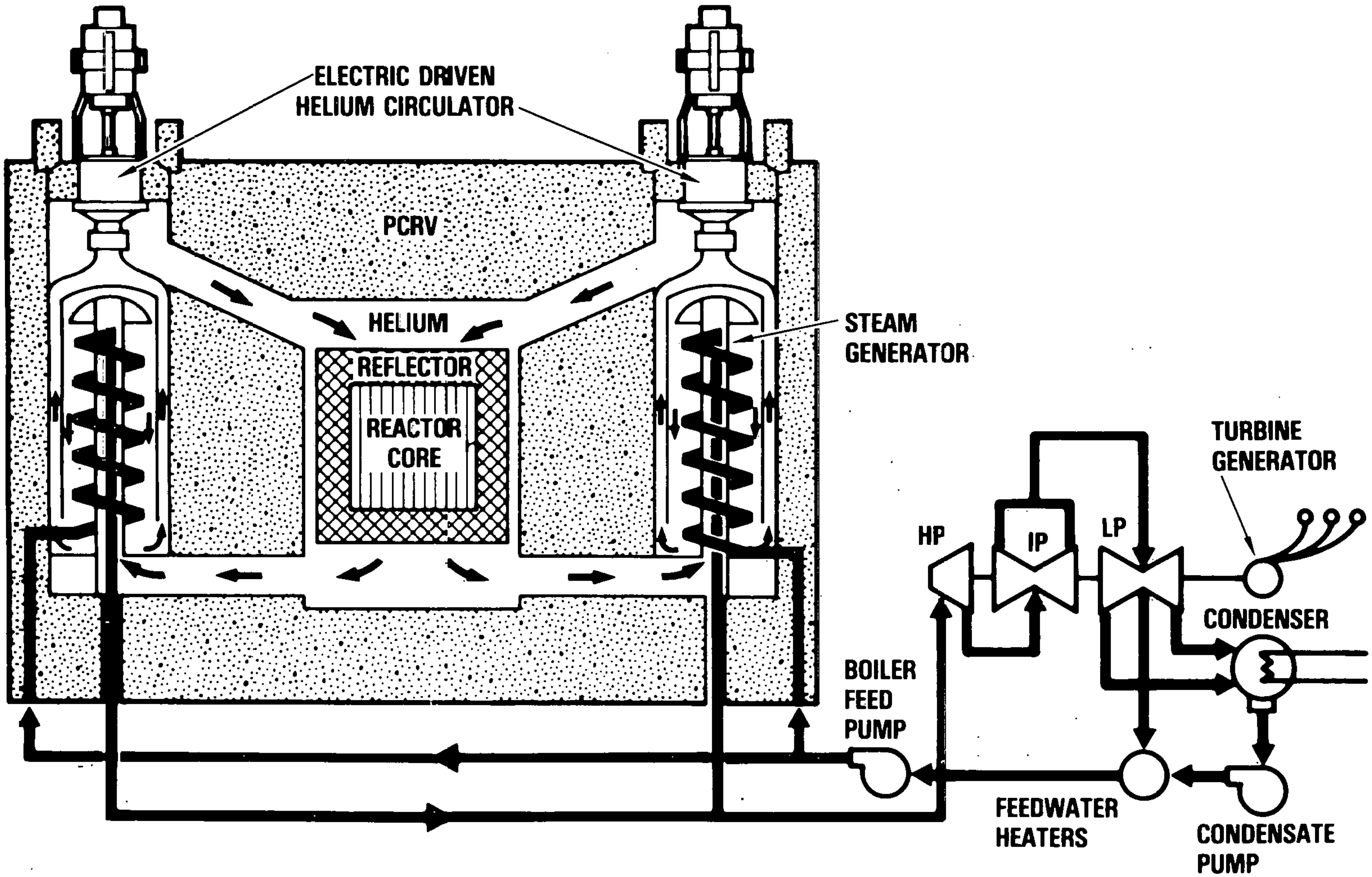

Fig. 2. Schematic flow diagram

TABLE 1
1170-MW(t) HTGR EXPECTED PERFORMANCE PARAMETERS

No. of primary coolant loops	2
Reactor power	1170 MW(t)
Nominal electrical output (net)	448 MW(e)
Helium inventory, total	7865 kg (17,340 lb)
Helium inventory, circulating	5792 kg (12,700 lb)
Helium flow rate	615.2 kg/s (4,886,500 lb/h)
Helium pressure at circulator discharge	7.2 MPa (1050 psia)
Total primary circuit pressure drop	127.5 kPa (18.5 psi)
Core inlet gas temperature	316°C (611°F)
Steam generator inlet gas temperature	686°C (1266°F)
Circulator inlet gas temperature	316°C (601°F)
Core power density	7.15 MW/m^3
Feedwater inlet temperature	221°C (430°F)
Superheater exit temperature	541°C (1005°F)
Feedwater inlet pressure	20.4 MPa (2959 psia)
Superheater exit pressure	17.3 MPa (2515 psia)

containment structure. Typically, balance-of-plant (BOP) systems and equipment are arranged and housed in separate buildings according to function and service. Ten years of spent fuel storage with railroad access for shipping and receiving is provided on site.

3.1. NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

The NSSS for the 1170-MW(t) HTGR plant includes those nuclear, control, and heat transfer systems and components used to generate steam for electric power generation. Significant features and characteristics of major system categories are described below.

3.1.1. PCRV and Reactor Internals

The PCRV consists of cavity liners, penetrations, and closures; a thermal barrier on the gas-side surfaces of the liner; and two independent pressure-relief trains. It functions as the primary containment for the reactor core, the primary coolant system, and portions of the secondary coolant system. It also provides the necessary biological shielding and minimizes heat loss from the primary coolant system. The prestressed-concrete portion of the PCRV and those portions of the penetrations unbacked by concrete, including their closures, form the primary coolant pressure-resisting boundary. The cavity and penetration liners, including the closures, form the continuous gas-tight boundary of the PCRV. Penetration and closures also restrict the leakage-flow area from the vessel to acceptable limits in the event of postulated failures. Liner and penetration anchors transmit loads from internal equipment support structures to the PCRV concrete. The PCRV core cavity, offset from the PCRV center, is surrounded by two steam generators and three core auxiliary heat exchanger (CAHE) cavities. Prestressing is provided longitudinally by vertical tendons and circumferentially by wire strands wound in channels in the outer wall of the PCRV. The PCRV diameter is 26.6 m (87.5 ft), and the height is 28.4 m (93 ft). A continuously welded steel liner provides a gas-tight primary coolant boundary covering the surfaces of the cavities, the communicating ducts, and access openings inboard of any penetration or closure anchorage system. Typically, the thermal barrier consists of layers of fibrous blanket insulation compressed against the liner by metal coverplates and seal sheets. Two independent pressure relief trains are provided, each sized for 100% relieving capacity for ultimate protection to limit PCRV maximum cavity pressure (MCP) to 7892 kPa (1130 psig). Instrumentation is also provided to sense, record, and alarm as required on liner and concrete temperature and strain data, including linear and circumferential prestressing loads.

The reactor internals consist of all the graphite components of the core-support floor, the permanent side reflector, and the core peripheral seal; the metal peripheral-seal support structure, including those items that attach the structure to the PCRV liner and others providing the interface with adjacent thermal barrier; the metal core-lateral-restraint and side-shield assemblies; and the metal plenum elements fitting over the top permanent-side-reflector blocks.

3.1.2. Reactor Core

The reactor core includes the fuel elements, the hexagonal reflector elements, the top layer/plenum elements, and the startup neutron sources. The fuel element is a graphite block that contains the fuel and acts as a moderator. Each fuel element consists of a hexagonal graphite block containing drilled coolant passages and fuel channels into which the fuel rods are inserted.

The individual fuel rods contain the fissile and fertile coated particles distributed in a graphite matrix. The initial core elements and the reload elements, whether containing fresh or recycle fuel, are of identical geometry. The fissile particle has a uranium carbide kernel with a TRISO coating. The TRISO coating has four layers: (1) an inner buffer layer of low-density pyrolytic carbon, (2) a thinner layer of pyrolytic carbon, (3) a layer of silicon carbon, which provides containment of gaseous and solid fission products, and (4) an outer layer of high-density pyrolytic carbon, which adds strength to the coating. The fertile particle has a thorium oxide kernel with a BISO coating. The BISO coating has two layers: (1) an inner buffer layer of low-density pyrolytic carbon and (2) an outer coating of high-density pyrolytic carbon which provides the containment.

The reference fuel cycle uses 20% enriched uranium/thorium and is currently optimized for no recycle. The ultimate goal, however, is to use a high-enriched uranium/thorium fuel with full recycle. Moreover, the plant, core, and fuel designs are such that flexibility in the fuel cycle design is retained to ensure that fuel recycle, higher uranium enrichment, or both may be adopted in the future. Depending on the fuel cycle being applied, the conversion ratio for the HTGR may vary from 0.6 to 0.92.

The fuel elements and hexagonal reflector elements are arranged in columns supported on core-support blocks, with each support block normally corresponding to one fuel region. Each region consists of seven columns of fuel elements, with a central column of control fuel elements and six surrounding columns of standard fuel elements. These fuel regions are surrounded by two rows of hexagonal reflector-element columns, which in turn, are surrounded by the permanent side reflector. The reflector elements may have coolant holes, control-rod and reserve shutdown holes, and shielding materials as required, but they do not contain fuel. The reactor core also contains top layer/plenum elements and startup neutron sources. The top layer/plenum elements are hexagonal alloy-steel components that provide the flow plenums for distributing the flow from the region flow-control valves to the individual columns, lateral restraint during refueling, and support for the flow-control valve and lower guide-tube assembly.

3.1.3. Primary Coolant System

The primary coolant system consists of the subsystems and components required to transfer heat from the reactor core to the secondary coolant system. The primary coolant system uses a constant inventory of helium to transfer heat from the reactor core to the system generators. The system utilizes two steam-generator modules in series with two helium circulators situated in cavities within the PCRV. The primary-coolant helium is forced downward through the reactor core by the two helium circulators, which derive their power from coaxial synchronous electric motors. The helium leaves the core through the core-support blocks, traverses the lower plenum, enters the two steam-generator crossducts, flows upward over the steam-generator surfaces, and enters the circulator inlet to complete the circuit. The entire system is contained within the PCRV.

Helium temperatures are measured at each core-support block exit and at the steam generator exit. These temperatures are controlled by adjusting the flow control valves or control-rod configuration. The flow control valves, located at the plenum element above the core, is manually adjusted for core region flow. Steam temperature is used for automatic regulation of the control rods.

3.1.4. Core Auxiliary Cooling System (CACS)

The CACS is an engineered safety system incorporated in the HTGR design for reactor core residual and decay heat removal. The system, installed in the PCRV, consists of three auxiliary primary coolant loops, each having a variable-speed electric-induction-motor-driven auxiliary circulator, an auxiliary shutoff valve, and a water-cooled heat exchanger. The CACS function is to provide a separate independent means of cooling the reactor core with the primary system pressurized or depressurized. Each loop is capable of cooling the core following loss of main primary loop circulation and reactor trip from full-power conditions with the PCRV pressurized. Any two loops can cool the core under the same conditions with the PCRV at containment building atmospheric pressure. The CACS is maintained in a standby mode when the main loops are in operation. This ensures system readiness.

3.1.5. Fuel Handling System

The fuel handling system consists of all the equipment and subsystems required for the remote fuel and reflector element handling. Major system equipment items are a fuel handling machine, fuel transfer casks, an auxiliary service cask, refueling equipment positioners, fuel transfer casks dollies, a refueling equipment transporter, reactor isolation and floor valves, fuel container loading equipment, a control station, and a fuel sealing and inspection facility (FSIF). This system handles both new and used fuel from an in-core location to the fuel storage facility.

Refueling operations are based on a 4-yr core life; one-quarter of the reactor core is replaced with new fuel each year; replaceable reflector elements that reside adjacent to active fuel elements are replaced at 8-yr intervals. Both fuel and reflector elements are transferred through refueling penetrations in the top head of the PCRV. With a few exceptions, these penetrations contain control rod drives which must be removed before the fuel handling machine is installed on the refueling floor. Each refueling region, normally consisting of seven columns of fuel and removable reflector elements, is entirely emptied of spent fuel before new fuel is placed. Spent fuel and reflector elements are transferred to the fuel container loading facility, transferred to the FSIF, sealed in helium-filled containers, and subsequently placed in storage.

3.1.6. Auxiliary Systems

The NSSS auxiliary systems are the reactor plant control, protection, and service systems. Functionally, they include the following:

1. Main and auxiliary circulator services provide lubrication, cooling, and buffer helium to the circulators.

2. Helium services provide purified primary coolant helium as a purge gas and static pressurization for sealing PCRV penetration closures.

3. Plant protection system (PPS) provides for safety operation or shutdown in the event of an abnormal or accident condition.

4. Plant control system (PCS) provides for safe automatic plant operation, by regulating reactor power and controlling NSSS steam conditions, and for automatic action to protect major plant components.

5. Plant data acquisition processing and display system (DAP) provides and records operating information.

3.2. BALANCE OF PLANT (BOP)

Structures, equipment, and systems not part of the NSSS are identified as balance of plant (BOP). For design and accounting purposes, the BOP is typically broken down into about six major categories: (1) structures and improvements, (2) turbine plant equipment, (3) electric plant equipment, (4) miscellaneous plant equipment, (5) waste heat rejection system, and (6) reactor plant BOP systems. Much of the turbine and electric plant is based on current subcritical power plant design practice. Significant features unique to an HTGR BOP are briefly described below and shown in Fig. 3.

3.2.1. Major Systems

The turbine plant design is based on a single tandem-compound, four or six flow turbine generator with no external moisture separation or reheat.

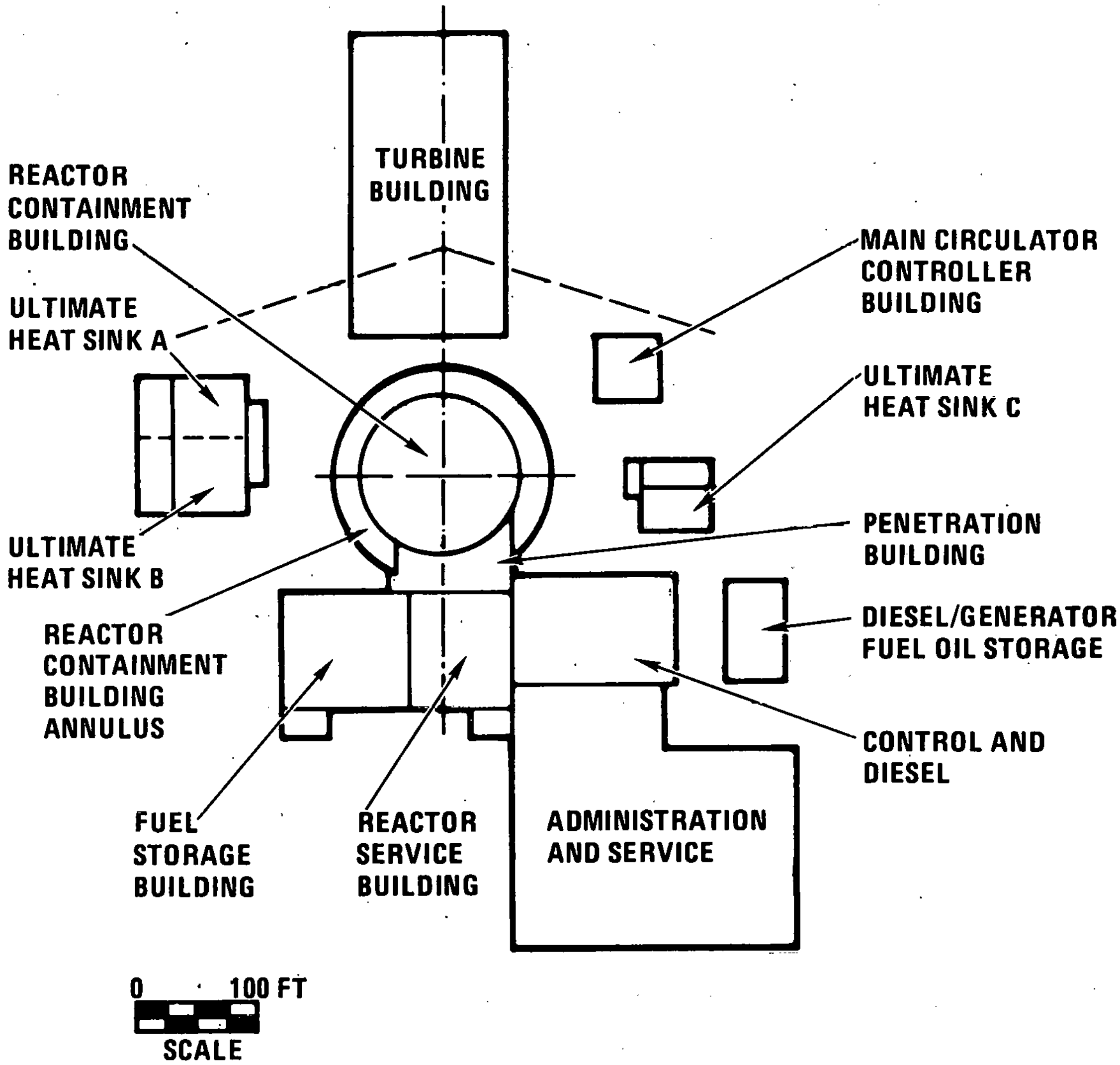

Fig. 3. 1170-MW(t) BOP

The turbine-generator converts 1180-MW(t) steam generator thermal output [1170 MW(t) core thermal output] to 480-MW(e) gross electrical output for a net station output of 448 MW(e). The turbine plant includes a full-flow condensate polishing system, six stages of feedwater heating, and two half-size turbine-driven boiler feed pumps. Main steam lines from each of the two steam generators penetrate the containment and are headered in the turbine building. For startup, shutdown, and other conditions of off-normal operation, a main steam bypass to condenser is provided. A closed cooling water system is provided to remove waste heat from all turbine plant components. This system is cooled by service water from the waste heat rejection system. The heat sink for the main thermal cycle and all plant service water during normal plant operation is assumed to be a conventional wet cooling tower for the reference plant design. Alternate waste heat rejection systems may be considered, depending on specific site conditions and resources.

The electric power plant control systems are similar to fossil plants. The electrical power system provides power to the NSSS loads in the event of a loss of off-site power. A safety-class nuclear service water system supplies cooling water for the essential reactor plant cooling water system, fuel handling and storage cooling water systems, and other reactor plant auxiliaries during emergency conditions. The system consists of two independent redundant trains that reject heat to separate auxiliary wet cooling towers. The CACWS provides a closed loop supply of cooling water to the CAHEs so that reactor decay heat is removed from the primary coolant and rejected to the atmosphere by air blast heat exchangers. Each of the three independent cooling water loops is normally in a standby mode and only activated upon loss of main loop cooling capability.

Plant buildings, enclosures, etc., are either Seismic Category I or non-Seismic Category I structures. Seismic Category I structures house all safety-related systems and equipment essential for safe plant operation, shutdown, and control. These are generally massive reinforced concrete structures. The reactor containment building (RCB) is a steel-lined, reinforced concrete cylinder with a hemispherical dome and circular base mat. The reactor service, fuel storage, and the control auxiliary and diesel generator buildings are major structures adjacent to the RCB for functional arrangement of operation and service to the NSSS. Ten-year fuel storage is provided on-site, and the facility can ship spent fuel by either truck or rail.

4. SAFETY CHARACTERISTICS

4.1. INHERENT SAFETY

The inherent safety features which make gas-cooled reactors of low public risk are listed in Table 2 and described below.

4.1.1. Helium Gas Coolant

A noncondensable gas totally occupies its space, and so confined, obeys a simple linear temperature-pressure relationship. Because no liquid-gas interface need be considered, unambiguous measurements of temperature and pressure indicate the coolant state and location.

A loss of coolant cannot occur. Depressurization can occur, but it is accommodated without degradation of fuel cooling capability, since no phase change occurs. Adequate core cooling is possible at atmospheric pressure.

Helium coolant is chemically and neutronically inert. It cannot react with core components, and it does not contribute to or affect the nuclear chain reaction.

4.1.2. Ceramic Core and Reflector

The 1.4×10^6 kg (3×10^6 lb) core and reflector structure is composed of graphite, a material which does not melt [sublimes at ~3800°C (6872°F)] and which retains good strength to above 2500°C (4532°F). The associated heat capacity, high temperature capability, and low power density of graphite insure that reactor temperature transients will proceed very slowly.

TABLE 2
SAFETY SIGNIFICANCE OF KEY INHERENT FEATURES

Inherent or Passive Feature	Relevant Properties	Safety Significance
Helium coolant	Single phase	No boiling, bubbles, liquid level, or pump cavitation. Coolant injection system not required. No ambiguity of signal indicating coolants presence.
	Neutronically inert	No reactivity effects.
	Chemically inert	No fuel/helium chemical interactions.
Graphite core	High heat capacity, low power density	Slow transient response. Time for prevention and mitigation of accidents.
	Graphite cannot melt but may locally sublime	Strength maintained to >3000°C (>5432°F).
Coated particle fuel form	Ceramic material	Maintains integrity at very high temperature.
	Multiple pressure vessels	Slow controlled release of volatile nuclides under no cooling conditions.
PCRV and associated liner	Multiplicity of tendons	Failure of individual structural members inconsequential.
	Tendons shielded	No change in properties.
	Tendons removable	In-service inspection possible.
	Integral arrangement	Primary system pipe/duct ruptures eliminated. Multiple structural failure required for air ingress.

The slow thermal response provides a forgiving reactor, since the behavior of the system is more readily predictable, and more time is available to prevent transients from progressing into major accidents. Time is available for equipment repair, system adjustment, or other corrective action. For example, 30 min interruptions in core cooling system operation can be tolerated before damage to the core flow orifices and control rods occurs.

4.1.3. Coated Particle Fuel

Another safety concern is the possible migration of fission products. The fuel particle coatings constitute tiny independent pressure vessels which contain the fission products. A total interruption of the core cooling systems would have to continue for ~3 h before any fuel damage would occur and ~20 h before 50% of the core radioactivity would be released, providing time for fission product decay and for mitigating operator actions.

4.1.4. PCRV

PCRV containment of the entire primary system is facilitated by a noncondensable coolant. The PCRV safety advantages derive primarily from the independence and redundancy of the load-bearing steel tendons, which provide a barrier to fault propagation within the vessel. These tendons are shielded by the concrete from irradiation effects. The steel liner functions as a nonload-bearing seal, which is always held in compression by the surrounding prestressed concrete. This design feature greatly limits the possibilities of fault propagation in the liner. The liner cooling arrangements also furnish an additional heat sink.

As summarized in Table 2, the gas-cooled safety accrues from the following:

1. Helium coolant.

2. Massive high-temperature (ceramic) materials for the fuel, moderator, and reflector, which give the core a high heat capacity.

3. Coated fuel particles, which act as miniature pressure vessels.

4. PCRV containment of the entire primary system.

4.2. ENGINEERED SAFETY

Several design safety features are discussed below.

4.2.1. Dedicated Forced-Circulation Decay-Heat Removal System

The normal mode of core cooling involves main loop circulators, steam generators, and associated systems. HTGRs also incorporate a dedicated decay-heat removal system, separate and independent of the main power conversion system, with a sole function of decay heat removal. This separation of function increases resistance to common-mode failures. The decay-heat removal system, which is simple and reliable, consists of three auxiliary cooling loops, each containing a motor-driven auxiliary circulator, a helium-to-water heat exchanger, and a water-to-air heat exchanger for ultimate heat rejection. Coolant need not be injected into the core following a primary system depressurization, because adequate heat removal is obtained with helium at the containment equilibration pressure.

4.2.2. Liner Cooling Ultimate Heat Sink

The 1170-MW(t) HTGR is small enough that heat transfer from the core to the cooled liner during a permanent and complete loss of all coolant flow is enough to limit peak core temperatures to below the graphite sublimation temperature. This ultimate heat sink is effective even when the reactor is depressurized. Heat removal from the liner is handled by a liner cooling system (LCS), but in an emergency, cooling water to the system could, in principle, be supplied by something as simple as a fire hose connection outside the containment. Although damage to the core and primary system might occur, the PCRV and secondary containment would not be breached; hence, the surrounding population would be protected.

4.2.3. Other Engineered Features

Engineered features are also used to minimize the risks that evolve when air or water come into contact with hot graphite. Although no chemical reactions can occur with helium, graphite would oxidize with any air or moisture present in the primary coolant as impurities or from an accident.

The potential for graphite oxidation by air is minimized by using an integral PCRV arrangement to encompass the entire primary coolant system. To obtain significant quantities of air ingress, very large openings in the PCRV or multiple openings at different elevations must be postulated. Both premises require multiple structural failures, events which are extremely unlikely. Furthermore, the limited amount of air available within the containment could oxidize only a very small fraction of the core/reflector graphite.

In addition to air, there are sources of water available for ingress into the core cavity. However, unlike the air reaction, the water-graphite reaction is endothermic and, therefore, inherently self-limiting. Engineered safety features include a steam generator moisture detection and an isolation and dump system to detect water in the primary coolant system to take corrective actions to limit ingress. Additionally, core heat removal systems are designed to cool the core to temperatures where the reaction potential is negligible.

4.3. ENHANCED SAFETY AND DESIGN FEATURES

The superior safety characteristics of the HTGR, due primarily to its inherent features, have long been recognized. Recently, probabilistic risk assessment has provided a systematic, disciplined approach for quantifying the HTGR public risk, as documented in the Accident Initiation and Progression Analysis (AIPA) study. Comparison with the corresponding assessment for light water reactors (LWRs) (Ref. 1) has clearly indicated significant safety differences between HTGRs and LWRs. The Three Mile Island incident has again focused attention on reactor safety, and LWR safety improvements can be expected. The HTGR also has the potential for improved safety, and to that end, a study is under way to utilize probabilistic risk assessment to quantitatively rank the important design modifications that will reduce public risk. The objective is to produce a quantitative ranking which will provide input for design optimization.

Although the enhanced safety study is not yet completed, several promising design modifications under consideration for the 1170-MW(t) plant can be indicated at this time.

1. Improve feedwater pump turbines. This is a relatively simple and inexpensive design change which provides time for cooling recovery from main loop rundown.

2. Natural convection decay heat removal. The capability to remove heat by natural convection through one or more CACS offers the potential of not only reducing public risk, but also reducing utility investment risk. Its importance was stressed by the Three Mile Island incident.

3. Intentional PCRV blowdown following a loss of cooling. This feature of the Fort St. Vrain plant would minimize activity release from the PCRV during a core heatup.

The capital costs of these features are small; developmental and licensing costs still need to be considered.

To implement these suggested modifications would require the following changes:

Modification	Change
Improve feedwater pump turbines	Install dual-pressure turbine and ducting and valves for atmospheric exhaust or any electric-driven feed pump.
Natural convection decay heat removal	Add a dedicated natural convection primary loop with remote manual valves and heat exchanger.
PCRV blowdown	Install a helium purification system able to withstand the high temperatures experienced during the first hours of a core heatup.

According to preliminary estimates, the enhanced features would substantially improve core heat removal reliability.

5. CONCLUSION

This report described the design of a medium-size HTGR that generates 450 MW(e) and which uses the experience gained in the design, construction, and operation of the 40-MW(e) Peach Bottom Nuclear Generating Station and the 330-MW(e) Fort St. Vrain Nuclear Generating Station. The reactor incorporates the inherent safety features of the HTGR concept and the engineered safety systems which increase the safety of the HTGR. In addition, this report presented potential enhanced safety design features which could be incorporated into the design of the medium-size HTGR.

6. REFERENCE

1. "Accident Definition and Use of Event Trees," Appendix I to "Reactor Safety Study," U.S. Nuclear Regulatory Commission Report WASH-1400, Appendix I, October 1975 (NUREG-75/014).

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