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PRELIMINARY SAFETY AND ENVIRONMENTAL INFORMATION DOCUMENT

VOLUME IV

MASTER

HIGH-TEMPERATURE GAS-COOLED REACTORS

January 1980

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM

U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR NUCLEAR ENERGY

DISCLAIMER

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FOREWORD

The U.S. Department of Energy (DOE) Nonproliferation Alternative Systems Assessment Program (NASAP) is a planned program of studies of nuclear power systems, with particular emphasis on identifying and then evaluating alternative nuclear reactor/fuel-cycle systems that have acceptable proliferation-resistance characteristics and that offer practical deployment possibilities domestically and internationally. The NASAP was initiated in 1977, in response to President Carter's April 1977 Nuclear Power Policy Statement.

The NASAP objectives are to (1) identify nuclear systems with high proliferation resistance and commercial potential, (2) identify institutional arrangements to increase proliferation resistance, (3) develop strategies to implement the most promising alternatives, and (4) provide technical support for U.S. participation in the International Nuclear Fuel Cycle Evaluation (INFCE) Program.

NASAP is not an assessment of all future energy-producing alternatives. Rather, it is an attempt to examine comprehensively existing and potentially available nuclear power systems, thus providing a broader basis for selecting among alternative systems. The assessment and evaluation of the most promising reactor/fuel-cycle systems will consider the following factors: (1) proliferation resistance, (2) resource utilization, (3) economics, (4) technical status and development needs, (5) commercial feasibility and deployment, and (6) environmental impacts, safety, and licensing.

The DOE is coordinating the NASAP activities with the U.S. Nuclear Regulatory Commission (NRC) to ensure that its views are adequately considered at an early stage of the planning. In particular, the NRC is being asked to review and identify licensing issues on systems under serious consideration for future research, development and demonstration. The Preliminary Safety and Environmental Information Document (PSEID) is the vehicle by which NASAP will provide information to the NRC for its independent assessment. The PSEID contains the safety and environmental assessments of the principal systems. Special safeguards measures will be considered for fuel cycles that use uranium enriched in U-235 to 20% or more, uranium containing U-233 in concentrations of 12% or more, or plutonium. These measures will include the addition of radioactivity to the fuel materials (i.e., spiking), the use of radioactive sleeves in the fresh-fuel shipping casks, and other measures. The basis for the safeguards review by the NRC is contained in Appendix A.

The information contained in this PSEID is an overlay of the present safety, environmental, and licensing efforts currently being prepared as part of the NASAP. It is based on new material generated within the NASAP and other reference material to the extent that it exists. The intent of this assessment is to discern and highlight on a consistent basis any safety or environmental issues of the alternative systems that are different from a reference LWR once-through case and that may affect their licensing. When issues exist, this document briefly describes research, development, and demonstration requirements that would help resolve them within the normal engineering development of a reactor/fuel-cycle system.

The preparation of this document takes into consideration the NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made to this document. Additional comments and responses were incorporated as Appendix B. Comments and requests for information that are beyond the scope and resources of the NASAP may be addressed in the research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

This PSEID was prepared for the DOE through the cooperative efforts of the Argonne National Laboratory, the Oak Ridge National Laboratory, and NUS Corporation.

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Chapter 1

GENERAL DESCRIPTION

1.1 INTRODUCTION

The high-temperature gas-cooled reactor (HTGR) plant selected for this study corresponds in design to the lead plant (Ref. 1), except for the fuel. The layout is shown in Figure 1-1.

Medium-enriched uranium (MEU) fuel is utilized, rather than highly enriched uranium (HEU). Two cycles are considered: a "throwaway" once-through cycle and a uranium-233 recycle with denaturing in situ and external makeup.

The design is a reoptimized and uprated version of the General Atomic Company's standard commercial plant, for which a standard safety analysis report (Ref. 2) was prepared and submitted to the U.S Nuclear Regulatory Commission (NRC). Modifications to the "standard design" are listed in Section 1.1.2.

1.1.1 DESCRIPTION

The reactor core is cooled with helium, moderated and reflected with graphite, and fueled with a mixture of uranium-235 and uranium-233. It is constructed of graphite blocks with vertical holes for coolant, fuel rods, and control rods.

The reactor is contained in a prestressed-concrete pressure vessel (PCRV). Helium coolant flows from six steam-driven circulators through the core, through the six main steam generators (each located in a cavity in the PCRV wall), and back to the circulators.

The superheated steam produced in the steam generators is passed through the high-pressure section of the main turbine and then to the helium-circulator drive turbines. On exit from the circulator turbines, it passes through the reheaters before it enters the intermediate-pressure section of the main turbine. Waste heat is removed from the steam by a water-cooled condenser and rejected through cooling towers to the atmosphere.

The components and systems described above constitute the nuclear steam supply system (NSSS). It is shown as a perspective cutaway in Figure 1-2, in cross section in Figure 1-3, and schematically in Figure 1-4.

In addition, a core auxiliary cooling system (CACS) is provided. It consists of three auxiliary gas/water heat exchangers with electric-motor-driven circulators located in cavities in the PCRV wall. Coolant gas is circulated from the core through the heat exchangers, giving up its heat to the core auxiliary cooling water system (CACWS) for rejection from cooling towers to the atmosphere.

The prestressed-concrete reactor vessel is housed inside a reactor-containment building. The building is a steel-lined, reinforced prestressed concrete cylinder with a hemispherical dome and circular base mat.

The steam-generator and circulator piping is headered outside the containment building and routed to and from the turbine building.

Besides the turbine building, the plant has the following balance-of-plant structures:

1. Reactor service building
2. Fuel storage building
3. Control and diesel-generator building
4. Access-control building
5. Two NSSS cooling towers
6. Nuclear service cooling tower
7. Core auxiliary cooling system water/air heat exchanger

1.1.2 MODIFICATIONS TO THE "STANDARD DESIGN"

In designing the lead plant, some features of the General Atomic Company's standard commercial plant (Ref. 2) were modified, altered, or upgraded. The resulting differences are as follows:

1. Core power density is reduced to approximately 7 W/cm^3, compared with 8.4 W/cm^3 in previous designs. The previously identified 5% stretch capability is incorporated into the plant nominal rating. This increased the rated output to 3,360 MWt, consistent with the largest available single turbine. The core is larger because of the greater output and lower power density.
2. Small control rods (power rods) have been added to reduce temperature fluctuations during load changes. This results in reduced temperature criteria for the fuel and the core-cavity components.
3. The core-cavity height has been increased to provide space for better mixing of the core-outlet gas before it impinges on the core-outlet thermocouples, the core-support posts, and the thermal barrier. This results in lowered design-temperature criteria for the internal components and steam generators. Larger margins for fuel-temperature criteria are also achieved, resulting in reduced fission-product release; this should benefit plant maintenance requirements.
4. The steam generator has a radial-flow reheater and a modified upper closure.
5. The core auxiliary cooling system loops are uprated to 100% duty under pressurized conditions.
6. The core auxiliary heat exchanger is redesigned from a helical tube bundle with entry and exit of cooling water at the top to a bayonet-tube design with entry and exit below the PCRV. This economizes on space and makes in-service inspection feasible.
7. All steam and feed pipework is run out of the bottom of the PCRV to avoid complication in the refueling area and pipe-whip problems in the annulus around the PCRV.
8. The shape of the PCRV support is changed from a star to a ring.
9. The primary-coolant loops and the core auxiliary coolant loops in the PCRV are asymmetrically located to separate safety-related and non-safety-related equipment. A saving in piping costs also results.
10. The steam pipes are headered outside the PCRV for better operational flexibility through ability to isolate a single steam generator in the event of a tube rupture.
11. A single-turbine generator is used; for the output planned, this is a significant saving over twin units.

The asymmetrical layout of the steam generators and core auxiliary coolant loops sets the overall layout criteria for the plant. The turbine building is located to minimize piping runs from the six steam generators. The control and diesel building is located to minimize cabling for control and for the core auxiliary coolant loops. The reactor service building is provided with access from both reactor-refueling floors by means of a bridge passing between two core auxiliary coolant loops.

1.1.3 PLANT DESIGN PARAMETERS

The principal parameters of the lead-plant design are as follows:

Type of cooling	Wet cooling tower
Life, years	40
Nominal net station efficiency, %	39.64
Nominal net station output, MWe	1,332
Capacity factor, %	80
Plant layout	Single unit with layout designed to accommodate a second unit
Maximum rate of load change (for changes >10%), % of rated load per minute	5
Maximum step-load change (no less than 2 hours between), % of rated load per minute	10
Load-following capability	Daily cycle with weekend shutdown to 25%
Number of primary-coolant loops	6
Reheat method	Gas/steam
Circulator type	Steam driven
Core auxiliary cooling system	
Reactor pressurized	3 x 100% loops
Reactor depressurized	3 x 50% loops

The key parameters are as follows:

Helium inventory, pounds	26,820
Helium flow rate, lb/hr	13,150,000
Helium pressure at circulator discharge, psia	780
Total primary circuit pressure difference, psi	17
Core inlet temperature, ^{o}F	620
Steam-generator inlet temperature, ^{o}F	1,320
Main steam flow, lb/hr	9,292,000
Steam-generator outlet temperature, ^{o}F	956
Steam-generator outlet pressure, psi	2,526
Reheater steam flow, lb/hr	9,151,000
Reheater outlet temperature, ^{o}F	1,002
Reheater outlet pressure, psi	631
Feedwater temperature, ^{o}F	405

The major dimensions of the NSSS are as follows:

Containment diameter, feet	143.5
Overall PCRV diameter, feet	111.5
Overall PCRV height, feet	89
Core-cavity diameter, inches	522
Core-cavity height, inches	583
Steam-generator diameter, inches	163.5
Core auxiliary heat exchanger diameter, inches	90
Number of control-rod drives	91
Number of fuel columns	661

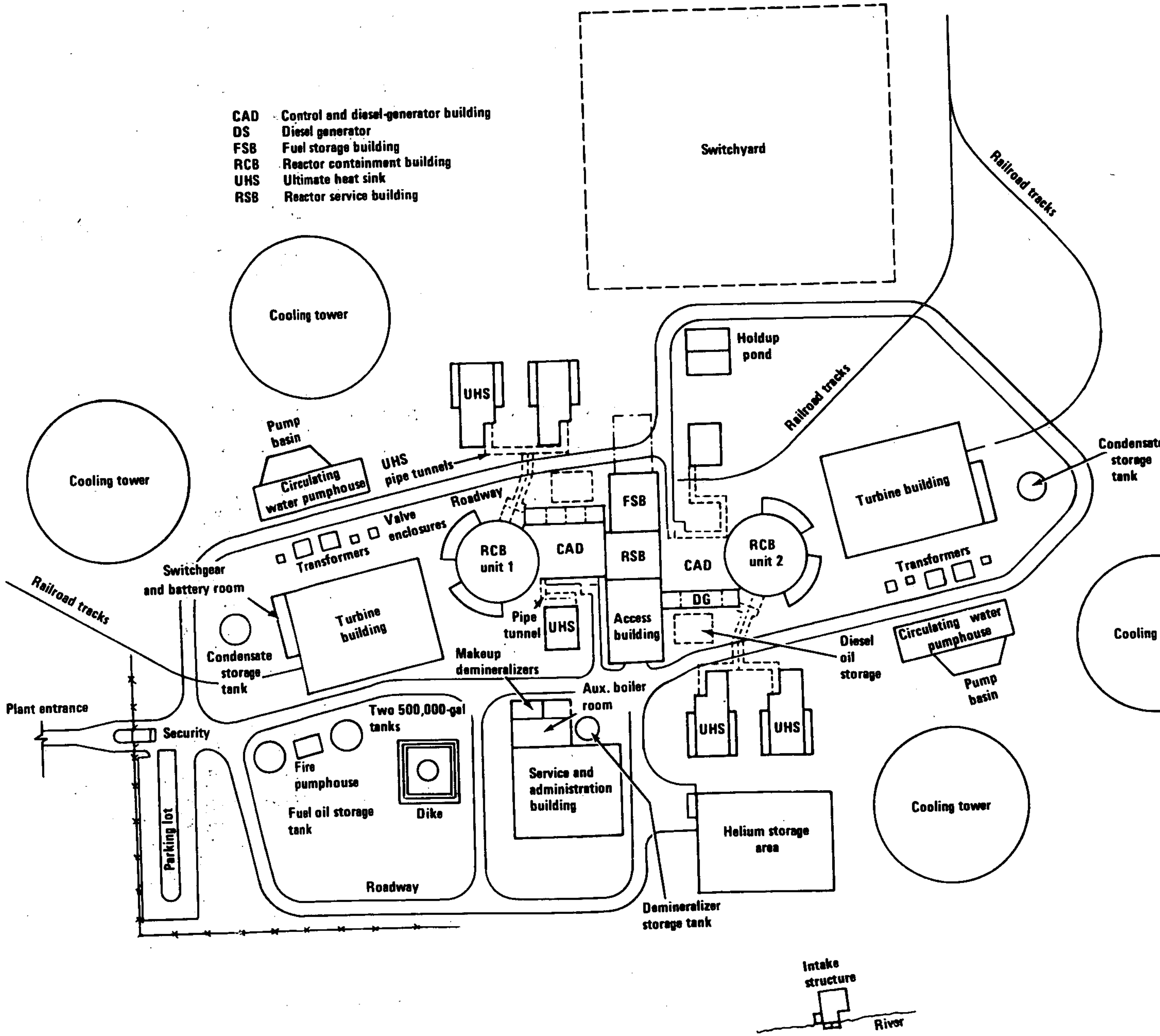

Figure 1-1. Plot plan for the steam-cycle HTGR plant.

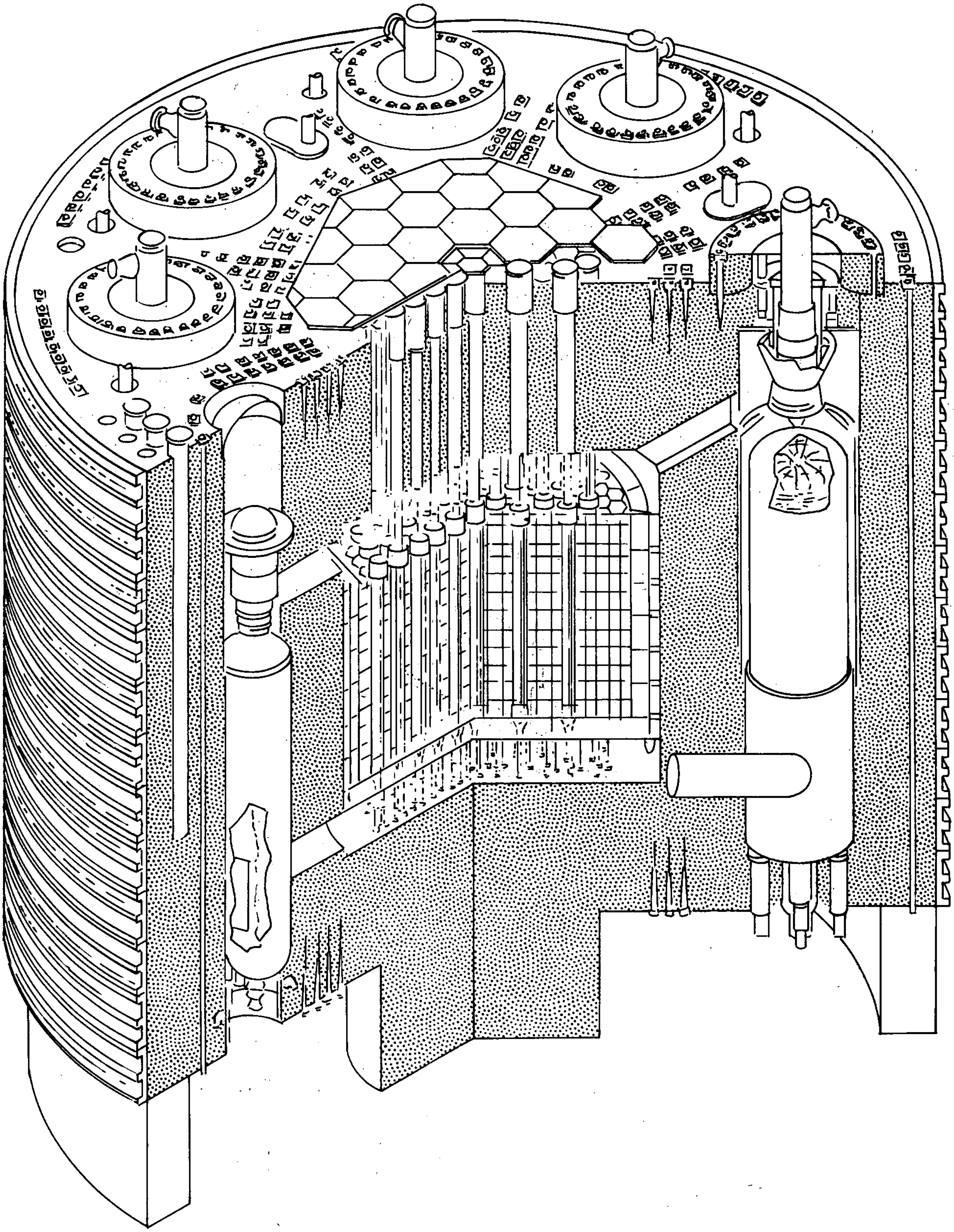

Figure 1-2. Nuclear steam supply system of the lead-plant HTGR.

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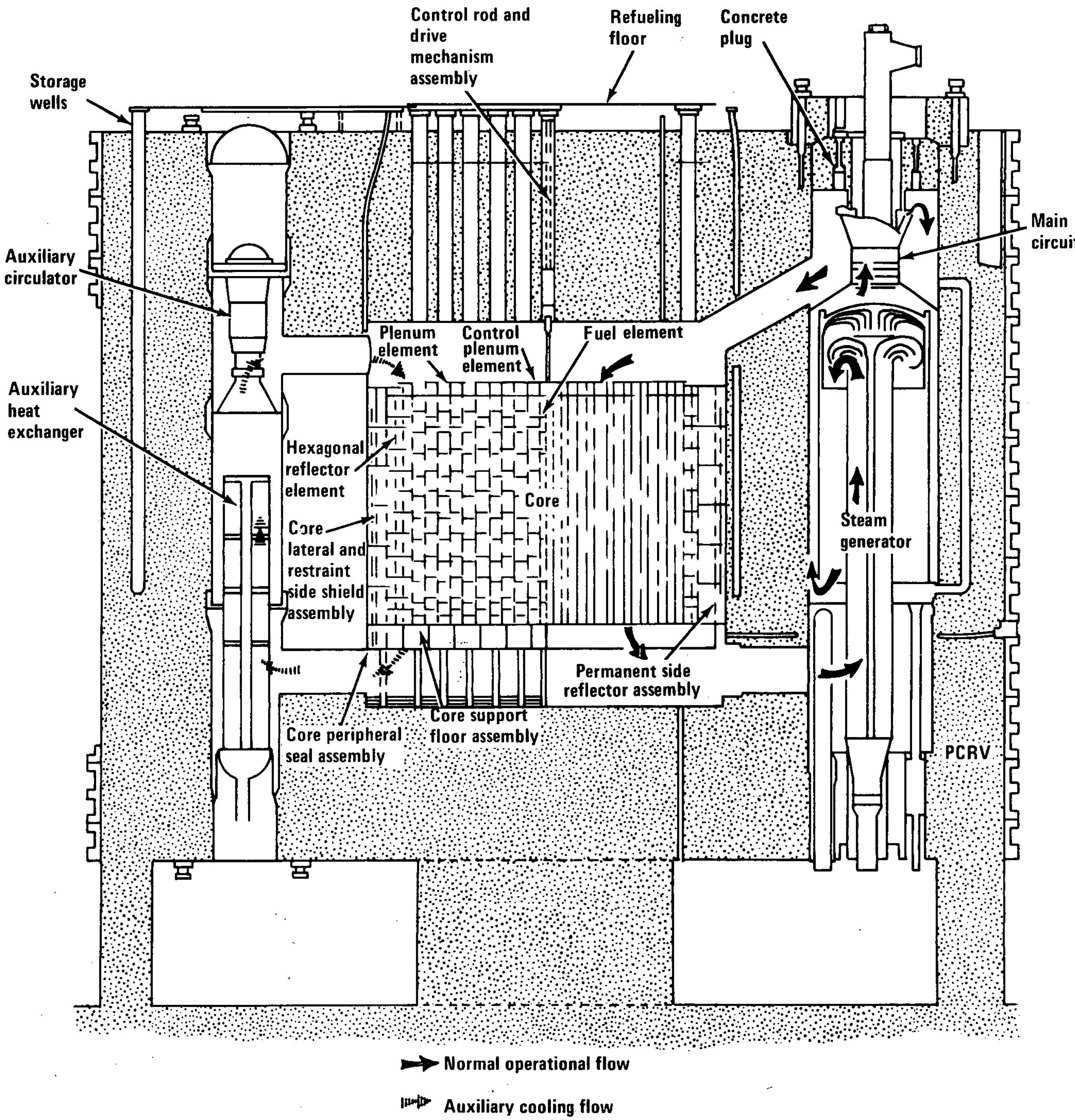


Figure 1-3. HTGR internal components.

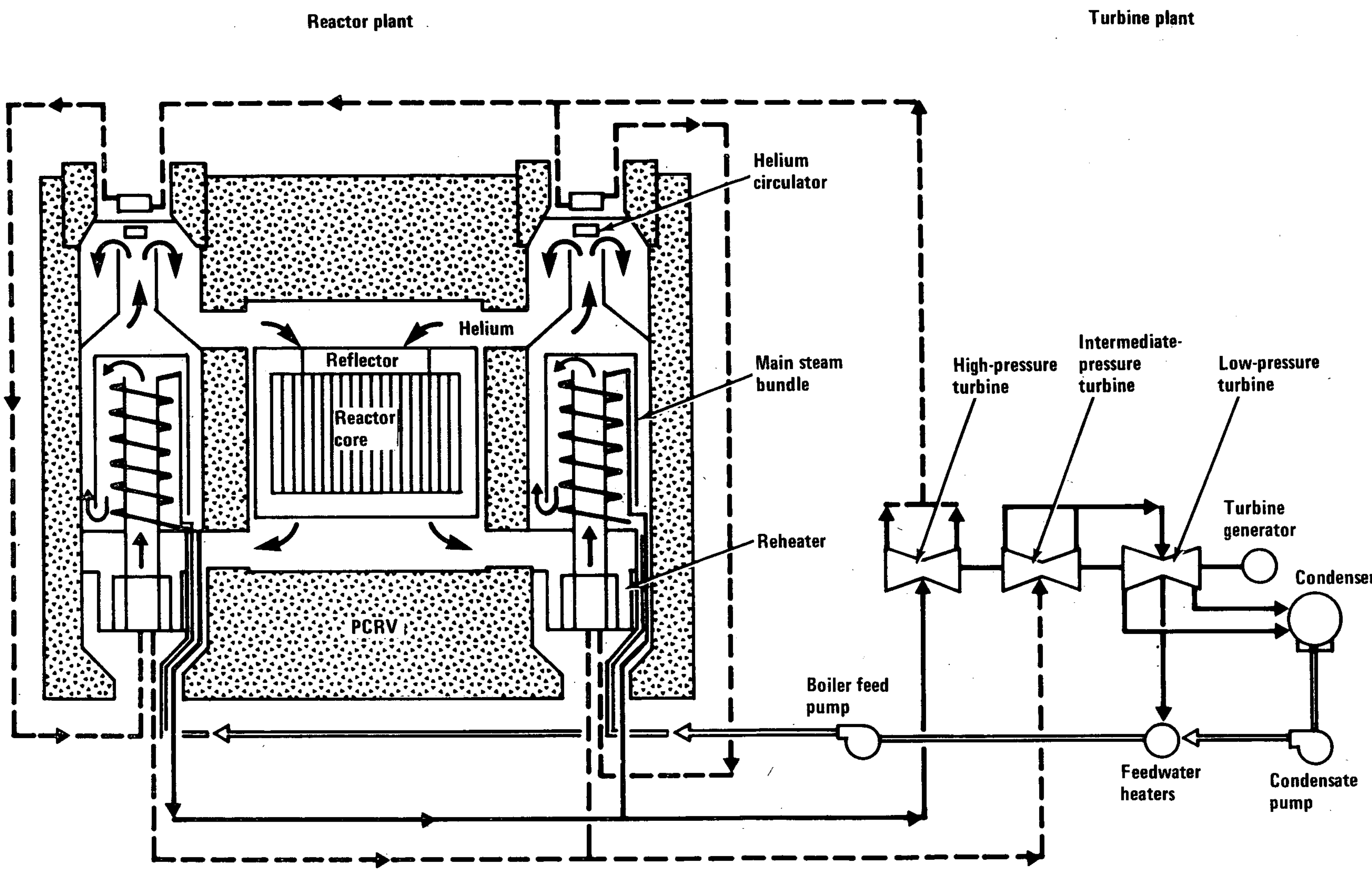


Figure 1-4. Schematic flow diagram of the lead-plant HTGR.

1.2 NUCLEAR STEAM SUPPLY SYSTEM

The PCRV is 111 feet 6 inches in diameter and 89 feet high. It contains multiple cavities: a central core cavity; six primary-coolant loop cavities, each containing a steam generator and a steam-driven helium circulator; and three core-auxiliary-coolant loop cavities, each containing a heat exchanger and a motor-driven circulator.

For design purposes, the NSSS is divided into the following systems:

PCRV
Reactor core
Reactor internals components
Primary coolant
Core auxiliary cooling
Neutron and region flow control
Fuel handling
Fuel shipping
Reactor service equipment and storage wells
Main and auxiliary circulator service
Helium purification
PCRV service
Plant protection
Plant control
Plant data acquisition, processing, and display system
Gaseous waste

Each of the foregoing 16 systems is described in the lead-plant design description for the steam-cycle HTGR (Ref. 1, Chapter 4). If more detail is desired beyond the following summary descriptions, Reference 1 should be examined.

1.2.1 PRESTRESSED-CONCRETE REACTOR VESSEL

The PCRV includes 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 closures, form the continuous gastight boundary of the PCRV. Penetrations 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. During construction, the liners serve as formwork for the concrete.

1.2.2 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 both 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 (Figure 1-5). 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: an inner buffer layer of low-density pyrolytic carbon, a thinner layer of high-density pyrolytic carbon, a layer of silicon carbide that provides containment of gaseous and solid fission products, and an outer layer of high-density pyrolytic carbon that adds strength to the coating.

The fertile particle has a thorium oxide kernel with a BISO coating. The BISO coating has two layers: an inner buffer layer of low-density pyrolytic carbon and an outer coating of high-density pyrolytic carbon. The latter provides the containment.

These fuel elements reside in the core until they are removed and replaced by the fuel-handling machine.

There are two types of fuel element, standard and control (see Figure 1-5). Both contain arrays of fuel and coolant holes, but the control elements also have holes for the insertion of control rods and reserve shutdown material. Approximately one-seventh of the fuel elements are of the control type.

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. The fuel regions are surrounded by two rows of hexagonal reflector-element columns, which are in turn surrounded by the permanent side reflector. The reflector elements may have coolant holes, control-rod and reserve shutdown holes, and shielding material as required, but they do not contain fuel.

In addition, the reactor core contains top layer/plenum elements and startup neutron sources. The former 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 guidetube assembly. The startup neutron source is californium-252, in a suitable container. It is inserted into core fuel elements to provide a source of neutrons of sufficient strength to ensure a safe, controlled approach to reactor criticality. The arrangement of the reactor core is shown in Figure 1-6.

1.2.3 REACTOR INTERNALS COMPONENTS

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.

1.2.4 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 overall system flow is shown in Figure 1-7. The major system components are the steam generator

(Figure 1-8), the main helium circulator (Figure 1-9), and the helium shutoff valves.

The primary coolant system uses a constant inventory of helium to transfer heat from the reactor core to the steam generators. The system utilizes six steam-generator modules in series with six helium circulators situated in cavities within the PCRV. The primary-coolant helium is forced downward through the reactor core by the six helium circulators, which derive their power from coaxial steam turbines driven by a variable supply of cold reheat steam. The helium leaves the core through the core-support blocks, traverses the lower plenum, and enters the six steam-generator crossducts, from where it flows upward over the steam-generator surfaces and enters the circulator inlet diffuser to complete the circuit.

The temperatures of helium and hot-reheat steam are measured at the exit of each core-support block and at the reheater exit, respectively. These temperatures are controlled by adjusting the core-region flow-control valve or control-rod configuration. Reheat-steam temperature is used for automatic regulation of the control rods.

There are various primary-coolant flow paths that allow bypass around the core. These are accounted for in plant performance predictions.

1.2.5 CORE AUXILIARY COOLING SYSTEM (CACS)

This system includes the auxiliary circulators and their drive motors, motor controls, diffusers and valves, the core auxiliary heat exchangers, control instrumentation, and hardware. It provides an independent means of cooling the reactor core with the primary system pressurized or depressurized. It is sized to maintain the temperatures of all components in the PCRV within safe limits.

The CACS consists of three separate and independent cooling loops, each capable of removing 100% of the core residual and decay heat for safe cooldown from 102% of reactor steady-state power level under pressurized conditions. Under depressurized conditions, each loop has the capacity to remove 50% of the core residual and decay heat. This function is accomplished by forced circulation of the primary coolant by the auxiliary circulator. The core-coolant gas is circulated through the auxiliary heat exchanger, where the heat is delivered to the CACWS for rejection to the atmosphere.

1.2.6 NEUTRON AND CORE-REGION FLOW-CONTROL SYSTEM

The neutron and region flow-control system consists of two major subsystems: the neutron-control subsystem and the core-region flow-control subsystem. The neutron-control subsystem consists of (a) the normal control and shutdown system of control-rod pairs, small control rods, and neutron detectors, (b) the reserve shutdown system, and (c) the movable in-core flux-mapping and startup flux detector system. The core-region flow-control subsystem consists of variable orifices and outlet-temperature thermocouples for 91 core regions. Appropriate actuation devices together with position and limit-of-travel sensors, controls, and indicators are included in each of the above subsystems.

The neutron-control subsystem uses out-of-core flux detectors and controllers, together with control rods and/or the reserve shutdown material, to adjust core reactivity as demanded by the plant control system, the plant protection system, or the plant operator. In-core flux mapping and startup flux measurements are also made, using movable detectors in selected core locations.

The core-region flow-control subsystem adjusts the helium flow through regions of the core by incrementally positioning each adjustable core-region inlet orifice valve when commanded by the plant operator. Temperature indications from the core-region outlet thermocouples are utilized by the plant operator to adjust region flow with the flow-control orifices.

1.2.7 FUEL-HANDLING SYSTEM

The fuel-handling system consists of a fuel-handling machine, fuel-transfer casks, an auxiliary service cask, a refueling-equipment transfer dolly, reactor-isolation valves, floor valves, a control station, and the fuel sealing and inspection facility. This system handles both new and used fuel between its in-core location and delivery to the fuel-storage facility.

1.2.8 FUEL-SHIPPING SYSTEM

This system consists of rail equipment designed to transport spent-fuel elements to an offsite storage facility and/or the recycle plant. It is also designed to ship recycle fuel elements from the recycle plant.

The rail shipping system consists of a rail cask, a rail car, and fuel-shipping containers. The rail cask has an inner basket that holds 12 fuel-shipping containers. Each fuel-shipping container holds six spent-fuel elements or five recycle-fuel elements within protective packaging. The cask body and the cask closure are shielded with depleted uranium.

1.2.9 REACTOR SERVICE EQUIPMENT AND STORAGE WELLS

The equipment involved in this system consists of the control-rod-drive storage wells, the reflector storage wells in the PCRV, the circulator-handling equipment, the in-core thermocouple service equipment, core service tools, and service facility tools.

1.2.10 MAIN AND AUXILIARY CIRCULATOR SERVICE SYSTEMS

The main circulator service system provides a supply of high-pressure water for lubricating and cooling the helium-circulator bearings. In addition, the service system supplies purified buffer helium to prevent inleakage of bearing water to the primary coolant system or outleakage of primary coolant, to recover helium dissolved in water drained from the helium circulators, and to supply high-pressure helium to actuate the circulator brakes and static seals.

The auxiliary circulator service system provides a supply of purified buffer helium to prevent inleakage of motor-bearing lubricant to the primary coolant system or leakage of primary coolant into the motor casing, motor cavity, and bearing-oil reservoirs; to remove oil vapor carried over in the purge helium from the circulators; and to remove and replace motor-bearing lubricant.

1.2.11 HELIUM-PURIFICATION SYSTEM

The helium-purification system removes helium from the primary coolant loop and processes it to remove particulates, chemical impurities, and radioactivity, so that the resulting gas can safely be used as a clean gas purge where needed throughout the plant. This system serves as the primary means of controlling the level of long-lived gaseous radioisotopes and chemical impurities in the primary coolant. The normal flow requirements for purified helium from the system are established by the various clean-helium-purge requirements throughout the reactor plant. The helium-purification system also compresses purified helium recycled from the main and auxiliary helium-circulator service systems to be used as purge gas.

1.2.12 PCRV SERVICE SYSTEM

The PCRV service system provides the capability for pressurizing the seal interspaces of selected PCRV penetration closures with dual elastomer seals. This prevents leakage of primary coolant and permits the integrity of these seals to be continuously monitored. The service system also provides clean-helium-purge flow where required.

1.2.13 PLANT-PROTECTION SYSTEM

The plant-protection system prevents any unacceptable releases of radioactivity that could constitute a hazard to the health and safety of the public by initiating actions to protect the fission-product barriers and to limit the release of radioactivity if failures occur in the barriers. The plant-protection system consists of the following subsystems:

1. Reactor trip system
2. Steam-generator isolation and dump system
3. Main loop shutdown system
4. Core auxiliary cooling system initiation system
5. Containment isolation system (CIS)
6. PCRV pressure-relief block valve closure interlock
7. Containment pressure protection
8. Rod-withdrawal interlock
9. Core auxiliary heat exchanger (CAHE) isolation system

1.2.14 PLANT CONTROL SYSTEM

The plant control system (Figure 1-10) is an integrated system that monitors and controls the plant. It includes the overall plant control loops that maintain rated steam conditions during normal operation and systems that protect major components and serve as a first line of protection for incidents that could otherwise result in the need for action by the plant-protection system. The control room consoles and boards are included, as in the non-safety-related analytical instrumentation for the NSSS, consisting of both analytical instrumentation and the associated piping and controls needed for gas sampling, gas conditioning, and related operations.

The plant control system is so designed that the plant operates in a load-following mode in which the reactor and steam generators follow the load established by the turbine generator and its controls.

As already mentioned, the plant control system provides automatic actions to protect major components and protective actions during certain incidents that would require response by the plant protection system. These control actions include those required as a result of failure of an active NSSS component, such as the main circulator.

1.2.15 PLANT DATA ACQUISITION, PROCESSING, AND DISPLAY SYSTEM

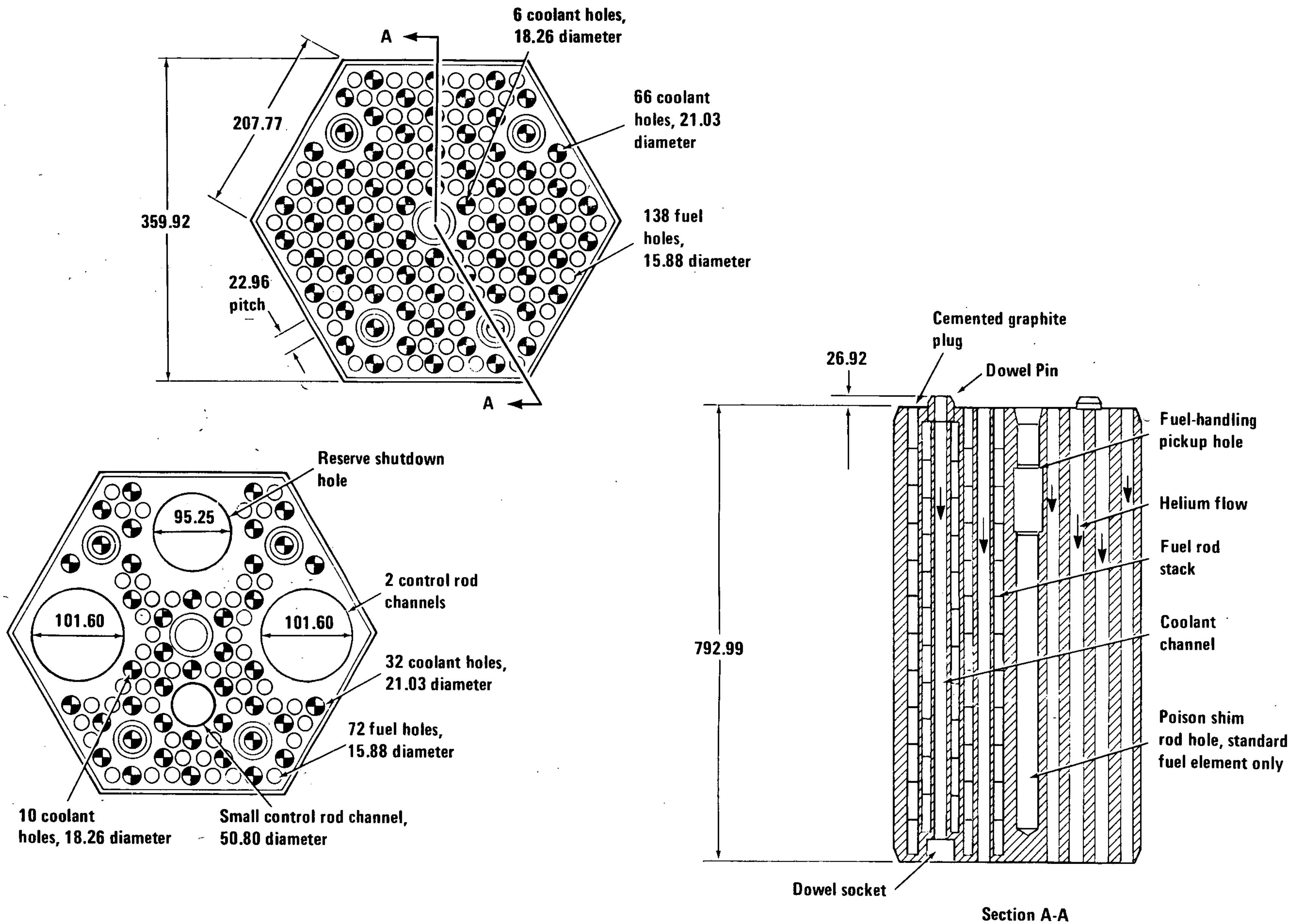
The data acquisition, processing, and display system is a dual computer-based interface between the plant instrumentation and the plant operator. Redundancy of computers and critical peripheral equipment is used for maximum availability.

This system converts certain instrument signals to engineering units, tests for alarm conditions, and provides visual and audible alarms, periodic logs, point trending, sequence-of-events recording, post-trip review, and displays of various operator information and procedural instructions on multicolor cathode-ray tubes. Various applications programs, executed in the system computers to provide operational or plant performance information, can be categorized as follows:

1. Core reactivity status
2. Core temperature and power distribution
3. Heat balance
4. On-line control-rod calibration
5. Plant performance calculations
6. Operator guides

1.2.16 RADIOACTIVE-GAS-WASTE SYSTEM

The radioactive-gas-waste system collects all radioactive and potentially radioactive gaseous wastes generated in the reactor plant, excluding PCRV leakage and other equipment leakage. The system also provides sample collections for radioactivity analysis of the contained gas.



All dimensions mm.

Figure 1-5. HTGR fuel element.

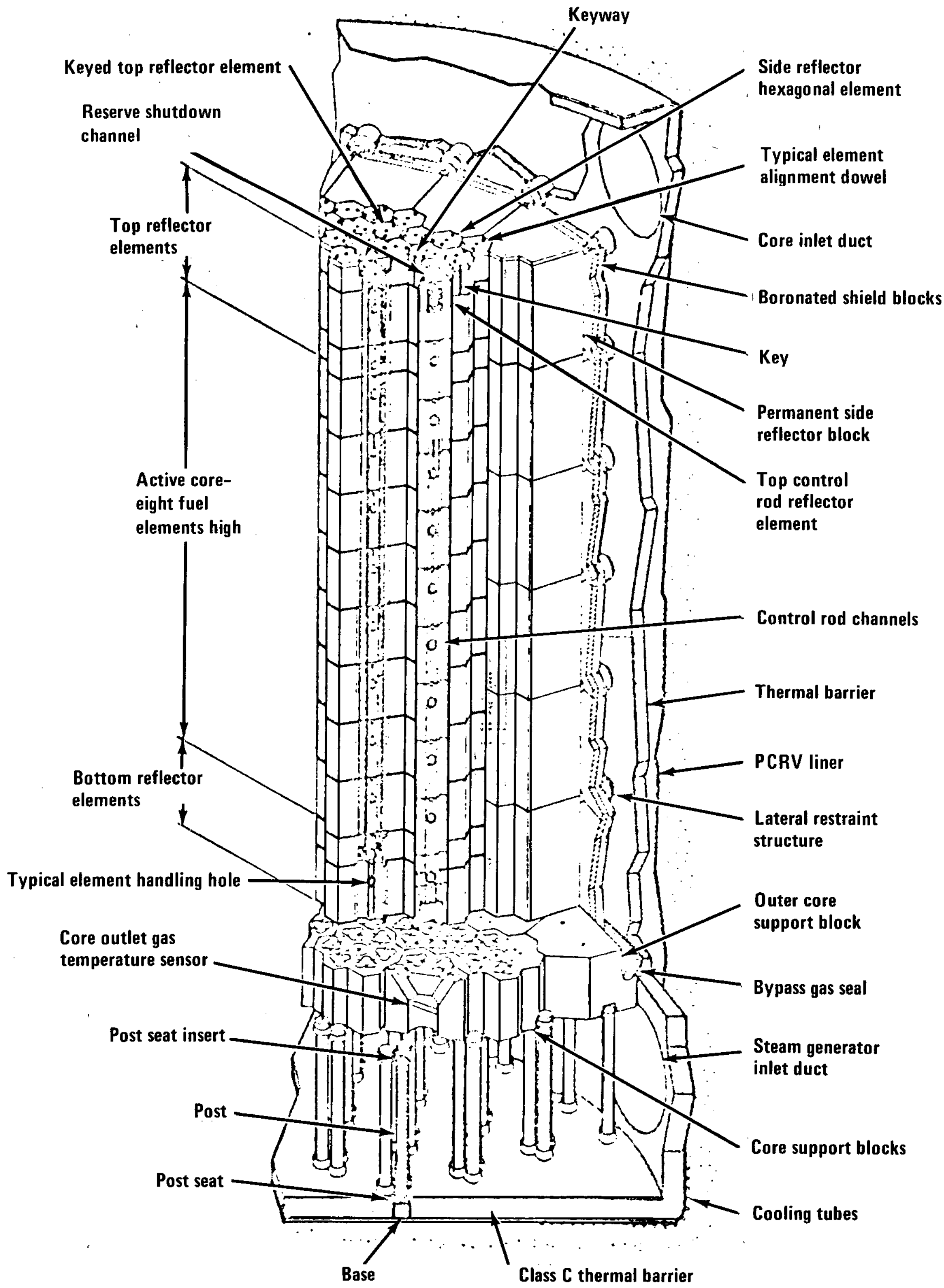


Figure 1-6. Reactor core arrangement.

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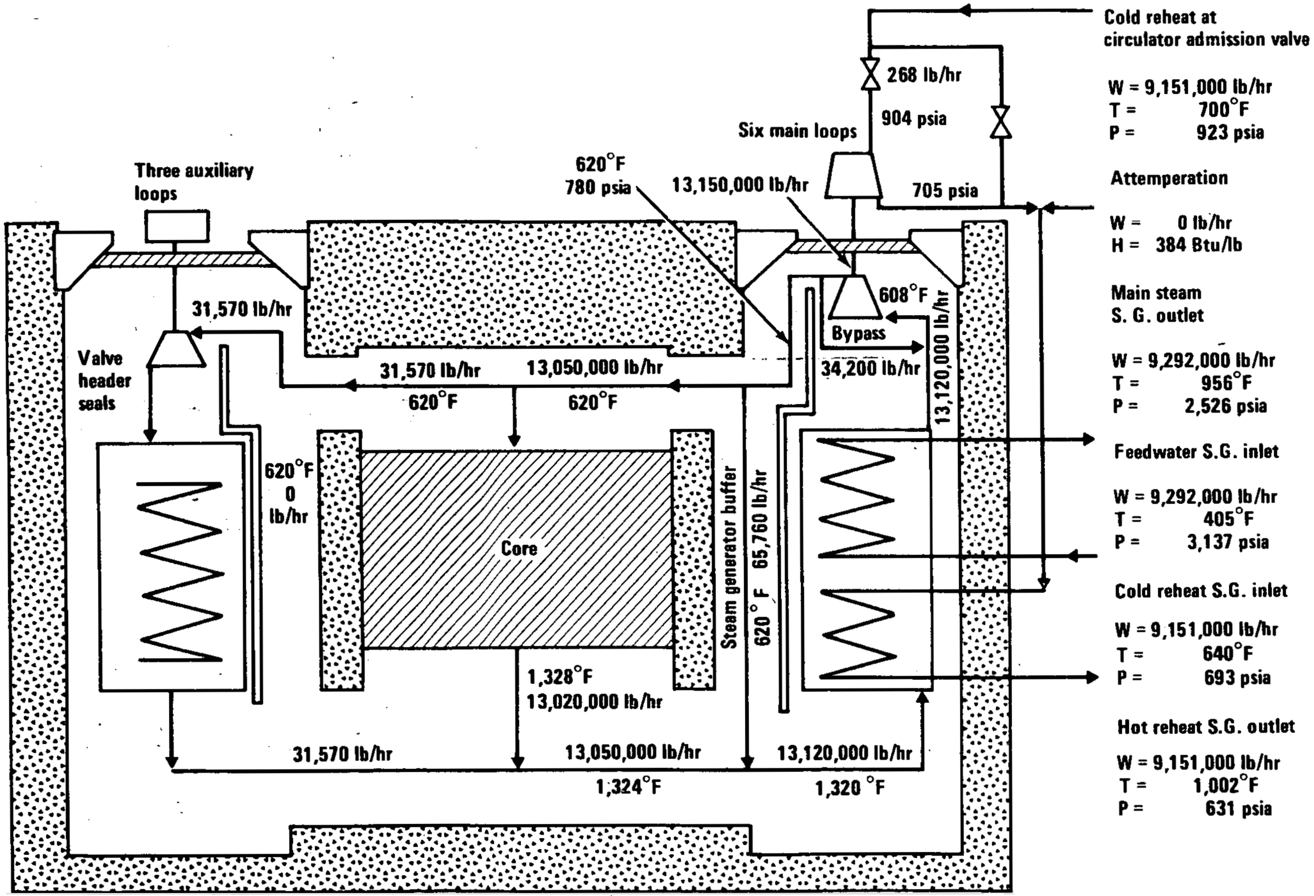


Performance

Heat generated by core	3360 MW
NSSS thermal power	3338 MW
Heat transferred by steam generators	3398 MW
Heat added by circulators	58.5 MW
Heat loss to CACS	9.00 MW
Heat loss to PCRV liner cooling system from	
S. G. cavity	4.91 MW
Core cavity	2.55 MW
CaHe cavity	0.85 MW
Miscellaneous heat losses	3.00 MW
Helium operating pressure	780 psia
Helium pressure drop	16.96 psia
Total circulating helium flow	13,150,000 lb/hr
Helium circulator turbine power	60.1 MW

Legend

W	Flow
P	Pressure
H	Enthalpy
T	Temperature
SG	Steam generator

Figure 1-7. Flow diagram for the primary and secondary coolant of the HTGR nuclear steam supply system.

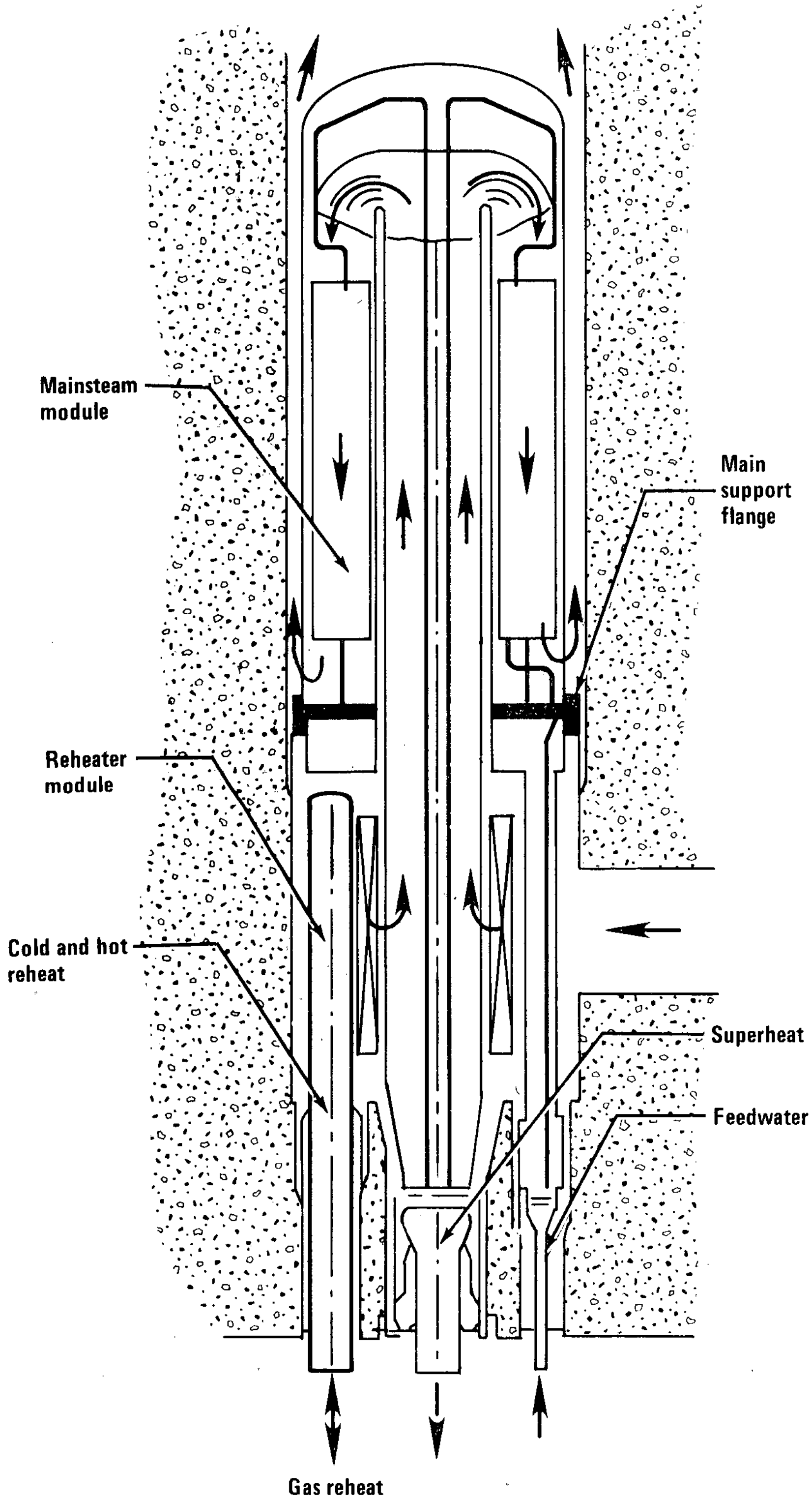


Figure 1-8. Steam-generator arrangement for the HTGR plant.

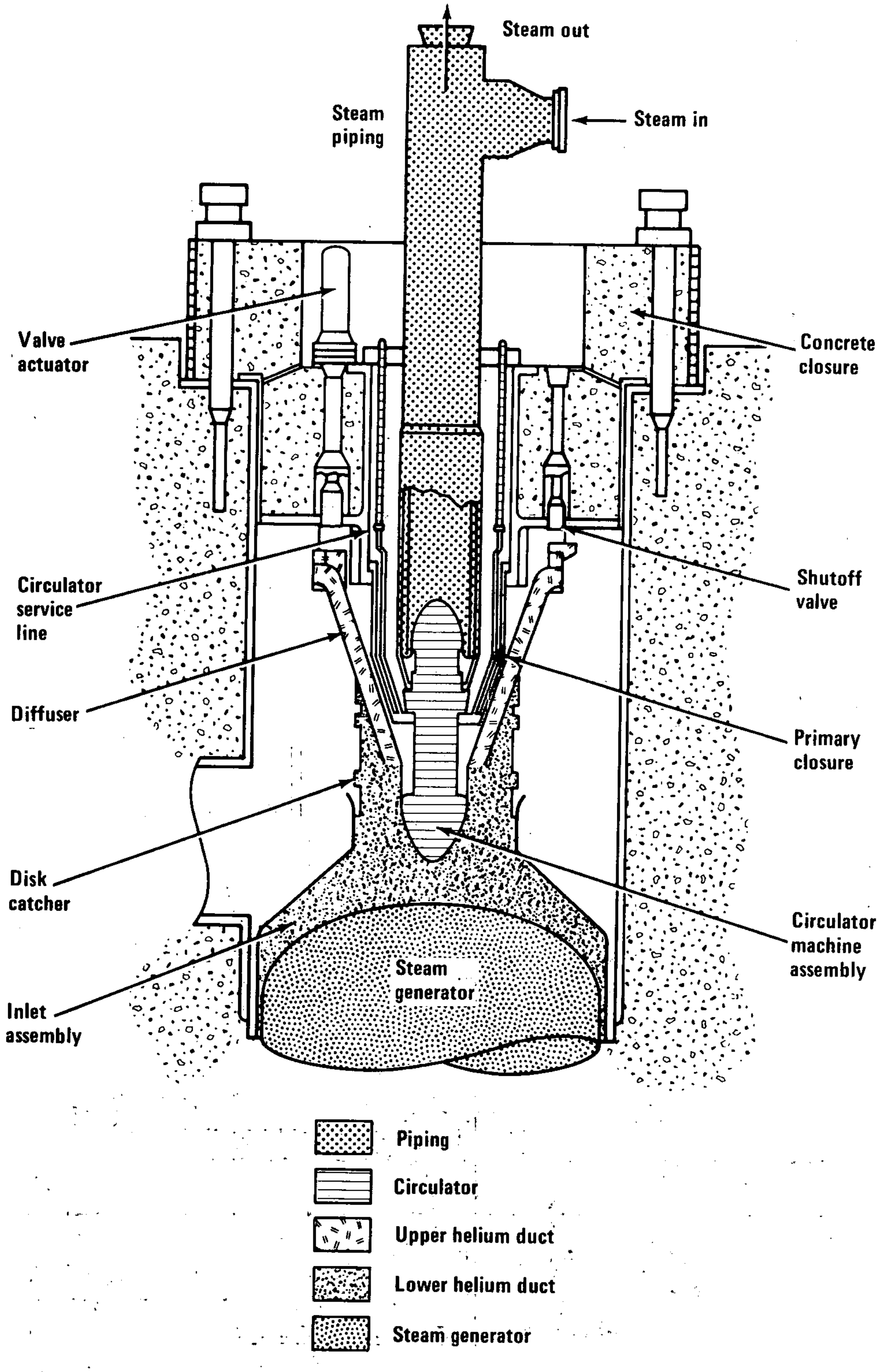


Figure 1-9. Helium-circulator arrangement for the HTGR.

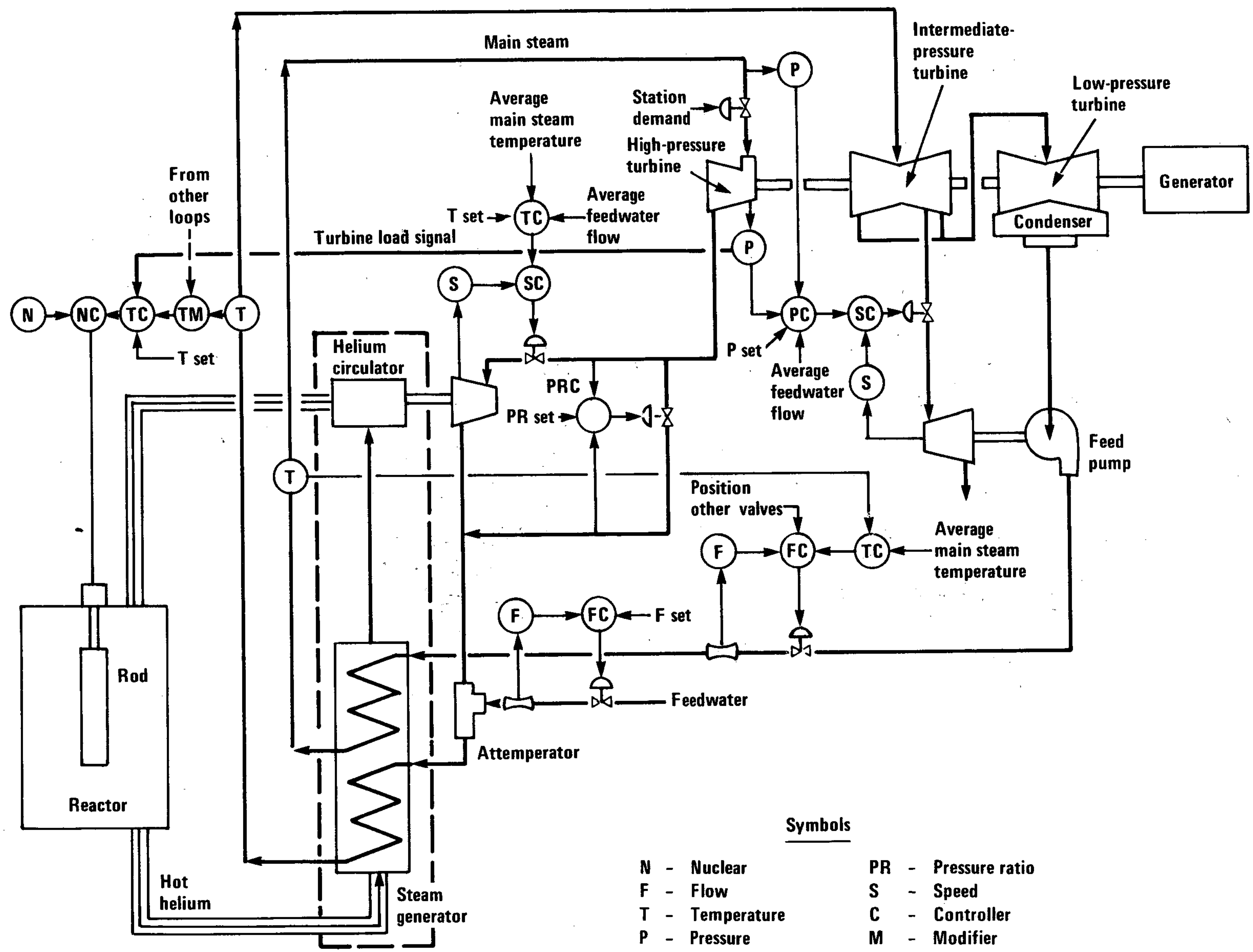


Figure 1-10. Plant control system.

1.3 BALANCE OF PLANT

1.3.1 MAJOR BALANCE-OF-PLANT SYSTEMS

Power conversion is accomplished by means of a single full-size, cross-compound, four-flow, 3,600/1,800-rpm turbine with 44.0-inch last-stage blades. The generator terminal power is 1,356.7 MWe at a turbine exhaust pressure of 2.5 inches of mercury.

The main steam system conveys steam from the NSSS to the high-pressure turbine. From the high-pressure turbine the cold-reheat steam is directed back to the NSSS, where it drives the helium circulators. The steam is passed on to the reheater in the NSSS, after which the hot-reheat steam is conveyed to the intermediate-pressure turbine. The exhaust steam is directed to the two low-pressure turbines, which in turn exhaust to one shell condenser. Some of the exhaust steam from the intermediate-pressure turbine is extracted and used to drive the steam-generator feedpump turbines.

Double containment-isolation valves are provided for each of the main steam and feedwater lines, while single isolation valves are provided for each of the cold-reheat and hot-reheat lines. The piping for each steam generator is individually routed beneath the PCRV to the piping vaults, where the isolation valves are located. The piping is headered outside and routed to the turbine building.

Steam is extracted from the intermediate-pressure turbine exhaust to drive two boiler feedpump turbines (55%, 26,700 brake horsepower), each of which drives a steam-generator feed pump (direct drive) and a booster pump through a reduction gear. The boiler feedpump turbines exhaust directly to the condenser.

The condensate and steam-generator feedwater system provides water to the steam-generator inlets at a pressure of 3,137 psia and a temperature of 405°F. The condensate and feedwater system consists of a single-shell, one pass, longitudinal condenser with a divided water box, three 50% condensate pumps, five stages of feedwater heating including a deaerator, a deep-bed polishing demineralizer, and two 55% feedwater pumps.

The circulating water system provides water to the main steam surface condenser for removing waste heat from the cycle. The water is circulated through the condenser by three 33-1/3% centrifugal pumps. The pumps take suction from the water basins of two 50% forced-draft evaporative cooling towers.

The reactor plant cooling-water system (RPCWS), in conjunction with the nuclear service-water system (NSWS), supplies cooling water to maintain the PCRV temperature within prescribed limits and to provide for process-heat removal from certain reactor-plant equipment and the HVAC control room, and decay-heat removal from the fuel-storage pool. The system consists of two independent and redundant closed cooling loops. Under normal plant conditions, cooling water is provided by the non-safety-related plant service-water system. Under emergency conditions, cooling is provided to each loop of the RPCWS and NSWS by a separate nuclear service cooling tower.

The CACWS provides a closed-loop supply of cooling water to the core auxiliary heat exchangers so that heat removed from the primary coolant may be rejected to the atmosphere. Three independent loops are provided, one for each core auxiliary heat exchanger, and operation of any two is sufficient to cool down the plant if the primary coolant system is depressurized, while any one is sufficient if the primary coolant system is pressurized. Each loop of the CACWS contains an air-cooled heat exchanger with air flow supplied by six fans driven by electric motors.

1.3.2 MAJOR BALANCE-OF-PLANT STRUCTURES

The reactor containment building is a steel-lined, reinforced prestressed concrete cylinder with a hemispherical dome and circular base mat. The building is an earthquake-resistant structure (Seismic Category I) and is designed to minimize leakage of radioactive fission products and to maintain the minimum containment pressure required for adequate operation of the core auxiliary cooling system under conditions associated with a design-basis accident. The design pressure is 58 psig. Housed within the containment building and supported by the base mat are the PCRV and portions of the main and auxiliary circulator service systems. The internal diameter of the containment is sized to provide a sufficient annulus area for rewinding the PCRV with the wire-winding machine.

The reactor service building and fuel storage building are earthquake-resistant (Seismic Category I) reinforced-concrete structures. They are on a common mat and share a common wall in the area of the fuel sealing and inspection facility. The reactor service building contains equipment necessary to serve the NSSS, such as control-rod-drive storage, radwaste-system; and fuel-handling, inspection, and shipping equipment. The analytical instrument room is also located in this building. The fuel storage building is based on containerized fuel storage. The design will allow expansion of the fuel storage with minor modification of the existing arrangements.

The control and diesel-generator building is a Seismic Category I reinforced-concrete structure adjacent to the containment building. The diesel generators are located within a separate portion of the principal structure. The building houses the main control room, computer room, twin cable-spreading areas, switchgear and battery rooms, and helium-purification equipment.

The turbine building is supported on reinforced-concrete spread footings and consists of steel framing with metal siding. The turbine-generator is supported by a high-tuned, reinforced-concrete pedestal within the building.

The access control building is a nonseismic structure built of structural steel and metal siding. It is used for access control and radiological facilities and contains equipment for the helium storage and nitrogen systems.

REFERENCES FOR CHAPTER 1

1. General Atomic Company, High Temperature Gas-Cooled Reactor--Steam Cycle, Lead Plant Design Description, GA-A14667, October 1977.

2. General Atomic Company, General Atomic Standard Safety Analysis Report, GASSAR-6, February 3, 1975.

Chapter 2

MEDIUM-ENRICHED URANIUM/THORIUM ONCE-THROUGH FUEL CYCLE

2.1 DESCRIPTION

This reactor/fuel-cycle combination is a high-temperature gas-cooled reactor (HTGR) using 20% uranium-235/thorium oxycarbide particle fuel operating on a once-through fuel cycle. Spent fuel will be stored at the reactor site or at an away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent-fuel repository. Low-level wastes from fabrication will be sent to a shallow land disposal site.

2.1.1 FUEL MECHANICAL DESIGN

2.1.1.1 Design Bases

The primary mechanical design basis for the reactor core is to provide an array of fuel and reflector elements that are capable of transferring the generated fission heat to the helium coolant efficiently while maintaining structural integrity and containment of the fission products under all normal operating conditions and anticipated transients. The position and structural restraint for the columns of fuel and reflector elements that make up the active core are provided by the core-support and lateral restraint structures.

To meet the primary fuel-design basis, certain specific design bases and limits are imposed on the mechanical design of the hexagonal fuel-element and reflector-element assemblies in the reactor. For example, for the fuel reflector columns, structural features are provided to maintain the alignment of coolant and poison channels within the reactor core to ensure coolant flow and neutron-poison insertion. The following limits are imposed on the graphite fuel elements themselves:

1. The maximum principal stresses in the graphite elements shall be limited to the values listed in Table 2-1.
2. The irradiation-induced dimensional change of the individual graphite elements shall be maintained within the following limits:

Element length	0.5% expansion, 5.0% contraction
Element width	0.5% expansion, 2.0% contraction
Element bowing	0.15 in.

3. The effect of seismic loads on the fuel elements shall not exceed the following:

    a. One-half safe-shutdown earthquake: No core element disarray or damage shall occur such that normal full-power operation cannot be maintained or resumed.

    b. Safe-shutdown earthquake: The core elements shall retain their structural configuration to allow sufficient control poison to be inserted into the core to ensure safe shutdown and allow sufficient coolant flow to be maintained through the coolant channels to remove the reactor-core decay heat.

A complete description of the design basis is given in Section 4.2.1.1 of Reference 1.

2.1.1.2 Design Description

a. Refueling Regions

The core consists of vertical columns of hexagonal elements arranged on a triangular pitch. These columns are grouped into refueling regions (see Figure 2-1) containing seven columns each, except at the outer edge of the core, where additional columns are used to fill out an approximately circular array. The pitch between columns within a region is 14.21 inches.

Each refueling region rests on a single large hexagonal graphite core-support block, which is a part of the core support, and lateral restraint structures (Figure 1-6). Each column is aligned on the support block with graphite dowels.

Each refueling region is directly below a refueling penetration that contains a control-rod-drive assembly during operation. Two parallel channels are provided for inserting the two shutdown control rods within the center column of each refueling region. A third channel is provided in the same column for inserting reserve shutdown absorber material. A fourth, small, channel is provided for a control rod used for power shaping and reactivity control under normal operation conditions.

Each seven-column region is keyed together at the top with steel elements containing rectangular vertical keys that mate to slots in adjacent elements. Certain peripheral columns are keyed at the top to the permanent side-reflector structure. This ensures column stability during refueling operations.

The elements within the center column of each region are displaced axially downward relative to the elements in the surrounding six columns. This prevents the possibility of a continuous shear plane at element interfaces across the core.

b. Columns

The vertical columns that make up the core assembly consist of fuel, control, and reflector elements. A typical fuel column consists of two bottom reflector elements, eight fuel elements, two top reflector elements, and a keyed plenum element.

A typical control fuel column has two bottom reflector elements, eight control fuel elements, and two top reflector elements. A typical removable side-reflector column has 12 solid-graphite reflector elements and a top keyed element. The elements within each column rest on the flat end-face of the element below. The alignment of the coolant channels and the control-rod channels within the columns is maintained by four graphite dowels, on the top face of each element, that fit into mating socket holes in the bottom face of the element above.

Neutron shielding for the prestressed-concrete pressure vessel (PCRV) and liner in the top and bottom heads are provided by the graphite reflector above and below the active core and by the use of boronated graphite. In each fuel column, a top reflector element and a bottom reflector element contain vented metal tubes (shield pins) filled with boronated graphite. The shield pins are located in blind holes between the coolant channels. The metal tubes in the top reflector are made of stainless steel, and those in the bottom reflector are made of Incoloy 800. Shield pins are not necessary in the control fuel columns. The top keyed steel elements of the side reflector columns are filled with boronated graphite. This eliminates the need for shield pins in the graphite side-reflector elements just below the keyed steel elements.

c. Fuel and Control Fuel Elements

The fuel elements are graphite hexagonal right prisms with arrays of fuel, coolant, and burnable poison holes. Control fuel elements are identical with the other fuel elements, except for three large-diameter holes that form the channels for control-rod and reserve-shutdown-poison insertion. Figure 1-5 shows both standard and control fuel elements. The designs of the hexagonal fuel element and the reflector element are similar to those in the Fort St. Vrain reactor.

Coolant channels extend through each element and are aligned with coolant channels in elements above and below. The active fuel is contained in an array of blind and plugged holes that are parallel with the coolant channels and occupy alternating positions in a triangular array. Additional holes are provided in the corners of the elements for loading the burnable poison.

A hole at the center of each fuel element is provided for handling purposes. The hole profile is shaped so that a lifting ledge is produced at the lower end. The grapple head of the fuel-handling machine bears against this ledge when lifting an element.

d. Fuel Rods

The fuel particles are bonded together into fuel rods. The bonding matrix consists of an organic binder and a graphitic filler. The rods are carbonized and heat-treated to outgas the binder. The fuel particles in the fuel rod are a mixture of fissile and fertile types and are uniformly blended to provide the necessary uranium and thorium content. Various blends are produced to provide the required heavy-metal loadings in the fuel elements. The rod is sized to give a close fit inside the fuel hole. The rods are stacked in the fuel hole to make up the total fuel length in the fuel-element assembly.

e. Removable Reflector Elements

The reflector elements are graphite hexagonal right prisms and vary in design, depending on their location in the core assembly. All of them, however, have the same hexagonal cross section, dowel pattern, and handling hole as the fuel and control fuel elements.

The removable side-reflector elements are of solid graphite. Two different lengths are used, either full length or half-length relative to the fuel elements. These elements have special features that alert the fuel-handling machine that a solid element is being handled: an extra long dowel, the absence of coolant holes, and a difference in weight from the elements in the fuel and control columns.

The bottom-reflector elements channel the flow from the individual coolant holes in the fuel elements to the flow passages in the core-support structure described. The coolant flow from the individual coolant holes in the fuel columns is collected into three large-diameter coolant holes in the half-length reflector element directly below the bottom fuel element. The coolant then passes through a full-length reflector element with matching coolant holes into the large support block. This reflector element also contains the shield pins. Dowels are provided to mate to the core-support structure. The flow in the control columns passes through a half-length reflector element directly below the bottom control fuel element with an identical array. This element also has two large-diameter holes aligned with the two control-rod channels in the control fuel element to allow complete insertion of the control rods. The coolant flow from the individual coolant holes is collected into a single plenum within the reflector element just above the core-support block. This element is a three-quarter-length element that permits the axial displacement of the control fuel elements in relation to the fuel elements as previously described. Horizontal slots in the bottom faces of this element and the neighboring elements in the fuel columns allow the coolant to be routed to the adjacent fuel columns and into the core-support block.

The top reflector consists of two graphite elements in each column just above the active core. The element just above the active core is a half-length element, and the next reflector element is a full-length element. Both elements contain an array of coolant channels, and in the case of the control fuel columns, an array of control-rod and reserve shutdown channels that match the array of channels in the fuel elements. The full-length reflector element also contains the shield pins. The steel keyed reflector elements are located above the full-length graphite reflector elements. The arrangement of the top-reflector elements is shown in Figure 1-6.

2.1.1.3 Design Evaluation

The fuel and control fuel elements described in the preceding sections have been evaluated to determine their structural integrity under all operating conditions. The areas evaluated were the following:

1. Methods of analysis
2. Graphite stresses
3. Graphite dimensional change
4. Handling-hole integrity
5. Dowel and socket integrity
6. Seismic impact loading

The mechanical performance analyses and evaluations show that the fuel and control elements will retain their structural integrity throughout the design lifetime under all operating conditions within the core. A complete description of the design evaluation is given in Section 4.2.1.3 of Reference 1.

2.1.1.4 Testing and Inspection Plan

The fuel for the HTGR is manufactured in accordance with a detailed generic specification that defines the process, product, inspection, and quality-assurance requirements. Raw materials are purchased in accordance with rigid material and quality-assurance specifications. Purchased components are fabricated and inspected in accordance with rigid product and quality-assurance specifications. The product is inspected and tested at each stage. Table 2-2 presents the typical parameters controlled and inspected during the manufacture of the reactor core components.

2.1.2 FUEL NUCLEAR DESIGN

2.1.2.1 Design Bases

The design bases for the nuclear design of the fuel and reactivity-control systems are as follows:

1. The core shall be designed to maintain a rated power level of 3,360 MWt at an average power density of 7.1 kWt/liter.
2. The reactor shall be designed to operate on a graded uranium/thorium fuel cycle. The basic fuel-management objective is to obtain a design that will have low fuel-cycle costs within the constraints of thermal and metallurgical performance limits.
3. In the equilibrium cycle, the fuel lifetime shall be designed to be the equivalent of 4 years at an 80% load factor at rated power.
4. The design shall accommodate partial refueling, wherein approximately 25% of the core can be replaced at each reloading on a nominally annual basis. The core is divided into four segments; the core layout is shown in Figure 2-1. Other options, such as more frequent refueling or a different fuel lifetime, are possible with this design.
5. Isothermal and fuel-temperature coefficients shall be negative from room temperature (300 K) to beyond 3,000 K. The coefficients tend to compensate for any reactivity insertion and enhance the reactor's stability against power oscillation. The fuel (Doppler) temperature coefficient provides a prompt, negative reactivity feedback mechanism.
6. The core shall be designed so that axial xenon oscillations will not occur. Instrumentation shall be provided to detect any radial flux tilt or radial or azimuthal oscillations that might occur. These conditions shall be correctable by appropriate control-rod motion.
7. The fuel and lumped burnable poison in the core shall be zoned to minimize radial and axial gross and local power tilts and to maintain the power peaks within design limits throughout life, with due allowance for uncertainties in calculations and loading. Axial zoning shall be designed so that core thermal design bases are not exceeded under normal operating conditions. Normal operating conditions permit partial insertion of control rods, as required near the end of a refueling interval and for load following, flux-oscillation control, and power-peak suppression.
8. Core excess reactivity shall be designed to be compensated by burnable poison and control rods. At 100% power, the burnable poison shall be worth about 0.10 Δk at the beginning of each cycle and shall be essentially fully depleted by the end of each cycle. The fuel cycle shall be designed so that the maximum excess reactivity to be controlled by rods is about 0.025 Δk after equilibration of protactinium-233, xenon-135, and samarium-149.
9. The primary shutdown-control system consists of movable rods arranged as pairs and containing a neutron poison. It is designed to ensure safe shutdown from any credible steady-state accident conditions. Safe shutdown shall be designed with a minimum margin of 0.010 Δk, including allowances for uncertainties, under any of the following conditions:

   a. Indefinite shutdown at room temperature with the maximum-worth rod pair stuck out.
   b. A minimum of 14 days (following extended power operation) at refueling temperature with the two maximum-worth rod pairs stuck out.

c. A minimum of 14 days with up to three nonadjacent rods withdrawn at refueling temperature.

The control-rod pair withdrawal speed is limited to 1.2 in./sec or less, primarily to limit the consequences of an uncontrolled-rod withdrawal accident. With this speed of withdrawal, the maximum controlled reactivity-insertion rate is 0.00038 Δk/sec at source power level and 0.00013 Δk/sec at operating power levels. Full rod-bank insertion after a trip signal requires 22 $\pm$ 3 seconds.

10. A reserve shutdown system (RSS) shall provide an independent shutdown reactivity control through a poison-insertion mechanism actuated independently from the primary system of control rods. With all hoppers operable, this backup system shall have sufficient negative reactivity to shut down the reactor from normal operating conditions and after anticipated transients without scram. The RSS shall be capable of maintaining a safe-shutdown condition for a period sufficient to effect a permanent cold shutdown with the primary reactivity-control system. Once activated, the RSS shall be as effective as the control-rod system in terminating reactivity transients.
11. Reactivity control under normal operating conditions shall be accomplished by means of small, neutronically "grey" control rods, one per refueling region. These rods shall be operated in banks, or subbanks, with one bank being all the rods in a given fuel segment. Some rod pairs may be used for control during startup. These "grey" rods are not required for shutdown (see items 9 and 10 above). Their function is to provide uniform reactivity control with minimum perturbation to the core power distribution.

2.1.2.2 Description

The HTGR utilizes a semihomogeneous graphite-moderated core based on the thorium/uranium cycle. The reference design uses partial refueling, with approximately 25% of the core being replaced at each reload on a nominally annual basis. The fraction of the core reloaded at a given refueling is called a segment. There are four segments in the core (their distribution is shown in Figure 2-1, in which each refueling region is designated as part of one of the four segments A, B, C, or D). Uranium enriched to approximately 20% is used as feed fissile material for the initial core and reload segments. A true equilibrium (i.e., repeating) cycle may never be achieved since variations in load factor and the fact that the four annually refueled core segments are not exactly the same size prevent one yearly cycle from exactly duplicating the nuclear behavior of the previous yearly cycle.

Fissile and fertile materials in the equilibrium core are radially and axially zoned to achieve temperature distributions within design limits. The radial power distribution is flattened in the reload segments to yield more uniform radial fuel temperatures. The axial power distribution is peaked toward the core inlet to yield a relatively constant axial fuel centerline temperature distribution. The control-rod program sequence is designed to supplement the fuel zoning in achieving desirable power and hence desirable fuel-temperature distributions.

The core is designed to have a net negative temperature coefficient at all credible core temperatures. The least-negative temperature coefficients occur at the end of the annual equilibrium fuel cycle, when the fission-product inventory and the fraction of fissions from uranium-233 are at their maximum. For a complete description of the fuel nuclear design see Reference 2.

2.1.2.3 Analytical Methods

For a complete description of analytical methods, see Reference 3.

2.1.2.4 Nuclear Design Changes

Nuclear design changes are summarized in References 2 and 4. In summary, compared to earlier designs and the Fort St. Vrain core, there have been three changes. First, an 8-row fuel element has been adopted as compared to the 10-row Fort St. Vrain fuel block. Second, small neutronically "grey" control rods, one per refueling region, have been added to the core. These rods are operated in banks, each of which consists of all the rods in a given refueling segment. They provide distributed and uniform power and reactivity changes during normal operation and improve load-following capability. They are not used for shutdown purposes. Third, 20% enriched uranium plus thorium, so-called medium-enriched uranium fuel, has been used in place of the original highly enriched uranium/thorium fuel cycle. Recent analyses (Ref. 4) show that this change has little effect on fuel performance and fission-product release.

2.1.3 FUEL MANAGEMENT

Fuel-management information is given in Table 2-3. Fresh and spent fuels are characterized in Table 2-4, which includes data on the heavy-element isotopic content for initial and equilibrium loadings and discharges. Fuel mass-flow data (charge and discharge) are presented in Table 2-5, and isotopic data for each core segment are given in Table 2-6.

The material flow diagram for the HTGR once-through (throwaway) fuel cycle is presented in Figure 2-2. The core layout identifying the segments and regions listed in Table 2-5 is shown in Figure 2-1.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram (Figure 2-2) and are discussed in the following sections of Volume VII:

Enrichment	Chapter 3
Fuel fabrication 7	Chapter 4
Spent fuel storage 3	Section 6.3
Waste disposal 1	Section 7.1
Waste disposal 3	Section 7.3

Table 2-1. Allowable stresses for the graphite components of reactor-vessel internals[a]

Operating condition	Primary[b] stresses	Primary plus secondary stresses (thermal)[b]
Normal and upset components that support the weight of the core	0.2	0.4
Other graphite	0.33	0.4
Emergency	0.33	0.67
Faulted	0.9	0.9
Test	0.2	0.4

[a]These stress values are allowed to be exceeded in local areas, such as Hertzian bearing stresses, etc., provided all three of the following conditions are satisfied:

1. The stress is strictly local--that is, only a small amount of material is affected.
2. The stress is "self-limiting"--that is, if the affected material fails, the stresses in the remainder of the structure will not exceed the allowable limits.
3. The required safety factor of the component must be demonstrated by tests.

[b]In terms of the specified minimum ultimate strength or modulus of rupture of the material, as appropriate.

Table 2-2. Typical quality inspections of reactor-core components

Component	Production stage	Inspected parameters	Reason for inspection
Fuel elements	Kernel	Composition	Heavy-metal loading and stoichiometry requirements
		Shape and size	Acceptance for coating and proper size range
	Coating	Thickness	Mechanical performance and migration allowance
		Density and isotropy	Irradiation performance and dimensional changes
		Defective coatings	Design basis for failed particles at end of life
		Surface contamination	Design basis for primary circuit activity
	Fuel rod	Fuel loading	Reactor-core fuel zoning requirements
		Fuel homogeneity	Limit fuel hot-spot temperatures
		Matrix structure	Irradiation structural integrity
		Dimensions	Assembly clearances and hot-spot temperatures
	Graphite	Strength	Design basis for stress analysis
		Density	Core carbon content for C/Th/U requirements
		Impurities	Core reactivity requirements
		Internal structure	Element structural integrity
	Burnable poison rods	Boron loading	Core reactivity requirements
		Matrix properties	Mechanical property requirements and irradiation stability
	Assembled element	Dimensions	Proper cooling, clearances, webs, etc.
		Fuel loading	Element loading and core loading requirements
		Burnable poison loading	Core reactivity requirements
		Permanent identification	Traceability and correct placement in core
	Site receiving	Visual inspection	Examination for shipping damage
Reflector elements	Graphite	Strength	Design basis for stress analysis
		Density	Reflector carbon content
		Impurities	Core reactivity requirements
		Internal structure	Element structural integrity
	Assembled element	Dimensions	Proper cooling, clearances, webs, etc.
		Permanent identification	Traceability and correct placement in core
	Site receiving	Visual inspection	Examination for shipping damage
Steel plenum elements	Steel	Mechanical properties	Purchase specification requirements, material performance design limits
		Chemical composition	
	Assembled element	Dimensions	Proper size, clearance, key engagement, etc.
		Welds	Strength, integrity, design stresses

Table 2-2. Typical quality inspections of reactor-core components (continued)

Component	Production stage	Inspected parameter	Reason for inspection
Control rods	Flow test with orifice valve	Flow characteristics	Calibration flow test with orifice flow valve to verify flow characteristics
	Site receiving	Visual inspection	Examination for shipping damage
	Poison material	Boron loading	Required negative reactivity worth
		Matrix properties	Mechanical property requirements and irradiation stability
	Cladding material	Mechanical properties	Purchase specification requirements, material performance design limits
		Chemical composition	
	Shock absorber	Deformation properties	Confirmation of shock-absorbing capabilities meet design requirements
	Flow test	Flow stability	Confirmation of vibration stability and pressure drop at rate flow conditions
	Assembled rod	Dimensions	Proper size, flow clearances, flexibility
		Welds	Strength, integrity, design stresses
	Site receiving	Visual inspection	Examination for shipping damage

Table 2-3. Parameters for the medium-enriched uranium/thorium once-through fuel cycle

Average capacity factor, %	75
Fuel form	Oxide or carbide coated particles
Fraction of core replaced annually	0.25
Enrichment-plant tails assay, %	0.2
Core power density, W/cm^3	7.1
Carbon-to-heavy-metal ratio	
Initial core	270
Equilibrium reload	380
Fuel-rod diameter, cm	1.17
Average fuel temperature, ^{o}C	880
Maximum fuel temperature, ^{o}C	1,350
Yellowcake requirements, ST/GWe	
Initial core	340
Equilibrium annual	144
30-year total	4,510
30-year cumulative, net[a]	4,280
Separative-work requirements, 10^3 SWU/GWe	
Initial core	309
Equilibrium annual	131
30-year total	4,100
30-year cumulative, net[a]	3,910
Core fuel loading, kg/GWe (initial core/equilibrium reload)	
Total heavy metal	30,600/5,360
Fissile material	1,350/576
Burnup, MWd/MTHM	
Average	130,000
Peak	165,000
Conversion ratio	
Beginning of life (initial core)	0.59
After equilibrium fuel loading	0.48
Average during equilibrium	0.54
Annual discharge, kg/GWe	
Fissile plutonium	29
Total plutonium	59
Uranium-235	47
Bred uranium-233	64
Total uranium	2,260
Total thorium	2,290

[a]The 30-year cumulative net is equal to the 30-year total less a credit for the savings in yellowcake and separative-work requirements due to the reuse, at the end of plant life, of partially consumed fuel in other HTGRs (fuel with 1 year or more unused burnup).

Table 2-3. Parameters for the medium-enriched uranium/thorium once-through fuel cycle (continued)

30-year cumulative discharge, kg/GWe[b]	
Fissile plutonium	950
Total plutonium	1,990
Uranium-235	2,010
Bred uranium-235	2,070
Total uranium	75,230
Total thorium	75,830

[b]The 30-year cumulative discharge is the sum of 30 annual discharges plus the partially consumed heavy metal in the reactor at the end of plant life.

Table 2-4. Characterization of HTGR fresh and spent fuel for the medium-enriched uranium/thorium once-through fuel cycle

Refueling method	Batch
Refueling frequency	1 year
Fuel-assembly characteristics	
Type	Oxide and carbide
Weight, kg	100
Length, m	0.79
Core load mass, kg HM/GWe	30,600
Annual reload mass at 75% capacity factor, kg HM/GWe	5,327
Design burnup,[a] MWd/MT	130,000
Dose rate at 1 m in air after 90 days, rem/hr	5,000

Heavy-element isotopic content[b]

Isotope	Fresh fuel element (kg) Initial	Fresh fuel element (kg) Equilibrium	Discharged fuel element (kg) Initial	Discharged fuel element (kg) Equilibrium
Thorium-232	6.0	2.5	5.6	2.3
Uranium-232	--	--	7.0×10^{-5}	2.9×10^{-5}
Uranium-233	--	--	0.14	0.07
Uranium-234	--	--	0.03	0.01
Uranium-235	0.34	0.58	0.03	0.05
Uranium-236	--	--	0.05	0.08
Uranium-238	1.37	2.34	1.2	2.1
Neptunium-237			0.005	0.009
Plutonium-238			0.003	0.004
Plutonium-239			0.012	0.020
Plutonium-240			0.008	0.013
Plutonium-241			0.006	0.010
Plutonium-242			0.008	0.012

[a]Discharge batch average.

[b]Multiply by 993 (fuel elements per GWe) for the isotopic content in kilograms per GWe.

Table 2-5. Fuel mass flows[a] for the medium-enriched uranium/thorium once-through fuel cycle

Segment	1	2	3	4	5	6	7	8	9	10	11	12	13
Region	A	B	C	D	A	B	C	D	A	B	C	D	A
Discharge time (yr)	1.00	2.00	3.00	4.00	5.00	6.00	7.00	8.00	9.00	10.00	11.00	12.00	13.00
Thorium charged	7,950.3	7,950.3	7,950.3	7,950.3	2,408.9	2,812.7	2,999.4	3,101.0	3,272.3	3,279.5	3,281.8	3,287.7	3,289.5
Uranium-235 makeup	445.8	445.8	445.8	445.8	712.9	726.2	741.6	761.9	802.5	782.3	768.3	756.6	735.1
Total uranium makeup	2,251.8	2,251.8	2,251.8	2,251.8	3,600.8	3,667.9	3,745.9	3,848.6	4,053.5	3,951.2	3,880.9	3,821.8	3,713.1
Total uranium loaded	2,251.8	2,251.8	2,251.8	2,251.8	3,600.8	3,667.9	3,745.9	3,848.6	4,053.5	3,951.2	3,880.9	3,821.8	3,713.1
Total metal loaded	10,202.2	10,202.2	10,202.2	10,202.2	6,009.7	6,480.6	6,745.3	6,949.5	7,325.8	7,230.7	7,162.7	7,109.5	7,002.6
Thorium discharged	7,813.9	7,679.4	7,546.3	7,413.7	2,224.2	2,598.5	2,772.5	2,868.4	3,030.5	3,038.1	3,040.1	3,044.9	3,045.8
Uranium-233 retired	102.3	150.7	173.1	182.7	62.3	72.8	78.2	81.3	85.7	85.6	85.3	85.2	85.2
Uranium-235 retired	220.3	113.3	60.9	35.4	49.0	51.2	54.2	58.5	65.7	65.6	64.3	62.2	59.2
Total uranium retired	2,119.7	2,033.7	1,967.5	1,910.6	2,803.7	2,868.5	2,937.0	3,024.1	3,196.1	3,117.6	3,061.9	3,014.0	2,926.8
Total uranium discharged	2,119.7	2,033.7	1,967.5	1,910.6	2,803.7	2,868.5	2,937.0	3,024.1	3,196.1	3,117.6	3,061.9	3,014.0	2,926.8
Fissile plutonium retired	19.5	23.8	24.7	24.5	35.0	36.7	38.4	39.9	41.7	40.4	39.4	38.7	37.9
Total plutonium retired	28.0	39.0	45.2	49.7	72.0	74.2	76.5	78.7	82.0	79.9	78.5	77.3	75.6
Total metal discharged	9,961.6	9,752.1	9,559.0	9,374.0	5,099.9	5,541.3	5,786.0	5,971.3	6,308.5	6,235.7	6,180.5	6,136.2	6,048.2

[a]Mass flows are in kilograms.

Table 2.5. Fuel mass flows[a] for the medium-enriched uranium/thorium once-through fuel cycle (continued)

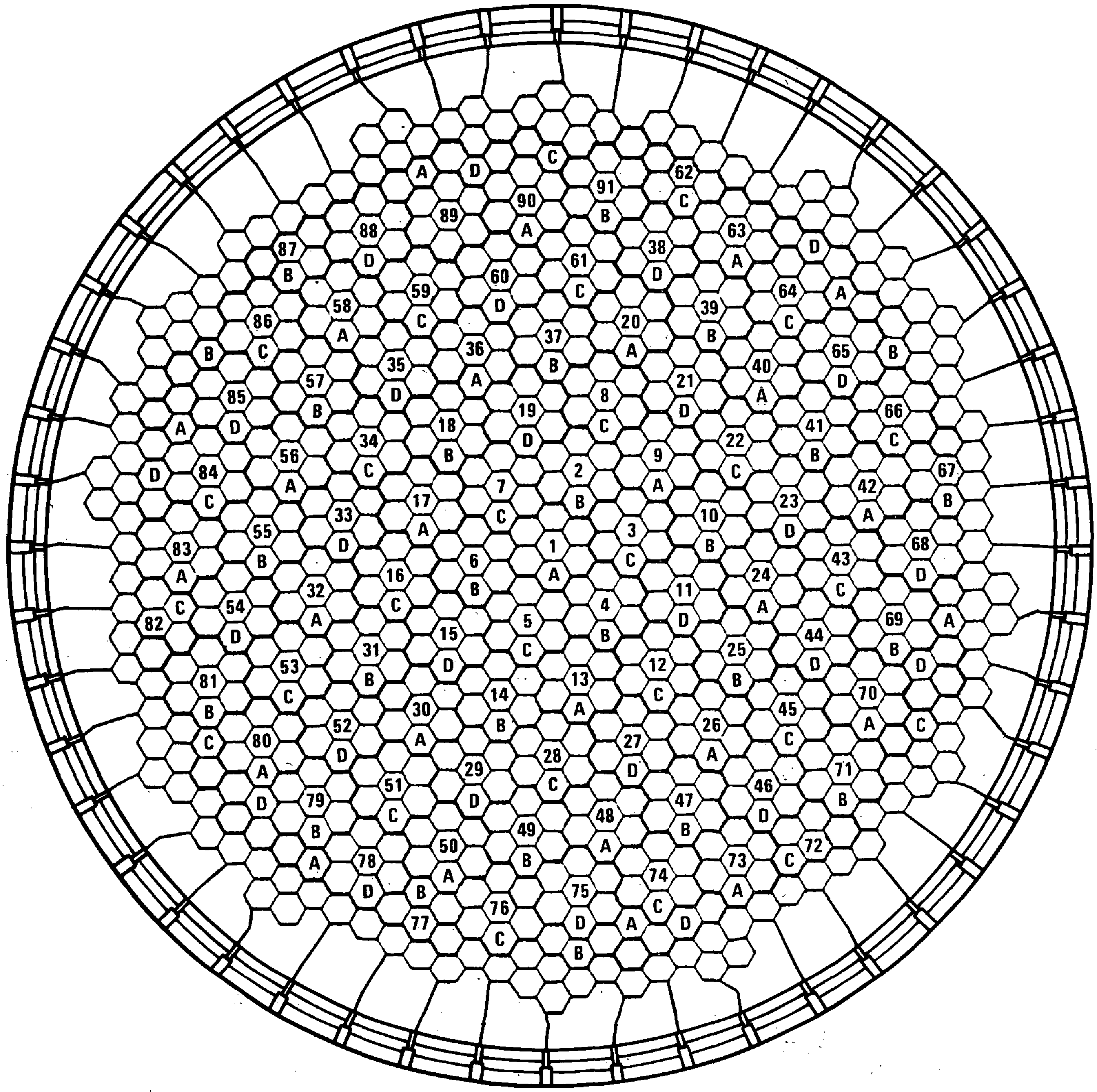
Segment Region Discharge time (yr)	14 B 14.00	15 C 15.00	16 D 16.00	17 A 17.00	18 B 18.00	19 C 19.00	20 D 20.00	21 A 21.00	22 B 22.00	23 C 23.00	24 D 24.00	25 A 24.00	26 B 24.00	27 C 24.00
Thorium charged	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5	3,289.5
Uranium-235 makeup	745.5	751.1	754.5	763.0	758.1	755.8	754.9	751.4	753.7	754.6	754.8	756.3	755.2	754.8
Total uranium makeup	3,765.7	3,794.0	3,811.1	3,853.9	3,829.1	3,817.5	3,812.9	3,795.3	3,807.1	3,811.7	3,812.5	3,820.0	3,814.4	3,812.6
Total uranium loaded	3,765.7	3,794.0	3,811.1	3,853.9	3,829.1	3,817.5	3,812.9	3,795.3	3,807.1	3,811.7	3,812.5	3,820.0	3,814.4	3,812.6
Total metal loaded	7,055.2	7,093.5	7,100.6	7,143.4	7,118.7	7,107.0	7,102.4	7,084.8	7,096.6	7,101.3	7,102.0	7,109.5	7,103.9	7,102.1
Thorium discharged	3,045.5	3,045.6	3,045.9	3,046.3	3,406.3	3,046.3	3,046.1	3,046.0	3,046.0	3,046.0	3,046.1	3,105.4	3,165.7	3,227.0
Uranium-233 retired	85.3	85.4	85.5	85.5	85.5	85.4	85.4	85.4	85.4	85.4	85.4	81.1	70.6	47.6
Uranium-235 retired	59.5	60.1	60.8	62.0	61.8	61.5	61.3	60.8	60.9	61.0	61.1	112.5	210.5	398.0
Total uranium retired	2,967.6	2,990.2	3,004.2	3,038.6	3,019.3	3,010.0	3,006.1	2,992.0	3,001.3	3,005.0	3,005.7	3,129.4	3,275.8	3,486.3
Total uranium discharged	2,967.6	2,990.2	3,004.2	3,038.6	3,019.3	3,010.0	3,006.1	2,992.0	3,001.3	3,005.0	3,005.7	3,129.4	3,275.8	3,486.3
Fissile plutonium retired	38.5	38.8	39.0	39.4	39.1	39.0	38.9	38.8	38.9	39.0	39.0	39.1	37.7	31.3
Total plutonium retired	76.6	77.2	77.6	78.2	77.7	77.5	77.4	77.2	77.4	77.5	77.5	70.5	61.0	44.3
Total metal discharged	6,089.8	6,113.0	6,127.7	6,163.0	6,143.3	6,133.7	6,129.7	6,115.2	6,124.6	6,128.5	6,129.2	6,305.2	6,502.4	6,757.7

[a]Mass flows are in kilograms.

Table 2-6. HTGR mass-flow data for the medium-enriched uranium/thorium once-through fuel cycle: equilibrium loadings at a 75% capacity factor

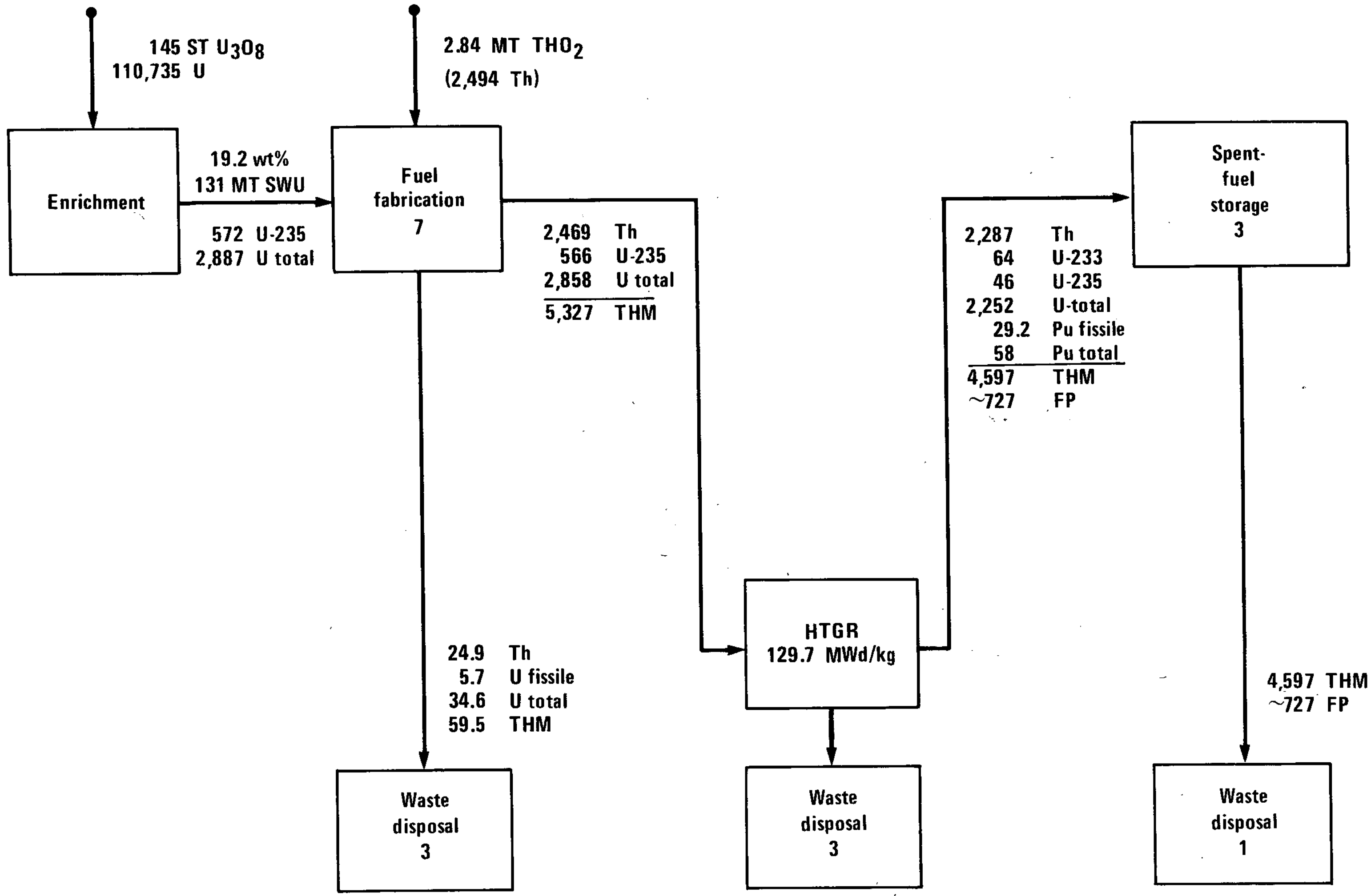
Isotope	Quantity (kg/GWe) Charged	Discharged
Thorium-232	2,469	2,287
Uranium-233	-	64
Uranium-235	566	47
Total uranium	2,858	2,252
Plutonium fissile	-	29.2
Total plutonium	-	58
Total heavy metal	5,327	4,597
Fission products	-	727

Note: Average charge/discharge data for years 20, 21, 22 (Table 2-5) normalized from a 1,332-MWe reactor.



Segment position	Volume fraction
A	0.2557
B	0.2496
C	0.2496
D	0.2451

Figure 2-1. Core layout for a 3,360-MWt HTGR.



Mass flows in kg per 0.75 GWe-yr.
Abbreviations: FP, fission products; THM, total heavy metal.

Figure 2-2. Material flow diagram for the HTGR medium-enriched uranium/thorium once-through fuel cycle.

2.2 SAFETY CONSIDERATIONS

2.2.1 GENERAL

The approach used in the United States to minimize undue risk to the health and safety of the public has been to rely on the "defense-in-depth" philosophy in the design of reactors. This concept requires that reactor systems tolerate a spectrum of operating transient and accident conditions while maintaining barriers to the release of fission products.

The primary assurance of safety is attained through a high degree of reliability and predictability obtained by the application of rigorous standards in the design, construction, and operation of the nuclear facility and through extensive quality-assurance actions. In addition, in accordance with the defense-in-depth concept, safety features and engineered safeguards systems are provided to prevent, or to accommodate the consequences of, accidents postulated to occur in spite of these measures.

Defense in depth includes the following:

1. Designing for safety in normal operation and maximizing the ability to tolerate malfunctions through intrinsic features of sound conservative design, construction, selection of materials, quality assurance, testing, and operation. Margins are incorporated into the plant by adhering to regulatory requirements and the many accepted codes and standards of organizations such as the American Nuclear Society, the American Society of Mechanical Engineers, the American Society for Testing and Materials, and the Institute of Electrical and Electronics Engineers.
2. Anticipating that some abnormal incidents will occur during plant life, provisions are made to terminate such incidents and to limit their consequences to acceptable limits, even though important components or systems fail. Even under these conditions, there are still significant margins provided as a result of utilizing conservative design practice and accepted codes and standards.
3. Providing protection against extremely unlikely events, which are not expected to occur during the life of a single plant, assuming failures of consequence-limiting equipment. From an analysis of these postulated events, features and equipment are designed into the plant to control the postulated events and to ensure that there is no undue risk to the public.

The Nuclear Regulatory Commission (NRC) regulations, as stated in 10 CFR 50, Section 50.34, require that each applicant requesting a construction permit or operating license for a nuclear power plant or a fuel-reprocessing plant provide an analysis and evaluation of the design and performance of the structures, systems, and components of the facility, with the objective of assessing the risk to public health and safety resulting from operation of the facility. These analyses are to establish (a) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (b) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of accident consequences.

The conditions analyzed range from relatively trivial events that result in essentially no risk to the public (such as releases within the criteria for routine operation) and that might occur with moderate frequency, to accident situations that have a theoretical potential for large consequences but are very unlikely. For HTGRs, some 31 types of events must be analyzed in Chapter 15 of the Safety Analysis Report (Ref. 5).

The radiological environmental effects are calculated for each of the above classes using reasonable assumptions, justifiable calculational models and techniques, and realistic assessments of environmental effects. The environmental impact is evaluated in relation to the natural background radiation already present.

2.2.1.1 Frequency Classification

The range of accidents considered can be categorized into three groups described as follows:

A. Events of moderate frequency (anticipated operational occurrences) leading to abnormal radioactive releases from the facility.
B. Events of small probability with the potential for small radioactive releases from the facility.
C. Potentially severe accidents of extremely low probability, postulated to establish the performance requirements of engineered safety features and used in evaluating the acceptability of the facility site.

It is highly desirable, for both safety and economic reasons, that group A (moderate-frequency) events, such as partial loss of forced reactor-coolant flow, should result in reactor shutdown with no radioactive release from the fuel and with the plant capable of readily returning to power after corrective action. Analysis and evaluation of these moderate-frequency conditions offer the opportunity of detecting and correcting faults in a particular plant design that might otherwise lead to more serious failures. Safety is certainly enhanced if all those events that can be identified as having a reasonable chance of occurring are shown to be covered by features designed to preclude and to prevent their occurrence and significant damage.

The second group of events, such as a complete loss of forced reactor-coolant flow or partial loss of reactor coolant from small breaks or cracks in pipes, must be shown to present minimal radiological consequences. The actual occurrence of such accidents may, however, prevent the resumption of plant operation for a considerable time because of the potential for failure of fuel-particle coating and the resulting requirement for replacement and cleanup.

Evaluation of these accidents must show that under accident conditions the engineered safety features and containment barriers function effectively to eliminate (or reduce to an insignificant level) the potential for radioactive releases to the environment. In this way, assurance is gained that these unlikely events would lead to little or no risk to public health and safety. These studies also show the effectiveness of safety features designed into the facility to cope with unlikely accidents and show the margins of safety that exist in the design by indicating the type of failures that can be accommodated without raising safety concerns.

To provide additional defense in depth, extremely unlikely accidents of the third group are postulated in spite of their low probability and the steps taken to prevent them. One of these hypothetical accidents is the loss of reactor coolant resulting in system depressurization.

Each of these accidents could result in damage to the fuel-particle coating and the release of radioactive material from the reactor fuel. A portion of this radioactive material could be transported through leakage paths in the containment barriers, and some portion of it could leak out into the environment. Each type of accident is analyzed to establish that adequate safety features have been engineered into the plant, in the form of passive barriers or active systems, to limit the consequences of a release of fission products from the reactor fuel, and to show that the maximum radiological doses would not exceed the values specified in 10 CFR 100, even under highly pessimistic assumptions.

2.2.1.2 Analysis Parameters

For the analysis parameters of the reference-plant HTGR, see Section 15.1.2 of Reference 1.

2.2.1.3 Trip Settings

For the safety-related trip settings of the reference-plant HTGR, see Section 15.1.3 of Reference 1.

2.2.1.4 Radiological Parameters

For the radiological parameters of the reference-plant HTGR, see Section 15.1.4 of Reference 1.

2.2.1.5 Computer Programs

For the computer programs used in the safety analysis of the reference-plant HTGR, see Section 15.1.5 of Reference 1.

2.2.2 GROUP A EVENTS

For the detailed safety analysis of Group A events, see Section 15.2 of Reference 1.

2.2.3 GROUP B EVENTS

For the detailed safety analysis of Group B events, see Section 15.3 of Reference 1.

2.2.4 GROUP C EVENTS

For the detailed safety analysis of Group C events, see Section 15.4 of Reference 1.

2.3 ENVIRONMENTAL CONSIDERATIONS

2.3.1 SUMMARY ASSESSMENT

The thermal effluent from the HTGR is less than that from the reference light-water reactor (LWR) because the HTGR plant has a higher thermal efficiency than the LWR. The chemical effluents are similar in kind and quantity to those from the reference LWR. The normal-operation radiological releases are such that the impacts are similar to those from the reference LWR, although there are specific differences in the relative amounts of various isotopes released. In summary, therefore, the HTGR impacts are very similar to those from the reference LWR, and there should be no impediment to HTGR licensing because of the environmental impacts of routine releases.

2.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM (RG 4.2/3.2)

The Fulton nuclear power station was selected as the reference plant design to provide quantified data on effluent releases from an HTGR plant. This design has been reviewed by the NRC staff, and a final environmental statement was issued in April 1975 (Ref. 6). The fuel for the Fulton plant is highly enriched uranium-thorium fuel and, as such, differs from the fuel considered here (see Chapter 2). Variations in the fuel used, however, are not expected to affect, significantly, the environmental impacts associated with the operation of the plant. The sections that follow provide the data base for the environmental assessment of the Fulton design, comment on the fuel variations and their possible effects on the effluent source term, and compare the HTGR with the reference LWR with a 30,000-MWd/MT burnup, once-through cycle. The effluents are normalized to 1,000 MWe to facilitate comparison.

2.3.3 STATION LAND USE

Comparison of various sites for LWRs shows that there is a wide variation in land requirements. This variation results from differences in specific site characteristics and specific plant design features. Similar differences would be expected for various HTGR plant designs and sites.

The land area committed for the plant structures and major components may be somewhat different for HTGRs than for LWRs, but again specific plant-related and site-related factors are more important. It is therefore concluded that the areas required for the various categories of land use (total land area required, disrupted area, and area committed) are not significantly different for HTGRs and for LWRs. Site-specific and plant-specific factors are much more important to land use.

2.3.4 STATION WATER USE (RG 4.2/3.3)

The reference design is assumed to use a closed-loop cooling-water system with natural-draft cooling towers for heat rejection, similar to that assumed for the reference LWR. As shown in Table 2-7, the maximum and average rates at which makeup is required is about 8,900 and about 5,300 gpm, respectively, for 1,000 MWe operation. In comparison, the reference LWR requires 11,500 gpm and 6,800 gpm, respectively.

2.3.5 HEAT-DISSIPATION SYSTEM (RG 4.2/3.4)

A 1,000-MWe HTGR plant will reject about 1,520 MWt of waste heat, mainly to the atmosphere. Any of several types of heat-dissipation systems may be used, depending on site conditions and other factors. One of the more commonly used is a wet natural-draft cooling tower. This type of system, with freshwater makeup, was assumed for this report.

A typical natural-draft cooling tower for a 1,000-MWe HTGR unit would have a single shell with a height of about 500 feet and a maximum diameter of about 380 feet. Heat is dissipated to the atmosphere by a combination of evaporation and sensible-heat transfer. Although evaporation predominates, the balance between the two modes of heat transfer depends on air temperature and humidity. The average rate of water use, therefore, varies from month to month. Blowdown is required to limit the concentration of solids in the circulating water. For the reference plant discussed herein, a maximum concentration factor of 5 is used, although other values are frequently found. Design data for the heat-dissipation system are shown in Table 2-7 for a site in the north central United States. Circulating water will be periodically chlorinated to control algae and other slime-forming microorganisms. Typically, chlorine is added as required to achieve a free residual chlorine content of 0.5 to 1.0 ppm for 1 to 2 hours per day. The cooling-tower blowdown may have a small residual chlorine content during periods of chlorination.

2.3.6 RADWASTE SYSTEMS AND SOURCE TERMS

2.3.6.1 Source Term (RG 4.2/3.5.1)

In the HTGR, radioactive material is produced by fission and by neutron activation of constituents of the primary helium coolant. Fission products escape through the pyrolytic carbon coatings into the graphite of the fuel elements and then diffuse into the primary helium coolant. Tritium is present in the coolant from ternary fissions and as a result of neutron reactions with the helium-3 and lithium-6 impurities present.

The design fuel for the Fulton power plant is high-enrichment uranium/thorium fuel and as such differs from the medium-enrichment uranium fuel used as the reference fuel for this study. The fuel-element technology for the medium-enrichment fuel is similar to that for the high-enrichment fuel, the primary differences being that the fissile kernel is increased from 200 to 350 micrometers in diameter, the coat-thickness is kept approximately constant, and the fuel-rod diameter is decreased by about 25%. The composition of the fuel-particle coatings, the graphite, and the rod matrix materials are not changed for any of the medium- or high-enrichment uranium fuel cycles. The thermochemical reactions between fission products and coating materials are, however, somewhat different for the medium-enrichment uranium fuel, and the source term is expected to be different.

The data base for medium-enrichment fuel has not been completed at present (see Section 2.4.2.3, item d). Preliminary data, however, (Ref. 4) indicate that the release of gaseous radionuclides should be about the same in both high- and medium-enrichment uranium fuel. The releases of cesium isotopes should be essentially the same, but the release of silver-110m may be about seven times higher in the case of medium-enrichment uranium fuel.

Solid fission products adhere to internal reactor component surfaces and constitute one of the sources contributing to occupational exposure during maintenance operations. The increase in silver-110m release in the case of the high-enrichment fuel is not significant since the predominant isotope in plateout activity, by far, is cesium-137 for both medium- and high-enrichment fuels.

The relative equilibrium activities in the primary circuit for cesium-137, cesium-134, and silver-110m are shown below for the case of medium-enrichment fuel.

Isotope	Activity
Cesium-137	130,000
Cesium-134	5,500
Silver-110m	1,800

Fission products and tritium are partially removed from the coolant in the helium-purification system, where iodines, tritium, and solid fission products are removed by adsorption and end up as liquid or solid waste. Noble gases are stripped, held up for decay, and released to the atmosphere at specified activity levels.

Solid fission products in the primary helium coolant adhere to internal reactor-core component surfaces and constitute the source for plateout activity. This activity may find its way to the environment as liquid waste from component decontamination operations. It is also one of the sources contributing to occupational exposure during maintenance operations.

Noble gases and iodines in the primary coolant can contaminate the containment building and service building by direct leakage of primary helium and secondary coolant, respectively. This activity is released to the environment from the containment during purging operations and from the service building through continuous venting.

The secondary coolant system can become contaminated with radioactivity by two routes: by the diffusion of tritium through the tube walls of the steam generator and by a possible helium leak in the reheater section of the steam generator, where the pressure of the primary coolant is higher than that of the secondary or steam system. The activity in the secondary side contributes to the liquid- and gaseous-waste inventory through leakage into the service and turbine buildings and intentional release from the main-condenser steam jet air ejector.

The sections that follow discuss the radioactive effluent paths to the environment from plant operations and the radioactivity expected to be released annually. The source terms were calculated by the RAD C code developed by the General Atomic Company and modified by the NRC staff. The principal parameters used in the source-term calculations are given in Table 2-8.

2.3.6.2 Liquid-Radwaste System (RG 4.2/3.5.2)

The flow chart of the liquid-radwaste system is shown in Figure 2-3; the estimated annual release of radionuclides in liquid effluents is shown in Table 2-9.

As shown in Figure 2-3, liquid wastes from the containment and service building drains are collected in sumps and transferred to liquid-waste storage tanks. From there, depending on the radioactivity level, they are routed either to the cooling-tower blowdown or through the liquid-waste processing train. Liquid wastes from the radiochemistry laboratory, the helium-purification system, the gas-recovery system, and the decontamination system are collected in holdup tanks; they are subsequently placed in containers for solidification and storage before shipment off the site.

Liquid wastes from contaminated showers and laundry are collected, clarified, and routed through the liquid-waste processing train. Liquid leakage in the turbine building is collected in a sump, filtered, and routed to an evaporator. The liquid evaporated in the evaporator is vented to the atmosphere.

2.3.6.3 Gaseous-Radwaste System (RG 4.2/3.5.3)

The flow chart of the gaseous-radwaste system is shown in Figure 2-4; the estimated annual release of radionuclides in gaseous effluents is shown in Table 2-10.

As shown in Figure 2-4, the principal sources of gaseous radwaste are (a) the gaseous wastes stripped from the primary coolant in the helium-purification system; (b) the direct leakage of helium in the containment and service buildings and subsequent venting of the buildings; (c) leakage of the contaminated secondary steam into the turbine building and subsequent venting; (d) ejection of radioactive gases from the main condenser air ejector; and (e) leakage of contaminated liquid in the turbine building and subsequent evaporation and venting.

In a more recent design there is provision for storing noble gases from the helium-purification system in charcoal-loaded tanks. This provision would reduce the krypton-85 activity release to the environment from 3,607 Ci/yr, as shown in Table 2-11, to 10 Ci/yr, and the total noble-gas release to 53 Ci/yr, with a corresponding reduction in impact in terms of doses to the skin and whole body.

2.3.6.4 Solid-Radwaste System (RG 4.2/3.5.4)

The solid wastes generated during plant operation are packaged in 55-gallon drums for subsequent offsite disposal. These wastes consist of the following:

1. Radioactive liquids from the gas-recovery system, decontamination system, radiochemistry laboratory, and contaminated laundry and shower drains mixed with a suitable adsorber (cement or urea-formaldehyde)
2. Dry contaminated materials such as paper, plastic film, tape, clothing, small tools, air-filter elements, etc.
3. Spent titanium sponge from the hydrogen-getter units of the helium-purification system
4. Spent radioactive-waste demineralizer resins, activated charcoal, and soda-lime absorbent from the radioactive-gas-recovery system mixed with solidifier.

Approximately 320 drums (55-gallon) containing 900 curies of low-specific-activity waste and 17 drums (55-gallon) of titanium sponges with approximately 12,000 curies of activity are expected to be generated annually.

In addition, some 108 reflector blocks in shipping casks that may contain a total radioactivity of 6,000 curies, including carbon-14, are expected to be shipped each year. All containers will be packaged and shipped to licensed burial grounds in accordance with the regulations of the NRC and the Department of Transportation.

2.3.6.5 Comparison of HTGR and Reference LWR Effluents

Tables 2-11 and 2-12 show the estimated annual releases of gaseous and liquid effluents from the reference Fulton nuclear power plant and the reference LWR. Both plants have been normalized to 1,000 MWe for the comparison.

In comparing solid wastes from the HTGR and the LWR reference plants, it is estimated that approximately 1,050 drums (55-gallon) of low-specific-activity waste will be shipped off the reference LWR site.

2.3.7 EFFECTS OF OPERATION OF THE HEAT-DISSIPATION SYSTEM (RG 4.2/5.1)

The heat-dissipation system is similar to that for the reference LWR; hence, the impacts will be qualitatively similar. The amount of heat dissipated at 1000-MWe power-generation level is 5.2×10^9 Btu/hr as compared to 6.7×10^9 Btu/hr for the reference LWR; the impacts will thus be proportionally reduced. The HTGR would, therefore, have some advantage over the reference LWR insofar as the impact of the heat-dissipation system affects licensability.

2.3.8 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS (RG 4.2/5.2)

The dose contributions of radionuclides in HTGR liquid pathways are presented in Table 2-13. The adult whole-body dose is higher than that from the reference LWR by a factor of 1.6, and the critical organ dose is lower by a factor of 0.35. The contributions to critical dose from noble-gas releases and releases of radioiodines and particulates are presented in Tables 2-14 and 2-15, respectively. These doses are also generally lower, the doses from iodine being much lower, than the corresponding doses from the reference LWR. The HTGR values are within Appendix I, 10 CFR 50, guidelines (applicable to LWRs) and therefore should not present any difficulties in licensing.

2.3.9 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

The largest volumes of chemical wastes are from cooling-tower operation. These and other chemical wastes are from similar operations and are similar to those from the reference LWR. The impacts are, therefore, similar both in kind and in magnitude to those from the operation of the reference LWR.

2.3.10 OCCUPATIONAL EXPOSURE

Based on the NRC review of the Safety Analysis Report for the reference nuclear power plant, it has been determined that HTGR-related individual occupational doses can be maintained within the limits of 10 CFR 20. It is also felt that, with implementation of NRC Regulatory Guide 8.8, the total occupational dose for the plant could be less than the estimated 450 man-rem/yr-unit that is based on the operating experience of LWR plants. The use of medium-enriched uranium fuel (rather than the highly enriched uranium fuel to be used for the Fulton nuclear power plant) may result in some adverse changes in occupational exposure because of possible differences in the quantities and isotopic distributions of plated-out radioactivity and possible differences in plant maintenance operations. This effect cannot be quantified because the source term for medium-enriched uranium fuel, especially for solid fission products, has not as yet been confirmed. As indicated, however, in Section 2.3.6.1, the adverse changes are not expected to be significant.

Table 2-7. Heat-dissipation system design data for the reference HTGR plant (wet natural-draft cooling tower)

Heat-dissipation rate (maximum full power), Btu/hr	5.2×10^9
Evaporation and drift (maximum full power), gpm	8,900
Evaporation and drift (annual average), gpm	5,300
Blowdown (maximum), gpm	2,300
Blowdown (annual average), gpm	1,300

Table 2-8. Principal parameters and conditions used in calculating the annual releases of radionuclides in the reference HTGR plant effluents

Defective fuel,[a] %	0.5
Active helium inventory, lb	2.07×10^4
Iodine plateout factor, % per pass	40
Plateout activity decay time, days	90
PCRV leak rate, lb/yr	760
Primary to secondary system leak rate, lb/yr	36.5
Steam flow rate to turbine, lb/hr	8.05×10^6
Steam leakage to turbine building, lb/hr	1,700
Helium leakage to service building, lb/hr	10
Time required for refueling, days	20
Volume of helium transferred to fuel handling system, scf	1.46×10^4
Volume of helium processed by refuel purge, scf	1.73×10^6
Helium-purification flow rate, lb/hr	2.07×10^3
Helium-purification system decay time for Kr and Xe, days	66

	Decontamination factors				
	I, Br	Cs, Ru	Mo, Te	Y,	Other
Air ejector	2×10^3				
PCRV concrete	1×10^2				
Liquid-waste-purification system	1×10^2	1×10	1×10^3	1×10	1×10^2

[a]This value is considered to be constant and corresponds to 0.5% of the operating power equilibrium fission source term.

Table 2-9. Estimated annual release of radionuclides in liquid effluents from one 1,000-MWe HTGR unit[a]

Nuclide	Radioactivity (Ci)	Nuclide	Radioactivity (Ci)
Iron-55	0.00003	Tellurium-129m	0.0002
Selenium-83m	0.00004	Tellurium-129	0.0002
Selenium-84	0.00013	Tellurium-131	0.00003
Bromine-84	0.00004	Iodine-132	0.00002
Bromine-85	0.0002	Tellurium-133m	0.00003
Rubidium-88	0.0003	Tellurium-133	0.00003
Rubidium-89	0.0001	Tellurium-134	0.00004
Strontium-89	0.0001	Cesium-134	0.016
Rubidium-90	0.00021	Iodine-136	0.00013
Strontium-90	0.00080	Cesium-137	0.031
Yttrium-90	0.0064	Barium-137m	0.029
Rubidium-91	0.0002	Cesium-138	0.00003
Yttrium-91	0.0002	Cesium-139	0.00003
Strontium-94	0.00002	Cesium-140	0.0001
Tellurium-127m	0.00014	Samarium-151	0.00003
Tellurium-127	0.00014		
Total	<0.1		

[a]Fulton nuclear power plant.

Table 2-10. Estimated annual release of radionuclides in gaseous effluents from the reference HTGR plant[a]

Nuclide	Source term (Ci/yr)					
	Waste-gas purification	Steam jet air ejector	PCRV	Turbine building	Service building	Total
Krypton-83m	--	2	--	--	2	4
Krypton-85m	--	3	--	--	3	6
Krypton-85	3,607	--	--	--	--	3,607
Krypton-87	--	3	--	--	4	7
Krypton-88	--	5	--	--	6	11
Krypton-89	--	1	--	--	2	3
Krypton-90	--	1	--	--	1	2
Xenon-133	--	2	4	--	2	8
Xenon-135m	--	2	--	--	2	4
Xenon-137	--	1	--	--	1	2
Xenon-138	--	1	--	--	1	2
Total noble gases	3,607	21	4	--	24	3,656
Iodine-131	--	--	--	--	--	--
Iodine-134	--	--	--	0.0001	--	0.0001
Iodine-136	--	--	--	0.0003	--	0.0003
Total iodines	--	--	--	0.0004	--	0.0004
Tritium	--	--	--	80	--	80

[a]Fulton nuclear power plant.

Table 2-11. Gaseous radioactive effluents from the reference HTGR plant[a] and the reference LWR plant

Nuclide	Radioactivity released (Ci/yr)	
	HTGR	LWR
Krypton-83m	3.5	1
Krypton-85m	6.0	11
Krypton-85	3,607	380
Krypton-87	8.0	2
Krypton-88	12.0	14
Krypton-89	2.5	1
Krypton-90	1.5	
Xenon-131m	--	44
Xenon-133m	--	80
Xenon-133	8.0	7,200
Xenon-135m	3.5	1
Xenon-135	6.0	50
Xenon-137	2	
Xenon-138	2	
Total noble gases	3,660	7,786
Iodine-131	--	0.05
Iodine-132	--	0.06
Iodine-134	0.0001	--
Iodine-135	0.0002	--
Iodine-136	0.0003	
Total iodines	0.0006	0.11
Tritium	78	580
Carbon-14	--	6
Particulates	--	0.05

[a]Fulton nuclear power plant.

Table 2-12. Liquid radioactive effluents from the reference HTGR plant[a] and the reference LWR

Nuclide	Radioactivity released (Ci/yr) HTGR	LWR
Bromine-82	--	0.00007
Bromine-83	--	0.0001
Bromine-84	0.00004	--
Bromine-85	0.0002	--
Selenium-84	0.0001	--
Rubidium-86	--	0.00004
Rubidium-88	0.0003	--
Rubidium-89	0.0001	--
Strontium-89	0.0001	0.0002
Strontium-91	--	0.00006
Strontium-90	0.0008	--
Strontium-94	0.00002	--
Yttrium-90	0.0064	--
Yttrium-91	--	0.0001
Yttrium-91m	--	0.00002
Zirconium-95	--	0.00002
Niobium-95	--	0.00002
Molybdenum-99	--	0.0003
Technetium-99m	--	0.0003
Ruthenium-103	--	0.00002
Rhodium-103m	--	0.00002
Tellurium-125m	--	0.00001
Tellurium-127m	0.00014	0.0001
Tellurium-127	0.00014	0.0002
Tellurium-129m	0.00017	0.0005
Tellurium-129	0.00017	0.0003
Tellurium-131	0.000025	0.0001
Tellurium-132	--	0.01
Tellurium-133m	0.000035	--
Tellurium-133	0.000026	--
Tellurium-134	0.000034	--
Iodine-130	--	0.0004
Iodine-131	--	0.14
Iodine-132	0.000017	0.01
Iodine-133	--	0.1
Iodine-134	0.000043	0.00007
Iodine-135	--	0.02
Iodine-136	0.00013	--
Cesium-134	0.015	0.01
Cesium-134m	--	0.00003
Cesium-136	--	0.005
Cesium-137	0.031	0.01
Cesium-138	0.000035	0.00002
Cesium-139	0.000026	--

Table 2-12. Liquid radioactive effluents from the reference HTGR plant[a] and the reference LWR (continued)

Nuclide	Radioactivity released (Ci/yr) HTGR	LWR
Cesium-140	0.000086	--
Barium-137m	0.029	0.01
Barium-139	--	0.00004
Barium-140	--	0.0002
Lanthanum-140	--	0.0001
Cerium-141	--	0.00002
Cerium-143	--	0.00001
Praseodymium-143	--	0.00002
Cerium-144	--	0.00005
Praseodymium-144	--	0.00002
Neodymium-147	--	0.00001
Sodium-24	--	0.0001
Phosphorus-32	--	0.00002
Phosphorus-33	--	0.0001
Cerium-51	--	0.0003
Manganese-54	--	0.00006
Manganese-56	--	0.001
Iron-55	--	0.0003
Iron-59	--	0.0002
Cobalt-58	--	0.003
Cobalt-60	--	0.0004
Nickel-65	--	0.00002
Niobium-92	--	0.00006
Tin-117m	--	0.00002
Tungsten-185	--	0.00002
Tungsten-187	--	0.0005
Neptunium-239	--	0.0002
All others	--	0.0001
Tritium	--	270.0

[a]Fulton nuclear power plant.

Table 2-13. Dose contributions of radionuclides in liquid effluents

Nuclide	Dose contribution (%)	
	Adult whole body	Critical organ
Cesium-134	44	37
Cesium-137	54	62
Others	2	1
Ratio of HTGR dose to LWR reference case	1.6	0.35

Table 2-14. Dose contributions due to releases of noble gases

Nuclide	Dose contribution (%)	
	Whole body	Skin
Krypton-83m	a	a
Krypton-85m	2	a
Krypton-85	15	91
Krypton-87	12	2
Krypton-88	44	3
Krypton-89	10	1
Krypton-90	6	1
Xenon-133m	a	a
Xenon-133	1	a
Xenon-135m	3	a
Xenon-135	3	a
Xenon-137	1	a
Xenon-138	4	a
Ratio of HTGR dose to reference LWR case	0.16	1.01

[a]Less than 1%.

Table 2-15. Dose contributions from releases of radionuclides and particulates

Nuclide	Dose contribution (%)	
	Infant thyroid	Child thyroid
Iodine-134	[a]	[a]
Iodine-135	[a]	[a]
Tritium	100	100
Ratio of HTGR dose		
to LWR reference	0.0008	
case	0.0008	0.004

[a]Less than 1%.

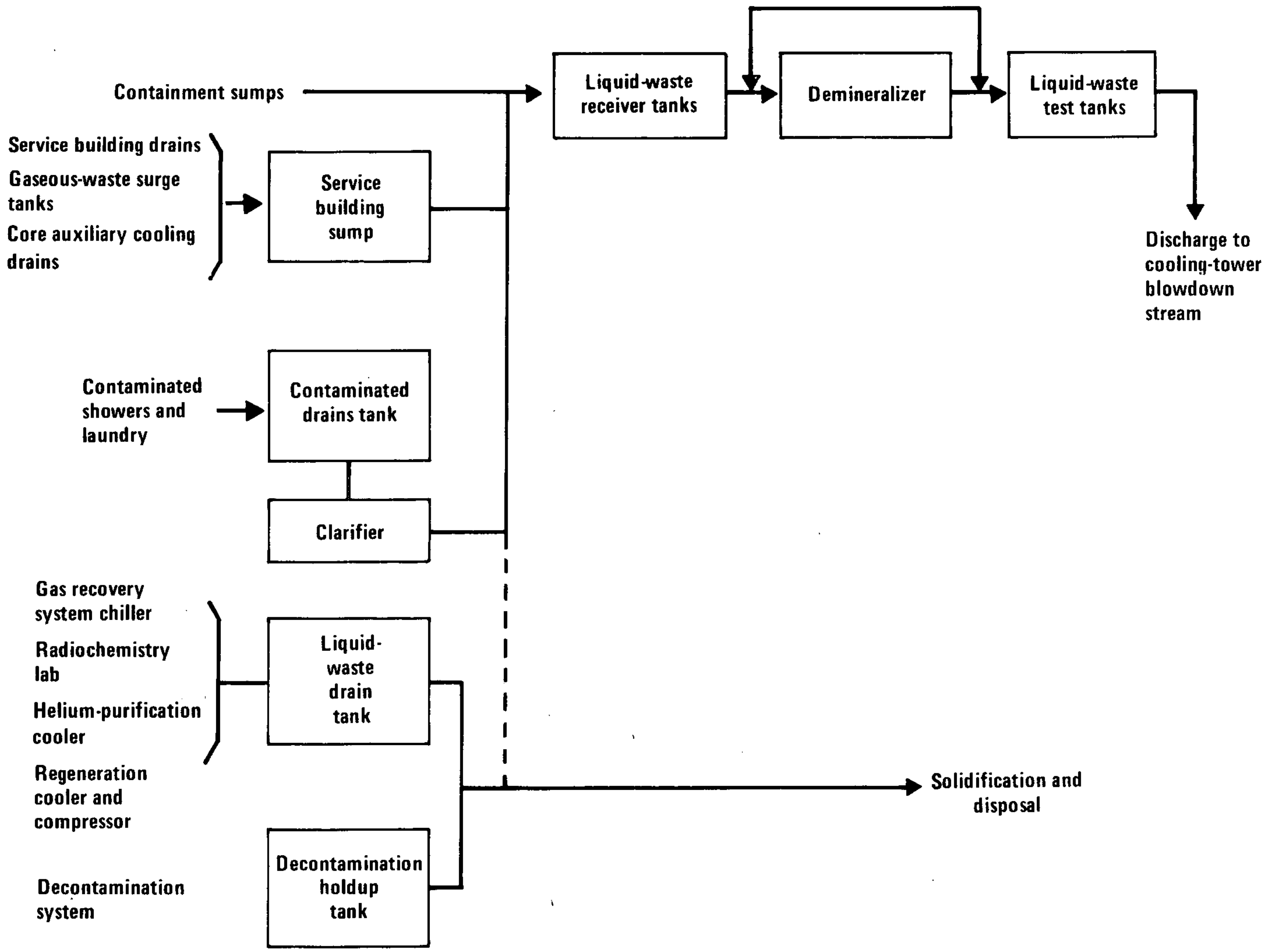


Figure 2-3. Flow chart of the liquid-radwaste system proposed for the Fulton Nuclear Power Station.

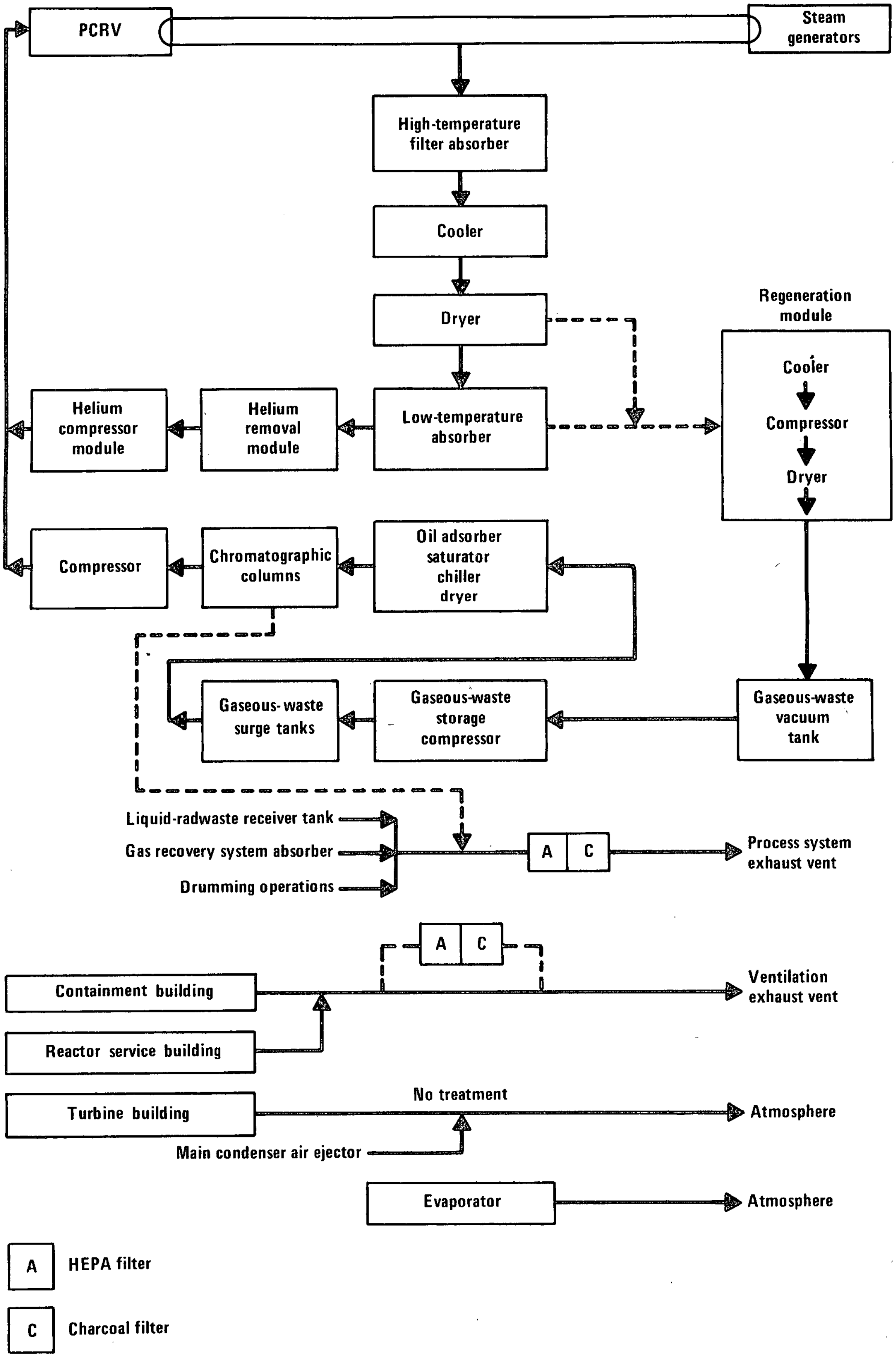


Figure 2-4. Flow chart of the gaseous-radwaste system proposed for the Fulton Nuclear Power Station.

2.4 LICENSING STATUS AND CONSIDERATIONS

2.4.1 INTRODUCTION

Experience with licensing HTGRs in the United States is not extensive but is second to that with present-generation reactors (LWRs). Two HTGRs have been licensed and built in the United States, including a 40-MWe prototype at Peach Bottom 1 and a 330-MWe demonstration plant at Fort St. Vrain. The safety analysis reports for two commercial-size plants (the Summit and Fulton stations) had been reviewed by the NRC and the Advisory Committee on Reactor Safeguards (ACRS), and limited work-authorization permits had been issued before the plants were canceled in 1975. At the time of cancellation, there were several outstanding licensing issues to be resolved before the issuance of an operating license. These included the in-service inspection program, anticipated transients without scram, design verification and support for prototype components, structural-graphite design criteria, core seismic criteria, and preoperational vibration assessment.

General Atomic later submitted a standard safety analysis report (GASSAR) for a reference commercial-size nuclear steam supply system (Ref. 1). Additional issues were identified in the NRC review of GASSAR, including thermal-analysis codes for core cooling and the selection of design-basis accidents. Activities associated with resolution of the key outstanding issues are being supported by the Department of Energy (DOE) and a utility group named Gas-Cooled Reactor Associates. A pre-application review by the NRC has been requested. There is no reason to suspect that any of these issues are not amenable to resolution. The overall licensing outlook is very favorable.

The philosophy under which all HTGRs are reviewed for licensing in the United States is that a comparable level of safety must be established for all reactor types, with the full recognition that the great majority of licensing criteria were developed for LWRs. The implementation of this philosophy in the establishment of HTGR criteria has taken the following three forms with respect to previously existing criteria: direct adoption, suitable adaptation, and recognition of the need for, and development of, specialized HTGR criteria. Fortunately, direct adoption of the existing criteria is possible in the great majority of instances and provides the best means for ensuring a comparable level of safety for the HTGR. Examples of direct adoption are numerous, ranging from criteria established by the Institute of Electrical and Electronics Engineers to most of the NRC regulatory guides. A list of regulatory guides applicable to HTGRs was presented at the 1974 Gatlinburg Conference on Gas-Cooled Reactors (Ref. 7). Almost all regulatory guides except those that deal with specific aspects of the nuclear steam supply systems or with accident analyses apply directly to HTGR licensing.

Three types of HTGR have been considered for licensing in the United States (all from a single manufacturer, the General Atomic Company of San Diego, California): the 40-MWe Peach Bottom Unit 1 reactor (Ref. 8), which was operated for 7 years by the Philadelphia Electric Company until 1974; the 330-MWe Fort St. Vrain reactor (Ref. 9), which is currently undergoing power-ascension testing by the Public Service Company of Colorado; and the "large" HTGR concept of the 700- to 1,000-MWe Study (Ref. 10), which was developed more fully during 1973 to 1975 when the reviews of the Summit and Fulton applications (Refs. 11 and 12) of the Delmarva Light & Power Company and the Philadelphia Electric Company, respectively, reached the stage where reports from the Advisory Committee on Reactor Safeguards were issued (Refs. 13 and 14).

The design parameters for Peach Bottom, Fort St. Vrain, GASSAR-6 (Reference 15), and the lead plant are compared in Table 2-16.

During the above licensing actions, the licensing considerations to be dealt with in progressing toward commercial status are the following:

Licensing action	Licensing considerations
Peach Bottom Unit 1, 1961 to present	1. Ceramic-core design 2. Fission-product transport and plateout 3. Delineation of HTGR hazards
Fort St. Vrain, 1966 to present	4. Prestressed-concrete reactor vessel 5. Retention of fission-products within coated fuel particles 6. Detailed definition of depressurization and core-heatup accidents
1,000-MWe Study, 1969	7. Reactor-containment requirements 8. Integrated primary coolant system
Summit and Fulton stations, 1973-1975	9. Containment-backpressure requirements 10. Performance of the emergency core-cooling system, including air ingress 11. Testing requirements of primary mechanical components 12. Steam-generator design 13. Vendor quality assurance 14. Decay heat rate
GASSAR, 1974 through 1977	15. Conformance of application with HTGR edition of Standard Format 16. Revised seismic and structural analysis 17. Detailed review of fission-product release from failed particle coating

During the power-ascension testing of the Fort St. Vrain reactor, power/temperature oscillations were observed. The first oscillations were observed on October 31, 1977, and were indicated by fluctuations in the steam temperature as observed by the control-room instruments. The oscillations were detected by nuclear channels, core-region outlet temperatures, steam-generator gas-inlet temperatures, and steam-generator-module steam temperatures.

The oscillation characteristics are as follows:

1. Outlet thermocouples for most refueling regions, steam generators, and nuclear detectors experience some degree of irregular and complex oscillation; the average reactor power remains essentially constant during the oscillations.
2. The period ranges from 5 to 20 minutes, with a 10-minute period characteristic for the northwest quadrant of the core.
3. Initiation and major amplitude occur in the northwest quadrant of the core: regions 20, 32-37, nuclear channels IV and VI, and steam-generator modules B-1-4, B-1-5, B-1-6, and B-2-6.

Short-range plans to better understand the oscillations include the installation of diagnostic instrumentation to detect the actual rates of flux and temperature change and any core motion, correlation of all oscillation events, and noise analysis. Long-term plans include the addition of instruments to the control-rod drives and more in-core instrumentation.

Pending resolution of the oscillation issue, the Fort St. Vrain nuclear power station is restricted to operation below 70% of rated power.

2.4.2 RESPONSES BY THE GENERAL ATOMIC COMPANY TO NRC QUESTIONS

The NRC recently submitted to the DOE (Ref. 16) a list of 29 questions and comments on 8 topics for those developing licensing and safety documentation for the proposed commercial HTGR lead-plant design. The NRC questions and comments were meant to reflect the current status of safety-related issues pertinent to licensing review of a commercial HTGR and are not to be considered as complete or definitive statements of anticipated licensing needs; they are presented in this section together with responses prepared by the General Atomic Company.

The topics covered by the NRC questions have an extensive history and are currently the subjects of DOE-funded development programs. The results of these programs are being used as inputs to a series of NRC review programs. Since these programs are currently active, their status is continually evolving and may be followed with the least risk of confusion by reference to the routine progress reports of the DOE HTGR Generic Technology Program and to the minutes of NRC generic review meetings on this subject.

Programs to verify the DOE-funded graphite work will be necessary for several more years. The total funding of these programs in the future is not expected to exceed $16 million.

2.4.2.1 Graphite as Structural Material

The NRC questions on graphite were as follows:

1. Identify the mechanical design requirements, including loading combinations of all graphite structures used in the reactor under normal, upset, emergency, and faulted conditions in the plant.
2. Provide and justify the design criteria for graphite structures under normal, upset, emergency, and faulted conditions in the plant. Discuss how these criteria accommodate considerations of secondary stress, thermal shock, fatigue, and corrosion.
3. What parameters are deemed to be significant in the graphite corrosion and what basis exists for those judgments?

a. Mechanical Design Requirements

The loading combinations used for graphite components of the HTGR are derived from those defined in the June 1978 edition of ANS-50 Policy 2.4, "Plant Design Conditions for Nuclear Power Generating Stations" (Ref. 17). The ANS-50 loading combinations are based on industry practice and NRC documents, including Regulatory Guide 1.48 and Branch Technical Position MEB-6.

b. Design Criteria for Graphite Structures

General Atomic generated and proposed a set of criteria in GASSAR, the generic safety analysis report. These were reviewed on behalf of the NRC by the Franklin Institute (Ref. 18), which made significant suggestions for changes. These suggested changes were extensively reviewed in NRC generic review meetings and subsequently by a joint subcommittee formed by the American Concrete Institute and the American Society of Mechanical Engineers (ASME) specifically for this purpose. This subcommittee's main objective is to generate a code section with consensus support. Many of the items before the subcommittee require experimental verification, which is being obtained from the DOE Generic Technology Program. Therefore the date on which the code section is issued will depend on completion of those programs. The NRC is represented on the subcommittee, but it is also planned to submit reports for NRC review via the licensing topical report format. This is scheduled for late 1979, with tentative adoption of a code by late 1980.

c. Parameters Significant in Graphite Corrosion

Experimental programs are currently in progress and will be reported as they become available. General Atomic's experimental and analytical work to date shows that oxidation under actual HTGR operating environments causes a predominantly surface attack and can be allowed for in structural analysis design by simply removing layers of surface material. Thus, it is General Atomic's position that a corrosion allowance will be made in design calculations and that the minimum safety factors required by the proposed design criteria will be available even at the end of life. Presentations on this technique were made to the NRC as part of the generic HTGR review program in 1976 and 1977 and specifically with respect to the integrity of the Fort St. Vrain core support in November 1977 and May 1978. The NRC has published minutes of these meetings.

2.4.2.2 Core Seismic Response

The NRC questions on core seismic response were as follows:

1. Provide and justify the seismic design criteria for the core and all other nonmetallic structures that support or otherwise relate to the integrity of the core.
2. Describe the seismic analysis methods for the core and related structures in conjunction with results from experimental verification programs.
3. Describe the function of any nongraphite materials in the reactor in terms of the core seismic response. Provide and justify the materials properties used for these materials in the seismic analysis.

a. Seismic Design Criteria

The design bases for fuel elements and reactor internals are established to maintain the integrity of the coolant flow geometry, to allow safe shutdown of the core, and to protect the integrity of the fission-product barriers within the core. The flow-control valves, the core lateral and lower support structure, and the graphite fuel and reflector elements define the coolant-flow geometry, while the fuel-particle coatings, fuel-rod matrix, and the graphite webs of the blocks act as barriers to the escape of fission products. The alignment of coolant holes and control-rod channels is maintained by the dowel system. Excessive rocking angles, which may cause disengagement, must be prevented.

For fuel-element and replaceable reflector graphite, the maximum principal stresses will be limited to the values listed in Table 4.2-I of GASSAR (Ref. 1) and will include adequate allowances for exposed-kernel swelling due to fuel hydrolysis and graphite strength reduction as shown. The seismically produced stresses are considered to be primary loads.

In graphite core-support components, including the core-support floor and posts and the permanent-side-reflector blocks, the maximum principal stress at a point will be limited to the values specified in Table 4.2-II of GASSAR. The effect of environment on the strength of graphite will be accounted for in the design such that the full safety factors are met at the end of reactor life. Because of the anisotropic nature and complex geometry of graphite core-support-structure components, it is considered acceptable to demonstrate, by representative testing in lieu of calculations, that the ratio of failure load to specified load is equal to or greater than the ratio of ultimate strength to allowable stresses.

b. Seismic Analysis Methods

Since a typical HTGR core can contain 8,000 blocks, a full three-dimensional model would require 48,000 degrees of freedom and would be prohibitively large. The symmetrical pattern of the core, however, lends itself to reduced models of one and two dimensions. The simplest model of the full-array core is in CRUNCH1D. This code represents a single line of blocks that is a strip of core at a single elevation. The two-dimensional version, CRUNCH2D, is a planar core layer at a single elevation. The columns of blocks are modeled in COCO, MCOCO, and COCOROD. COCO contains a single column, whereas MCOCO models the entire diametral line of columns, including side-reflector columns and spring packs. The COCOROD code contains the single COCO column with the control rod hung inside the blocks. Together these five codes provide the capability of studying seismic loads in the three-dimensional core blocks and supporting structure for three directions of earthquake motion.

The test program provides information on force, block motion, and block velocity. To obtain individual block properties, the collision dynamics and basic rocking tests were performed. The 73-block horizontal array tests provide in-plane block grouping characteristics for time-history motion, while the single-column shake test provides data on the characteristics of the column of blocks. The full-array tests provide the full-system data and characteristics of the total core. The computer codes rely on the test data for the parameter values used in the models (collision dynamics and basic rocking tests); the large-array tests have been used to verify the codes and to give information on the characteristics of the core for design purposes.

c. Function of Nongraphite Materials in Terms of Core Seismic Response

The core support and lateral-restraint structure should withstand any differential movements of the PCRV and the core, including those resulting from temperature, pressure, PCRV prestress, and creep, without interfering with the normal operation of the core. The lateral-restraint metal spring packs in conjunction with the permanent side reflector will limit seismic impact loads and deflection such that the plant can operate without interruption through an operating-basis earthquake and can safely shut down after a safe-shutdown earthquake.

The design stress-intensity values for metallic construction materials, including spring packs and plenum elements, will be extracted from Section III of the ASME Boiler and Pressure Vessel Code. Allowable stresses for metallic materials not included in the Code will be derived in a manner similar to that for Section III, Class 1, values. Where the material may creep at elevated temperatures, the allowable stresses or strains and analytical techniques will be as in Code Case 1592. The allowable stress-intensity limits for all operating conditions will be the same as those given in Article NG-3000.

2.4.2.3 Fuel Transient Response

The NRC questions on fuel transient response were as follows:

1. Provide a complete description of the conditions (thermal, mechanical, and irradiation) to which the fuel and fuel blocks will be exposed.
2. Describe the response (under the same plant conditions) of reactor materials other than the fuel that could potentially affect fuel integrity. As an example, this answer should include a discussion of the potential for blockage of the fuel coolant holes by fibrous insulation material.
3. Provide a description of the reference fuel. This description should take into account that research and development is continuing on HTGR fuel. State what design aspects and manufacturing process variables can be considered as fixed at this time and what aspects may change as the consequence of further research. Describe any effects that changes in the fuel design or process variables would have on the fuel's transient response.
4. Summarize the fuel irradiation data base supporting the reference design and the responses described in Question 3 above. Justify the use of data that were not clearly obtained with the reference type fuel.
5. Describe the basis which exists for predicting the fuel response to accidents and transients for defined but arbitrary operational histories.

a. Thermal, Mechanical, and Irradiation Conditions of Fuel Exposure

A complete description of the thermal, mechanical, and irradiation conditions for the fuel and fuel blocks under normal conditions is given in Chapter 4 of Reference 1. Additional information is provided in References 2 and 19.

The HTGR core contains some 3,000 fuel blocks, 400,000 fuel rods, and about 10^{12} fuel particles. Moreover, the fuel is loaded in segments, and each fuel region is individually orificed. Thus, it is not feasible to provide a complete description of the operating conditions for all the fuel in summary form. However, some typical, representative data can be presented.

Figure 2-5 shows the radial temperature profile in an average fuel channel under normal conditions, and Figure 2-6 shows the axial temperature distribution in a high-power fuel region. The overall fuel temperature distribution as a function of volume is shown in Figure 2-7; typical fuel temperature histories during irradiation are shown in Figure 2-8. The core volume distribution of the fast-neutron flux and the burnup of fertile and fissile fuel as a function of fuel age are shown in Figures 2-9 and 2-10, respectively. The mechanical conditions of the fuel, including its design basis and stress limits, are described in Sections 4.1 and 4.2 of Reference 1. The stress criteria are currently under investigation (see Section 2.4.2.1).

Core behavior under accident conditions depends on the particular initiating event and subsequent history, including possible actions by the plant-protection system. These accident conditions are described in Chapter 15 of Reference 1, including the calculated temperatures, power levels, and mechanical conditions of the fuel.

b. Effects of Other Reactor Materials on Fuel Integrity, Including Coolant Flow Blockage

Small debris in the primary system, such as graphite chips and pieces of insulation, can be postulated to block or restrict flow in coolant passages in the core. However, only a limited range of material sizes can be postulated to block a coolant hole because the blocking material must pass through the region flow-control-valve port. The maximum size of a single piece of debris is defined by the valve port, which is, when fully open, an approximately rectangular opening measuring 5 by 10 inches. The smallest particle that can lodge in the core-coolant passages and restrict flow must be larger than the 0.717-inch diameter of the smallest coolant hole.

The consequences of such coolant-hole blockage have been investigated, and the results are described in detail in Section 15.2.3 of Reference 1. In this analysis a range of coolant-hole blockages was investigated over a wide variation in power levels, and conservative assumptions were made on core operating conditions (e.g., blockage occurring in the highest power region, no thermal-reactivity feedback effects to mitigate the consequences, etc.).

The immediate consequence of a blockage is an increase in fuel temperature in the region of the hole. However, the temperature change is slow and is limited by the thermal properties of the graphite and of the coolant; the time constant is on the order of minutes. Some local fuel failure can be expected, with a corresponding increase in coolant activity.

Outside the core, a severe flow blockage can result in high temperatures of components in the primary-coolant pressure boundary because of hot-streak effects. A potentially worse hot-streak effect can result from the sudden unblocking of the blocked channels, resulting in the reintroduction of flow in coolant channels with abnormally high temperatures. However, analysis indicates that temperatures will remain below critical safety limits regardless of actions taken to terminate the event. In a severe blockage, it would be necessary to shut down the plant if the primary-coolant activity exceeded technical specification limits.

The analysis shows that, for the range of events considered, no release of radioactivity to the environment will occur.

Research and development that is continuing on HTGR fuel includes the following:

1. Investigation of alternative types of medium-enriched fuel kernels; examples are uranium oxycarbide, mixed thorium and uranium oxides, and uranium dioxide with zirconium carbide buffer (Refs. 20-23).
2. Development of a process whereby the fuel rods are outgassed and carbonized within the graphite fuel element--that is, cure in place (Ref. 24).
3. Development of medium-enriched fuel performance models that account for kernel migration, pressure failure of the coatings, and the reactions of silicon carbide with fission products (Refs. 20-23, 25, 26).

Research is also continuing on the formation and spheroidization of medium-enriched fuel kernels, coating technology, and reductions in particle manufacturing defects.

It is anticipated that further research may lead to the development of the uranium oxycarbide medium-enriched fuel kernel and to cure-in-place processing of fuel rods. It is also expected that improved specifications will lead to the presence of fewer defective particles in the fuel elements and to improved coatings with enhanced irradiation performance.

Data from irradiation experiments (Ref. 27) indicate that no detrimental effects should be expected for HTGR fuels experiencing load-following transients.

No change in particle design is likely to lead to an adverse effect on the transient response of the fuel since work to date has shown that fuel performance (kernel migration, pressure failure of the coating, and the reactions of silicon carbide with fission products) during normal and accident conditions is similar for a wide range of potential fuel designs (Refs. 20-23, 25, 26). In fact, the development of cure-in-place processing, improved kernel formation, spheroidization, and improved coatings are expected to reduce in-service failure and have a beneficial effect on the transient response of the fuel. Furthermore, studies are continuing on silicon-alloyed BISO particles. In addition to such advantages as lower cesium release and increased tensile strength, these particles allow heavier loadings in the reactor, reduced coating thickness and more fuel volume in the core, greater fuel-loading flexibility, and the use of more filler in the fuel rod, thus increasing thermal conductivity.

c. Description of the Reference Fuel

The reference fuel materials are medium-enrichment uranium (MEU) (about 20% uranium-235 for an MEU core) in the carbide form and fertile thorium in the oxide form. Initially, all of the fissile loading is uranium-235; however, the design of the reactor provides for the use as a feed material of recycled uranium-233, derived from thorium-232, when it becomes available.

The fissile MEU kernels of uranium carbide are TRISO coated. There is a low-density porous pyrolytic carbon buffer layer adjacent to the kernel followed by a layer of isotropic pyrolytic carbon, a layer of silicon carbide, and a final (outer) coating of pyrolytic carbon.

The fertile thorium dioxide kernels are BISO coated. There is an inner coating of low-density, porous pyrolytic carbon and an outer coating of isotropic pyrolytic carbon.

The use of different coatings on the fissile and fertile particles simplifies the separation of the fissile species during reprocessing.

The fissile and fertile fuel particles are bonded together with a carbonaceous matrix to form fuel rods. The bonding matrix consists of a graphite filler and an organic binder heat treated to outgas and carbonize the binder. The fissile and fertile particles are uniformly blended to provide the necessary uranium and thorium content in each fuel rod. Various blends are produced to provide the required heavy-metal loadings in the fuel elements. The rods are sized to give a close fit with the fuel holes drilled in the graphite hexagonal right prism; the rods are stacked in the fuel hole to make up the total fuel length in the fuel-element assembly. A more complete description of the fuel has been presented in Reference 28.

d. Fuel-Irradiation Data Base

The fuel-irradiation data base supporting the earlier high-enrichment uranium (HEU) fuel design has been described in Chapter 4 of Reference 28. In addition, a general discussion of the fuel development data base was presented in References 24 and 29.

Test plans for the initial two MEU irradiation capsules (HRB-14 and HRB-15B) were presented in recent HTGR Fuels and Core Development Program quarterly reports (Refs. 20-23). Both of these capsules included MEU oxycarbide along with several other types of kernels. A large integral test of MEU oxycarbide fuel is being planned for irradiation in capsule R2-K13, a joint experiment by General Atomic and Kernforschungs Anlage Juelich under the auspices of the umbrella agreement between the United States and the Federal Republic of Germany for cooperation in gas-cooled-reactor development.

Data for HEU fuels were usually obtained with reference-type fuel. Until such time as irradiation experiments can be completed and analyzed, and out-of-reactor tests can be performed on both irradiated and unirradiated MEU fuels, the data base for MEU fuels can be derived from data on HEU and low-enrichment uranium (LEU) fuels because the exposure conditions, fuel design, and fission-product inventory bracket the MEU conditions. There is extensive documentation for the many experiments and tests performed on HEU and LEU fuels from both HTGR and LWR fuel systems (Ref. 29).

Evaluation of kernel-migration data has shown that the migration of irradiated MEU fuel particles is less than or equal to that of unirradiated particles. Data are now being developed for MEU particles, primarily on irradiated samples. Correlation of existing data on LEU and HEU particles from both in-reactor and out-of-reactor tests show good agreement on the predictability of fission-product reactions with coating materials. It is expected that MEU fuel performance data generated in both in-reactor and out-of-reactor experiments will be predictable and consistent with those for LEU and HEU fuels (Refs. 22, 30, and 31).

e. Basis for Predicting Fuel Response to Accidents and Transients

A considerable amount of analysis and experimental work has been performed in determining HTGR fuel response to accidents and transients. The analysis work is summarized primarily in Chapter 15 of Reference 1, which considers the consequences of a wide range of reactivity transients, loss of forced circulation, steam and water ingress, earthquakes, and other events.

The basis for the calculations and predictions includes measurements of basic kinetic data, temperature coefficients, reactivity worths, power distributions, and temperatures in the HTGR critical experiments, the Peach Bottom reactor cores I and II, and in the Fort St. Vrain core. This information is summarized in Reference 3. The calculational basis for predicting fuel response is described in References 32, 33, and 34.

The basis for predicting the response fuel particles and the core graphite components under accidents and transients is summarized in References 35 and 36, respectively. Finally, the basis for the fission-product release calculations is summarized in References 37, 38, and 39.

2.4.2.4 In-Service Inspection and Testing

The NRC submitted the following questions on in-service inspection and testing:

1. State your criteria for determining the need for in-service inspection of any portion of a structure, component, or system of the primary coolant system or the primary coolant boundary; identify the portion excepted, and justify the exception.
2. Describe your plans and program for the development of in-service inspection techniques and instrumentation to meet the intent of 10 CFR Part 50.55a(g).

a. Criteria for Determining the Need for In-Service Inspection

Criteria for determining the need for in-service inspection of primary-coolant-system components, including pressure-boundary and non-pressure-boundary portions, are contained in the proposed Section XI, Division 2, of the ASME Boiler and Pressure Vessel Code, "Rules for Inspection and Testing of Components of Gas-Cooled Plants," Subsections IGB, IGC, IGG, and IGK. The categories of affected components include those required to function in support of (a) shutdown-heat removal operations, (b) the control of nuclear reactivity, (c) the detection or control of chemical ingress, or (d) a controlled primary-coolant depressurization. All components essential for these functions are candidates for in-service inspection. It is the plant owner's responsibility to determine the frequency and extent of in-service inspection in accordance with Section XI of the ASME Code.

b. Development of In-Service Inspection Techniques and Instrumentation

State-of-the-art equipment and practices are adaptable to current ASME Code requirements for component in-service inspection and testing. Development of special methods, techniques, and instrumentation for application to the HTGR is not contemplated.

2.4.2.5 Low-Probability Accidents

The NRC asked the following questions on low-probability accidents:

1. Describe the best estimate and uncertainty determinations of the consequences of selected low-probability accidents. Where applicable, the calculations performed for the Accident Initiation and Progression Analysis (AIPA) study may be used. Critical assumptions for each accident analysis should be identified. These accidents should include but not be limited to control-rod ejection, core drop, large moisture ingress combined with reactor depressurization or core heatup, depressurization areas greater than 100 square inches, depressurization combined with containment failure, and unrestricted core heatup in combination with containment failure.
2. Identify research programs that are in progress or planned that relate to critical assumptions made in the accident study. What design features or design changes provide a "fall-back" position if these research programs fail to verify the assumptions in question?
3. How is gas-cooled-reactor experience in the United States and abroad being factored into the study of low-probability accidents?
4. What nonprobabilistic criteria are being used to distinguish between design-basis accidents (Class 8) and accidents sufficiently remote that they can be excluded from the design basis (Class 9)?

5. Why is the MHFPR (maximum hypothetical fission-product release) accident, as in Summit and Fulton, which implies integrity of the PCRV liner cooling and of the secondary containment, an appropriate siting event?

a. Consequences of Selected Low-Probability Accidents

A comprehensive assessment of public risk from HTGR accidents is reported in Reference 42, the Phase II status report for the HTGR AIPA study. A preliminary assessment of a wide spectrum of initiating events was employed to identify the more important low-probability (Class 9) accident sequences. Based on the results of this evaluation, unrestricted core heatup in combination with containment failure was studied in great detail. The consequence point estimate, uncertainty ranges, and critical assumptions for each scenario are presented in Sections 4.3 and 4.4 of Reference 40. The consequences of the HTGR accident sequences are shown to be low compared with those of other nuclear power concepts.

Cases of moisture ingress and depressurization that could arise from steam-generator failures and failure of the dump and isolation system were also studied. The results, including assumptions, consequences, and uncertainty ranges, are summarized in Section 5.1 of Reference 40. Likewise, accident sequences that include PCRV depressurization and containment bypass were analyzed for assumed reheater failures. The assumptions and risk estimates are presented for a spectrum of reheater leak accidents in Section 5.2 of Reference 40.

Control-rod ejection, core drop, and depressurization areas larger than 100 square inches in the context of probabilistic risk assessment were found to be even lower risk contributors because of their estimated low probability and therefore have not been analyzed to a comparable level of detail.

b. Research Programs Related to the AIPA Study

As a result of the Phase II AIPA study, four major areas of continuing research programs have been identified, largely in an effort to reduce uncertainty bands on frequencies and consequences for Class 9 accidents. The following areas for safety research and development have been identified: (a) continued study of new initiating events and accident sequences; (b) containment-atmosphere response to accidents; (c) fission-product transport under accident conditions, including plateout; and (d) earthquake frequencies. Other ongoing programs include (a) the study of fires and other event sequences that lead to core-heatup conditions, (b) analytical modeling of important containment-response conditions under key accident sequences to reduce associated uncertainties, (c) laboratory experiments for correlation with the PADLOC code plateout models, and (d) earthquake modeling and refinement of earthquake response spectra. These areas were treated with large uncertainties in the AIPA study. However, even with such large uncertainties, the inherent safety features of the HTGR (i.e., a massive graphite core, coated fuel particles, inert coolant, and concrete PCRV) were found to limit the consequences to such a degree that no early fatalities are predicted for HTGR Class 9 accidents over a meaningful frequency range. (Details of these analyses are given in Chapter 3 of Reference 40.)

Since the upper uncertainty bands for accident consequences were already based on limiting cases, it appears that "fallback" positions may not be necessary. In fact, the results presented in Reference 40 indicate that HTGR inherent safety may permit simplifications of the design. However, should future experimental work reveal inadequacies in the critical assumptions, probabilistic risk-assessment techniques will be employed to identify the most appropriate design alterations.

c. The Use of Data from Gas-Cooled-Reactor Experience in Low-Probability Accident Analysis

The relevant system and component operating experience from Peach Bottom Unit 1 and Fort St. Vrain has been compared with the European gas-cooled-reactor and U.S. light-water-reactor experience bases to establish reliability parameters for HTGR components and systems important in the progression of accident sequences. For some systems, fossil-fired power-plant data were also considered applicable. These data were then used to quantify accident-sequence frequencies with the fault trees and event trees employed in the AIPA study of HTGRs (Ref. 40). A limited study of European gas-cooled-reactor accidents was also performed. This operating experience therefore provided quantitative input for assessing the risk of low-probability accidents.

d. Nonprobabilistic Criteria for Distinguishing Between Class 8 and Class 9 Accidents

A rational approach for identifying the key factors that distinguish Class 8 from Class 9 accidents is to use the quantitative methods of probabilistic risk assessment demonstrated in Reference 40. The traditional, well-established, nonprobabilistic methods are also employed, as they have been in the past, to distinguish between Class 8 and Class 9 accidents. This includes the assumptions that for Class 8 accidents no more than one "initiating event" occurs during any accident sequence and that no more than one "single failure" occurs in the systems required to respond to any initiating event. Sequences with simultaneous "initiating events" and multiple "single failures" are included in Class 9 accidents.

e. Appropriateness of the MHFPR Accident in Establishing Siting Safety

Part 100 of 10 CFR requires that the fission-product-release hazard for siting calculations not be exceeded by those from any accident considered credible. Both the Summit and Fulton stations met this 10 CFR 100 condition for the nonmechanistic MHFPR release treatment used in their license applications. However, the "nonmechanistic" release assumptions were based on precedents established in the licensing of LWRs (Ref. 41). This approach has resulted in very conservative siting requirements that do not recognize many of the unique safety features of the HTGR.

Since the Summit and Fulton applications, greater understanding of reactor safety has emerged as a result of the Reactor Safety Study (Ref. 42) for LWRs and the AIPA study for HTGRs. In both of the studies, the consequences of Class 9 accidents were evaluated. These Class 9 studies included consideration of containment and the continued operation of the cooling system for the PCRV liners. Therefore, in the future development of the licensing process, it may be inappropriate for HTGRs to be licensed with the same nonmechanistic precedents established for LWRs.

2.4.2.6 Containment Requirements

The NRC questions on containment requirements were as follows:

1. What are the criteria for the selection of the design bases for the containment system?
2. What are the containment design bases?
3. How would the evolution of carbon monoxide from oxide or oxycarbide fuels during an unrestrained core heatup accident and its combustion in the containment impact the containment pressure and temperature?

a. Criteria for Selecting Design Bases for the Containment System

The Fort St. Vrain gas-cooled reactor was built with a non-leaktight containment. For the larger HTGRs (Fulton and Summit designs), however, General Atomic agreed to the requirement for a conventional leaktight LWR-type containment to obtain approval of the large HTGR by the Advisory Committee on Reactor Safeguards. Based on the analysis provided in Reference 42, this appears to be an overly conservative approach that should be reevaluated. Analysis of a depressurization of the PCRV, limited to 100 square inches of blowdown area, was included as a design-basis depressurization accident (DBDA), in accordance with an Atomic Energy Commission requirement. To demonstrate site acceptability, a siting-event source term equal to the initial activity released by the DBDA plus a time-delayed release of the LWR release fractions given in Reference 43 was chosen. With the conventional containments, the resulting doses were a small fraction of those specified in the 10 CFR 100 guidelines.

The selection criteria for the design-basis events were chosen to meet the intent of the General Design Criteria of Appendix A to 10 CFR 50. The principal General Design Criteria dealing with containment design are criteria 11, 16, 17, 18, 19, and 38.

b. Containment Design Bases

The containment design bases that were employed in the Fulton and Summit plants are discussed fully in Section 6.2.1 of the preliminary safety analysis reports (PSARs) (Refs. 43 and 44). Briefly, the requirements were that the containment be designed to be leaktight (i.e., the containment leak rate not exceed that assumed in the siting-event dose calculations) and that the containment leak rate not be exceeded under the safe-shutdown-earthquake conditions. Furthermore, the environmental conditions within the containment must not imperil the effective operation of other safety-related systems after a design-basis depressurization accident.

The information generated in the AIPA study, in particular that reported in Reference 42, suggests that the containment design bases should be reevaluated for HTGRs because the present bases fail to fully recognize the inherent safety features of the HTGR.

c. Evolution of Carbon Monoxide During an Unrestrained Core Heatup Accident and Its Effect on Containment Pressure and Temperature

The evolution of carbon monoxide from oxide or oxycarbide fuels by such reactions as $ThO_2 + 4C \longrightarrow ThC_2 + 2CO$ at elevated core temperature does not produce sufficient quantities of carbon monoxide to reach flammability conditions within the containment. The maximum amount of carbon monoxide that could be produced from the fuel provides only 50% of the quantity necessary to reach the lower flammability limit. Therefore, this reaction has a small impact on the containment design pressure and temperature.

2.4.2.7 Primary-System Integrity

The NRC submitted the following questions on primary-system integrity:

1. Provide thermal and mechanical design criteria for all essential components, structures, and systems of the primary coolant system for reactor operating conditions of normal, upset, emergency, and faulted. Discuss how these criteria accommodate considerations of secondary stresses, thermal shock, fatigue, and corrosion.
2. Describe how the primary system will meet General Design Criteria 14 and 15.
3. Describe how the design of the primary-system boundary accommodates hot streaks in both the upper and lower plenums. In considering hot streaks in the upper plenum due to flow reversal, assume among the cases studied that restart of forced convective cooling is not achieved until 2 hours after reactor scram. Consider both laminar and turbulent hot streaks in the lower plenum. Discuss the formation and decay of hot streaks. What data base supports this discussion?
4. How will past PCRV experience be used in the design of the asymmetric PCRV being considered for the 900-MWe steam-cycle HTGR. Is model testing anticipated and, if so, what scale is deemed adequate to confidently predict prototype performance?
5. What features of the conceptual designs for PCRV penetrations and closures will protect against sudden and rapid depressurization of the PCRV? Will these features be capable of inspection and testing during reactor operation?
6. What are the bases for the levels of acoustic excitation in the primary system?

a. Thermal and Mechanical Design Criteria

As indicated in Section 5 of the Fulton plant PSAR (Ref. 43), the primary coolant system is contained entirely within the PCRV (including valves, piping, penetrations, liners, and thermal barrier). Details of the design criteria and methods of analysis used in the design are covered in Section 5.4 of the PSAR. Criteria for the design of the steam generators and circulators are found in Section 5.5; criteria pertinent to the design of the core auxiliary cooling loop are contained in Section 6.3.

The reactor-coolant pressure boundary (RCPB) is designed to accommodate system pressures and temperatures for all modes of plant operation. Specific thermal and mechanical design criteria for all components of the RCPB are given in Section 5.2 of the PSAR. The RCPB is designed to accommodate the static and dynamic loads imposed by the temperature and pressure transients, listed as normal, upset, emergency, and faulty in Table 5.2.1-1 of the PSAR.

The design of components (e.g., vessels, piping, valves, and pumps) is governed by the requirements given in Regulatory Guide 1.48.

The above criteria, including those based on the ASME Boiler and Pressure Vessel Code, accommodate considerations of secondary stresses and fatigue. Corrosion effects, if significant, are analyzed separately to ensure that the material remaining after such corrosion is sufficient to meet the allowable loads. Corrosion in the inert helium environment is not expected to be significant, except in two potential areas: metal carburization in the top head and oxidation in the lower (graphite) core-support blocks because of impurities in the helium. These areas are under continued evaluation by the General Atomic staff, with program results and recommendations expected within 1 to 2 years.

b. Compliance with General Design Criteria 14 and 15

Compliance with General Design Criteria 14 and 15 is discussed in Sections 3.1.10 and 3.1.11 of the Fulton generating station PSAR (Ref. 43). The acceptability of this response was confirmed by the NRC in its review of the application (see Section 3.1 of Reference 12). Further discussions of these criteria can be found in Sections 3.1.2.5 and 3.1.2.6 of Reference 1.

c. Accommodation of Hot Streaks in the Upper and Lower Plenums

Hot streaks are defined as localized temperatures above the average temperature. Such streaks are accommodated in the primary-coolant-system design by the promotion of better gas mixing, while still attempting to minimize unrecoverable pressure losses. In both the upper and lower plenums this is achieved by

1. Delivery and reception of the gases in a symmetrical manner from and at the six steam-generator loops. This is difficult because of the clustered arrangement of the three core auxiliary heat exchangers in the upper plenum.
2. The use of many protrusions (flow barriers) in each plenum; e.g., control-rod guide tubes in the upper plenum and core-support posts in the lower plenum.
3. Forced 90-degree turns required of the gases as they enter (or leave) the core from (or to) the plenum.

In the event of a loss of forced cooling (LOFC), some of the hotter fuel regions can experience a flow reversal (upflow) because of natural convection effects. The hot plumes from these reversed-flow regions may impinge on the coverplates of the top head thermal barrier, causing local hot spots. For the Fort St. Vrain plant, an LOFC transient was assumed to last for 2 hours, after which primary-coolant flow was restored. Mixing effectively reduces the plume temperatures before the plumes impinge on the coverplates and additional cooling is provided by radiation from the coverplates to colder structures. Therefore, the coverplates are not expected to exceed the allowable 1,500°F during the 2-hour LOFC condition. A similar analysis for the 3,360-MWt HTGR has not been carried out.

Column hot streaks originate within a refueling region and appear at the outlet of a fuel column. They are created by the power variation from column to column within a refueling region. The column peaking factors are influenced by items such as control-rod position, fuel/poison loading, fuel-element location, and age (or burnup) of the fuel. Hot-streak decay (attenuation) in the lower plenum occurs as a result of the mixing chamber of the core-support block, region-to-region temperature control (operator control of flow control valve orifice position), passage of the gas through the maze of core-support posts, mixing with cold core-bypass flows, and other less significant effects.

Temperatures at various critical points in the lower portions of the HTGR as a function of time are calculated by codes such as COLUMN. These codes solve mass/energy balance equations using input from core-depletion codes. Experimental data are used to complement the analytical methods. Where experimental data are lacking (e.g., mixing in the plenum itself), a high degree of conservatism is employed. Experimental data from core-support-block (CSB) tests (Ref. 45) have provided influence-coefficient data on column-to-column mixing within the core-support block. Very recently, General Atomic completed influence-coefficient tests with an optimized CSB design based on the latest mixing pressure-difference data. These newly developed influence coefficients will be used in the hot-streak codes. The results of these tests will be published in a formal report in approximately 6 months.

The complex flow conditions in the lower plenum are very difficult to model, necessitating conservative assumptions for mixing. Laminar-flow hot streaks are currently modeled by RECA (Ref. 46), and preparations are under way for flow-distribution tests to define mixing behavior in the turbulent-flow regime. A 1/20th-scale test loop to be used with water and injected dyes or gas bubbles is approximately 25% constructed. Installation and shakedown of the loop are expected to be finished by January 1979, with testing to be completed by mid-April 1979.

d. Use of Past PCRV Experience in Designing the Asymmetric PCRV

Past PCRV design experience has been accumulated from engineering experience with gas-cooled nuclear reactors and supported by extensive research and development programs at General Atomic and abroad. Development programs such as studies on PCRV concrete properties, large-tendon qualification tests, prestressing steel relaxation tests, and wire-winding-machine tests are generic in nature and should be equally applicable to the design of the asymmetric PCRV. The analytical and model techniques previously developed and employed in connection with the multicavity PCRV design have been duly verified and documented in Reference 47. The validity of the finite-element method used primarily in the PCRV design is independent of geometry and boundary conditions of the structure. It is recognized that the design of an asymmetric PCRV will require a more extensive analytical effort. Preliminary assessment of the asymmetric PCRV for the 900-MWe steam-cycle HTGR is based on elastic two-dimensional planar-section analyses with three-dimensional structural effects estimated from the results of previous analyses of the multicavity PCRV. The preliminary satisfactory evaluations of the asymmetric PCRV layout remain to be confirmed by more exacting three-dimensional finite-element analyses. It is anticipated that the asymmetric PCRV design will be further confirmed by a scale-model test to be conducted by the Oak Ridge National Laboratory. The model scale, consistent with the recommendation in ASME Code Section III, Division 2, should allow sufficient instrumentation and realistic modeling of significant features of a multicavity PCRV. A scale between 1/14 and 1/10 is considered adequate for pressure testing the asymmetric PCRV model.

e. Protection Against Sudden and Rapid PCRV Depressurization

Most PCRV penetrations and closures are designed, fabricated, and examined to the same rules as LWR vessels (ASME Code Section III, Division 1, Subsections NB-2000 through NB-5000). In-service inspection of these penetrations and closures will be the same as for LWR vessels, in that every weld region whose failure could lead to rapid vessel depressurization is subject to volumetric examination on a periodic basis. Thus, like LWR vessels, these penetrations and closures are not postulated to fail. A further discussion of this subject is contained in Reference 48.

Two types of closure do not fall in the above category. The first is a closure constructed of prestressed concrete, such as those used for large heat-exchanger cavities. These concrete closures are designed, constructed, and examined according to the rules of ASME Code Section III, Division 2, and inspected to the rules of Section XI, Division 2. They are in a state of net compression, maintained by the prestressing forces. Thus, even though a crack is nonmechanistically postulated in these closures, the prestressing force will keep the crack closed and prevent rapid depressurization of the vessel. Where the prestressing force is maintained by metallic elements, multiple, independent members with considerable redundancy are used. Thus, gross failure of such concrete closures is not considered credible just as gross failure of the PCRV is not considered credible. This precludes rapid depressurization through such concrete closures.

The second type of closure that is not in accordance with ASME Code Section III, Division 1, Class 1, is a steel closure whose temperature exceeds that allowed by the Code, as may occur at a steam-pipe penetration. Such penetrations are designed to meet the rules of high-temperature Code cases, such as Code Case 1592. In addition, to protect against rapid depressurization of the vessel resulting from a postulated gross failure of this type of closure, flow restrictors are provided. Flow-restriction devices include items normally available to limit free-flow area and items specially provided to limit free-flow area or limit movement of the failed closure. Such flow restrictors are subject to in-service examination in accordance with the rules of ASME Code Section XI, Division 2. Flow restrictors are also provided to limit the free-flow area from a penetration in the postulated event of complete rupture of a large pipe that is attached to the vessel and contains primary coolant.

These design features of penetrations and closures with their respective inspection programs protect against sudden and rapid depressurization of the PCRV.

f. Bases for Establishing Levels of Acoustic Excitation in the Primary System

The Acoustic and Vibration Plant Specification (Ref. 49) provides a detailed list of design sound-pressure levels in nine frequency bands throughout the primary-coolant circuit. (Although written for a previous six-loop design, Reference 51 is applicable to the current four-loop design.) The specification also lists the maximum permissible strengths of four classes of acoustic sources: (a) main circulators, (b) core regions, (c) steam generators, and (d) all other sources. Each component must be designed to withstand the specified acoustic pressures while not radiating more than the maximum permissible sound level.

An acoustic-propagation analysis has been performed to ensure that the maximum acoustic source strengths of the specification are consistent with the design sound pressures. Most of the analysis is defended in Reference 50. The computer code VIBRAPHONE is used for low-frequency circuit analysis (Ref. 51). A scale-model acoustic systems test, scheduled for 1980 and 1981, will verify the sound-propagation analysis and provide a limited amount of structural vibration data.

The turbomachinery is expected to make most of the noise; its acoustic source strength receives the most attention. Many measurements of single-stage axial fan noise appear in the literature; the most relevant is Reference 52. Full-scale noise measurements are planned. Furthermore, sound radiation from other components is measured as part of the various design verification and support programs. All measurements are used in the specification (Ref. 51) with the intent that the specified acoustic source strengths are in fact greater than any actual sources that might exist in the reactor.

2.4.2.8 Emergency Core-Cooling Provisions

The NRC submitted the following questions on emergency core-cooling provisions:

1. State the performance criteria for the emergency core-cooling system (ECCS).
2. Discuss the role of containment backpressure and loop-isolation valves in relationship to expected ECCS performance. What are the design provisions that assure these features will perform in accordance with criteria established for engineered safety features? Identify development programs supporting the design provisions of these features.

a. Performance Criteria for the Emergency Core-Cooling System

Performance criteria for the core auxiliary cooling system (CACS) (which provides the LWR functions of emergency core cooling and residual heat removal) are fully discussed in Section 6.3.1 of the PSAR for the Fulton nuclear power plant.

b. Role of Containment Backpressure and Loop-Isolation Valves

The HTGR CACS is designed to operate in two modes of cooling: pressurized and depressurized. Depressurization of the primary system results from a gross failure of a structural member in a major penetration closure of the PCRV. The event with the maximum rate of depressurization is referred to as the design-basis depressurization accident (DBDA). Since the HTGR is cooled by circulating a gaseous coolant through the core, the performance of the CACS is dependent on the gas density inside the PCRV. Therefore, during a DBDA, CACS performance is dependent on the design minimum equilibrium pressure between the PCRV and containment. The conservatively calculated containment backpressure is always greater than the minimum required backpressure. Additional information related to backpressure requirements for adequate CACS performance may be found in Section 6.3.3.2.2 of the Fulton plant PSAR and also in Reference 53.

Adequate operation of the CACS during pressurized or depressurized cooling is dependent on isolation of the main-loop cooling system. Loop-isolation valves are Safety Class 2, Seismic Category I, and act automatically to isolate the main loops and prevent core-bypass flow during CACS operation. Nevertheless, the CACS is designed to provide adequate cooling for all credible events assuming a failure of one main loop-isolation valve or one CACS loop.

The CACS has several design requirements to ensure that the system will meet the appropriate criteria established by the NRC for engineered safety features. In particular, it is designed to meet the single-failure criterion and is a Safety Class 1/2 and Seismic Category I system. The CACS is designed to operate adequately in any containment environment resulting from any credible event. It is capable of operating from either onsite or offsite power sources and is capable of resuming proper operation and supplying adequate cooling after a loss-of-offsite-power event at any time during any credible accident sequence. The auxiliary circulator is capable of operating without flow instability or surge throughout the operation range. Also, a depressurization event through a CACS penetration must not prevent the loop from performing its safety function. In addition, the CACS design considers uncertainties in all relevant parameters in order to clearly demonstrate the ability of the system to provide adequate cooling in all plant conditions. Two development programs are planned that support the design provisions of the CACS: (a) the CACS testing criteria program whereby plans for preliminary CACS testing would be developed, and (b) development of a computer program for assessing the stability margin for the core auxiliary heat exchangers. The CACS testing criteria program is part of a long-term effort to perform preoperational design verification testing and online testing during plant startup of the CACS.

Table 2-16. HTGR design parameters

Parameters	Peach Bottom	Fort St. Vrain	GASSAR-6	Lead plant U-235/Th once-through	Lead plant U-233/Th recyle
Net electrical output, MW	40	330	1,159	1,332	
Overall station net efficiency, %	34.6	39.2	38.6	39.64	
Containment type	Steel	Atmospheric confinement	Reinforced concrete/steel	Reinforced concrete/steel	
Number of main/emergency cooling loops	2/2	2/2	6/3	6/3	
Reactor core output, MWt	115	842	3,000	3,360	
Core diameter/height, ft	9.16/7.5	19.6/15.6	27.7/20.8	36.6/28.6	
Helium coolant inlet pressure, psig	305	688	725	780	
Average coolant temperature, reactor inlet, ^{o}F	650	762	606	620	
Average coolant temperature, reactor outlet, ^{o}F	1,380	1,445	1,392	1,328	
Average power density, kWt/liter	8.3	6.3	8.4	7.1	
Average conversion ratio	0.44	0.60	0.65	0.56	0.77
Fuel material	Th/U-235, 95% enriched/U-233, recycle			MEU-233/Th	MEU-233/Th
Element length/minimum width, in.	144/3.5	31.22/14.7	31.22/14.7	31.22/14.17	
Total quantity of U-235/Th (initial), kg	220/1450	882/19,458	1,747/37,487	1,784/31,800	1,797 (U-233)/41,316
Average fuel burnup, MWd/MT	60,000	100,000	98,000	130,000	48,000
Reactor vessel type	Steel pressure vessel	Prestressed-concrete reactor vessel			
Maximum external dimensions, diameter/height, ft	14.5/35.5	49/106	100.5/91.2	111.5/89.0	
Helium-circulator type	Centrifugal, electric drive	Single-stage axial flow, steam-turbine drive			
Steam-generator type	Forced recirculation	Once-through, helical coil with integral reheat			
Reactivity control	Control rods and emergency shutdown canisters				
Scram method rods	Hydraulic/electric	Gravity	Gravity	Gravity	
Emergency core-cooling system	Pony motors, natural convection	Uses existing main circulators, water turbine	Three independent cooling loops, electric motor	Three independent cooling loops, electric motor	

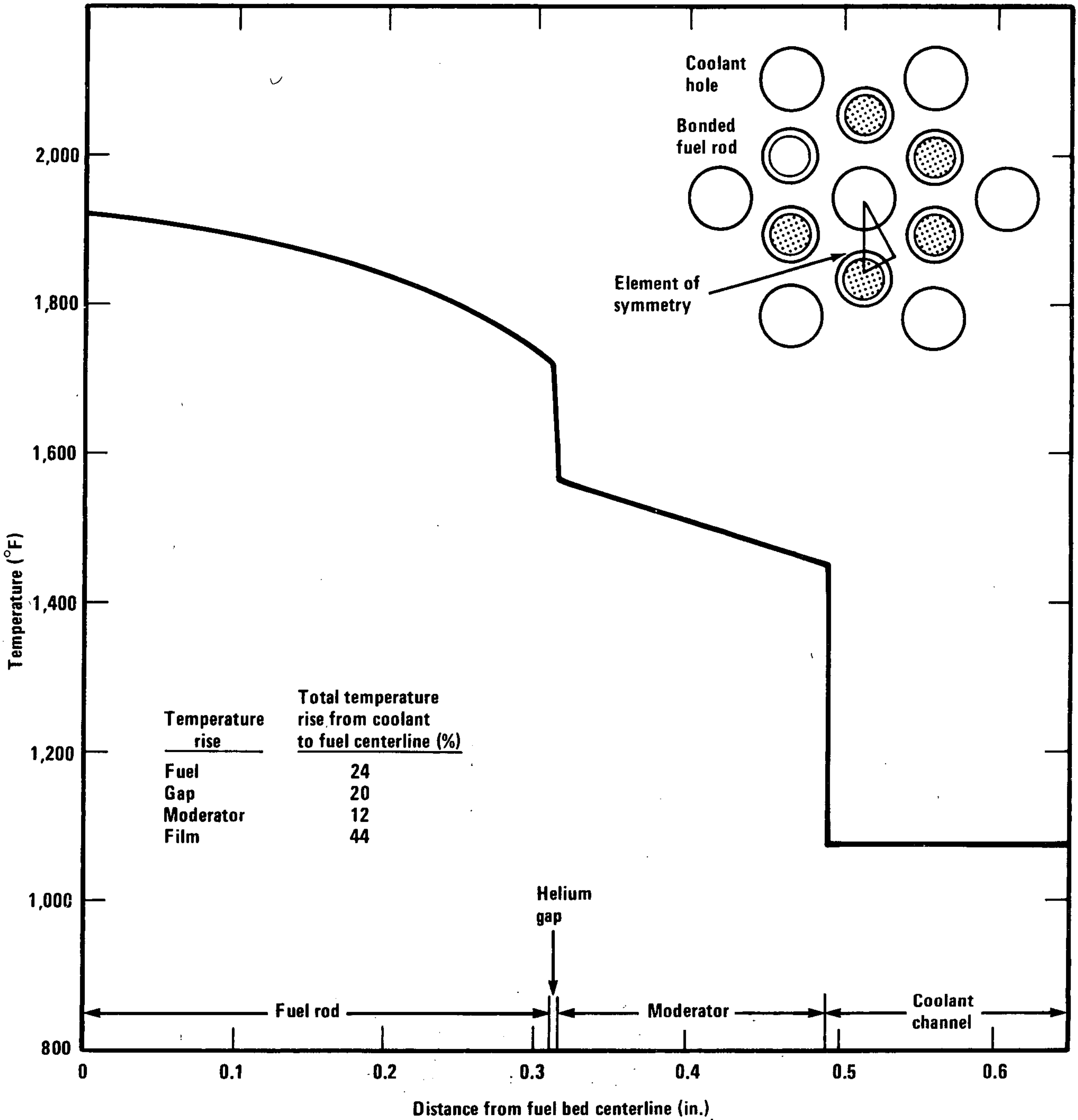


Figure 2-5. Radial temperature profile in an average power channel at a fractional core length of 0.5.

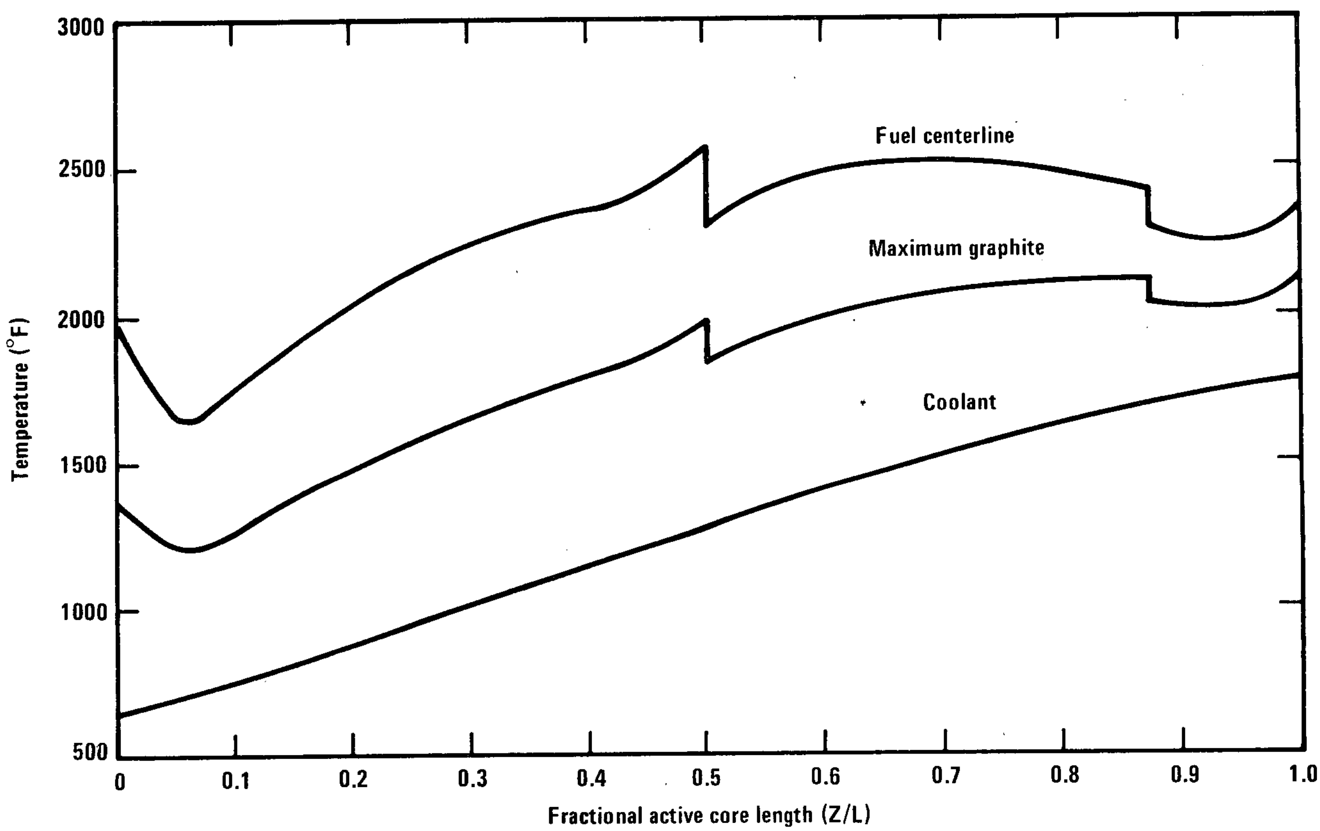


Figure 2-6. Axial temperature distribution in high-power channel.

1072 iv

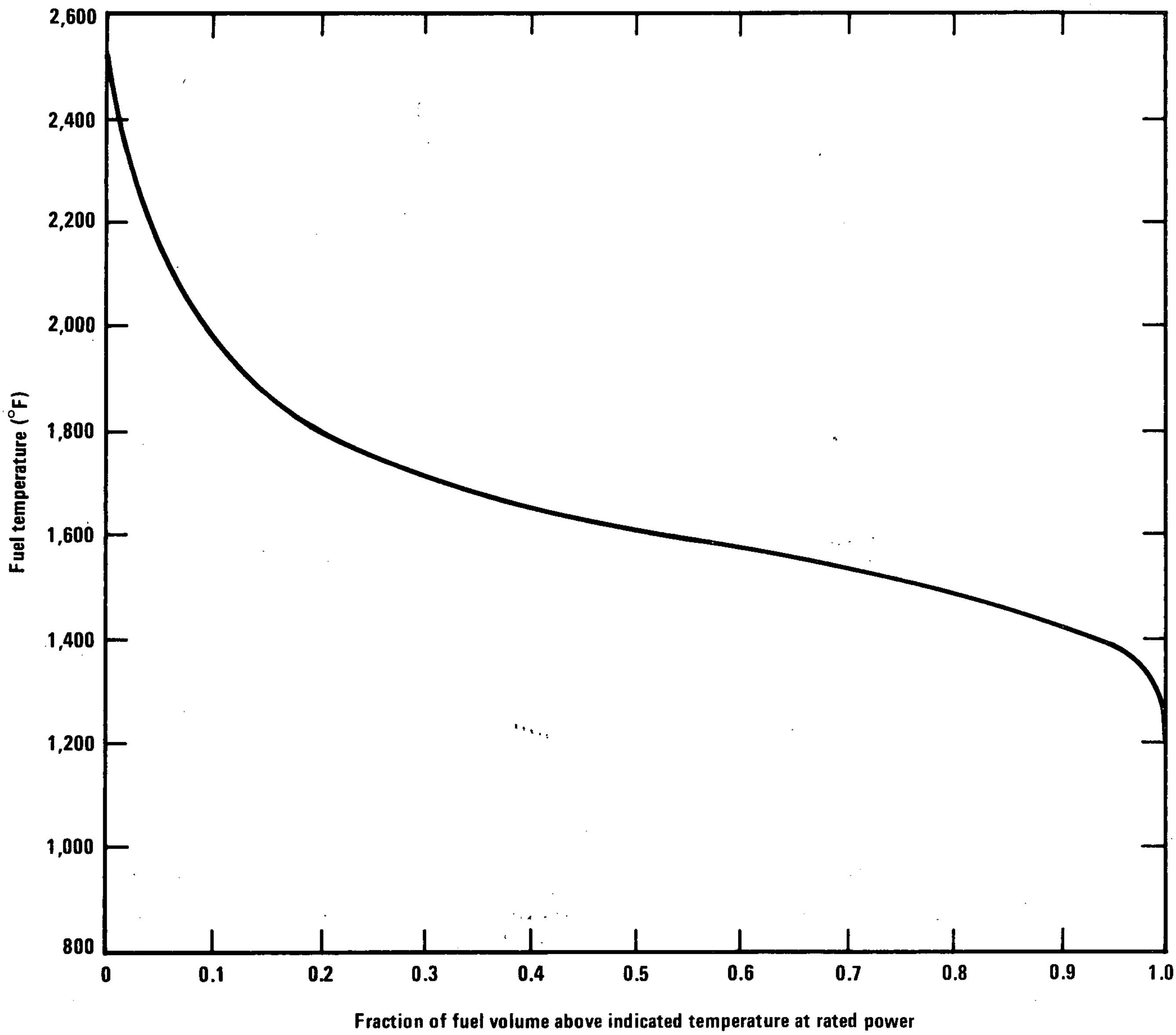


Figure 2-7. Fraction of fuel volume above indicated temperature at rated power.

1072 iv

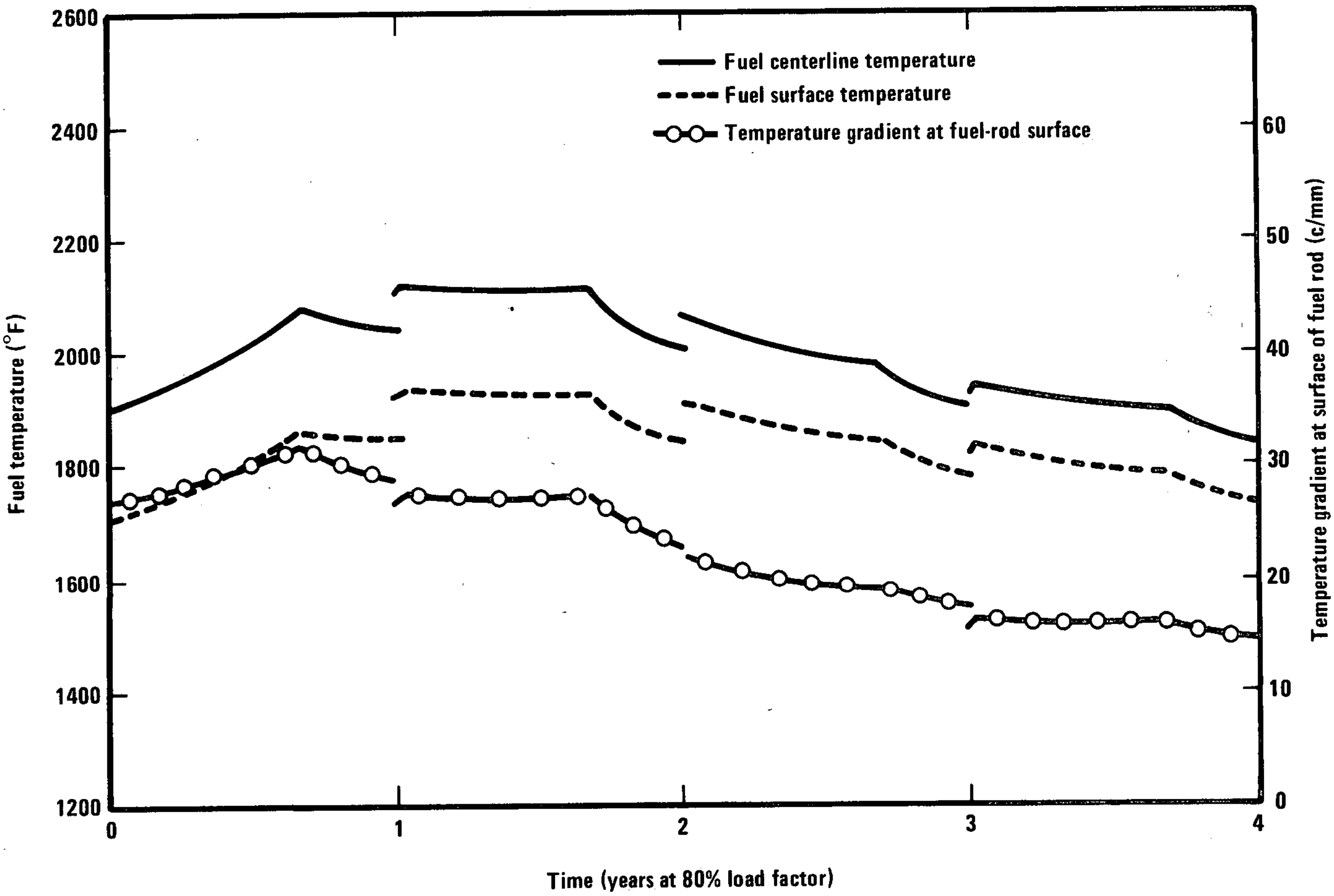


Figure 2-8. Temperature histories for an unrodded region.

1072 iv

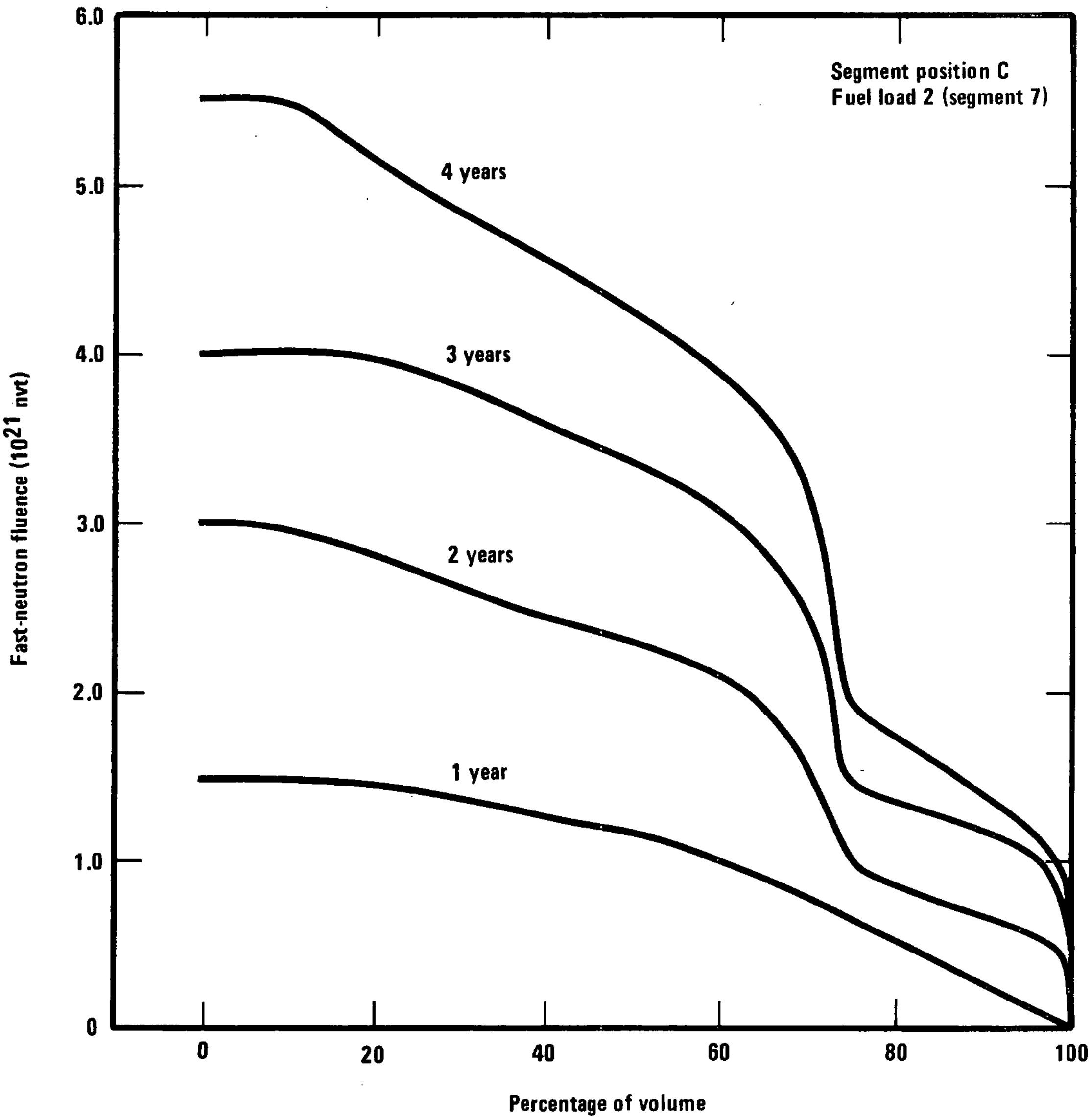


Figure 2-9. Core volume distribution (segment average) of fast-neutron fluence as a function of fuel age.

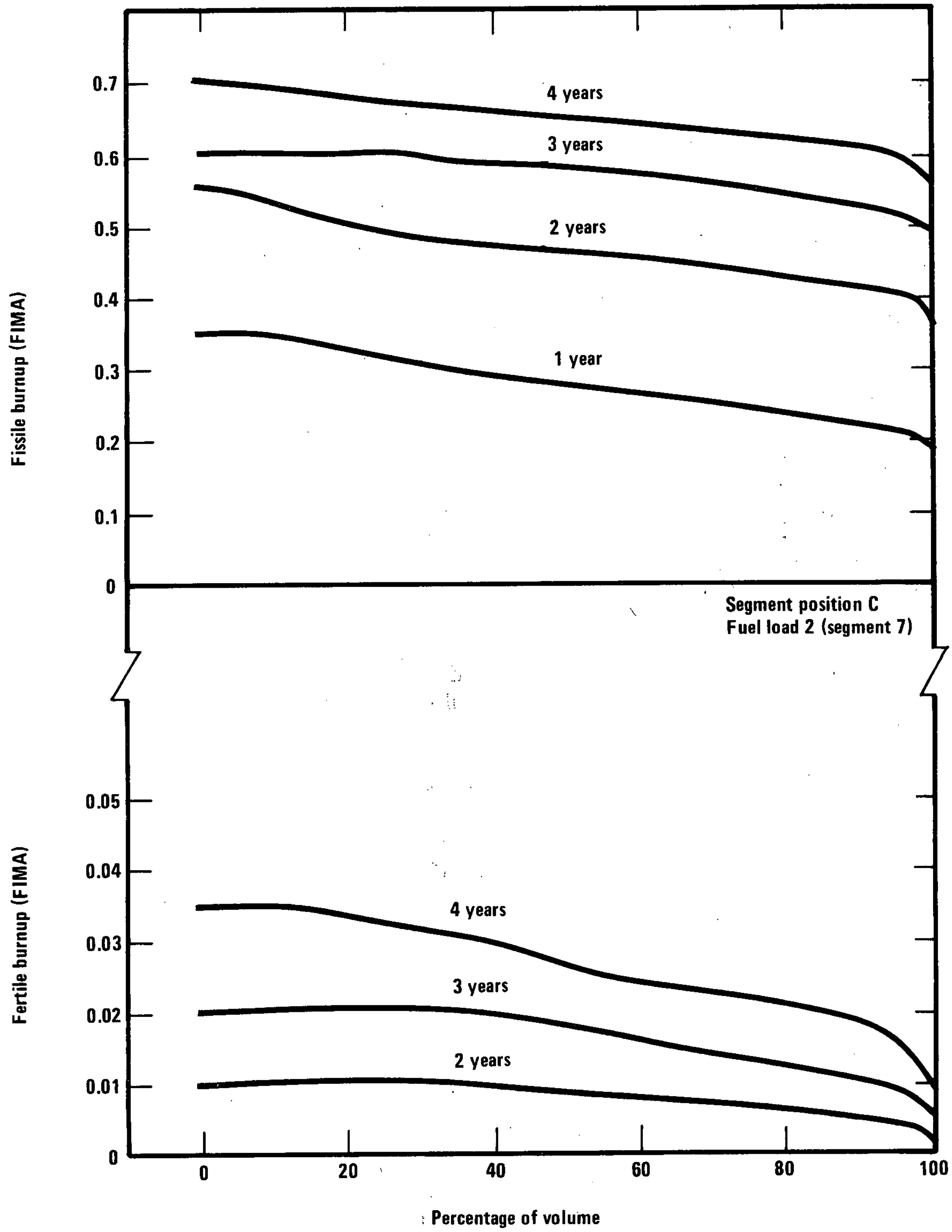


Figure 2-10. Core volume distribution (segment average) of fertile and fissile burnup as a function of fuel age.

2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

2.5.1 RESEARCH AND DEVELOPMENT

The HTGR concept has been under development for almost 20 years, and its feasibility has been established by the Peach Bottom Unit 1 prototype plant and the Fort St. Vrain demonstration plant. The basic mechanical, thermal-hydraulic, and materials requirements of operating a high-temperature system with a graphite moderator and a helium coolant have for the most part been addressed and solved.

A significant research and development program (including in certain cases full-scale prototype tests) formed the basis for the design and key elements in the foregoing projects and is continuing to provide a basis for generic design development. The HTGR research and development programs are sponsored both by private industry and the government. The major U.S. participants are the General Atomic Company and the Oak Ridge National Laboratory, supported by other organizations and test facilities. A major element in this ongoing research and development program has been the highly successful cooperative program, initiated in 1972, between General Atomic and the French Commissariat ã l'Energie Atomique. Another, more recent, development is the four-party (United States, Federal Republic of Germany, France, and Switzerland) government-level Umbrella Agreement to participate jointly in cooperative gas-cooled-reactor development programs. Initiated in 1977, this cooperation is coming increasingly into effect.

A major part of the research and development work being performed and planned is related to mechanical, thermal-hydraulic, and materials factors. The programs cover the following major areas:

1. Development of testing and analytical computer methods in structural mechanics and thermal and fluid mechanics. In the area of analytical methods, the work includes computer-program improvement and verification, making use of Fort St. Vrain experience and test programs at various research laboratories.
2. Acquisition of basic materials data.
3. Development of fuel and core materials. Irradiation and out-of-pile testing on fuel and graphite materials continues, along with the gathering of data from the Fort St. Vrain core. Fission-product and coolant-chemistry studies are also being pursued at the General Atomic Company and the Oak Ridge National Laboratory.
4. Surveillance testing at Fort St. Vrain and Peach Bottom. Surveillance tests continue as operation progresses at Fort St. Vrain. The Peach Bottom end-of-life program, aimed primarily at the verification of materials and fission-product methods, has recently been completed.
5. Research and development programs to verify the design of major plant components, such as the PCRV, steam generators and core auxiliary heat exchangers, main and auxiliary circulators, refueling equipment, reactor internal structures, and thermal barrier and control-rod assemblies.

2.5.1.1 Thermal-Hydraulic Program

Specific research and development activities related to thermal-hydraulic factors include the following:

1. Lower-plenum gas-mixing studies
2. Lower support-block gas mixing
3. Development of methods for analyzing plenum pressure drop and flow distribution
4. Development of methods for thermal and fluid-mechanics analysis

2.5.1.2 Materials Program

In the materials area, the research and development program includes investigations of many types, including the following:

1. Wear technology
2. Aging and cold-work effects
3. Helium effects on design properties
4. Fatigue properties
5. Structural ceramics
6. Improved thermal-barrier materials
7. Design properties of welds
8. Design data accumulation and analysis
9. Crack propagation and toughness

2.5.1.3 Structural Engineering Program

In the structural engineering area, the primary research and development objectives are to develop the technological data base and confirm the recently optimized designs for the PCRV, liner and penetrations, thermal barrier, and reactor internals.

2.5.1.4 Heat-Exchanger Equipment Program

The objectives of the research and development program related to heat-exchanger equipment are to develop and improve generic analytical and design methods as the basis for heat-exchanger designs for the steam-cycle, direct-cycle, and process-heat HTGR applications.

2.5.1.5 Refueling Equipment Program

The objectives of the research and development program for refueling equipment are to provide component development and operation tests to verify the adequacy of evolutionary changes in the design of the refueling system.

2.5.1.6 Rotating Machinery Program

Prototype tests are planned for the development of the electrically driven circulators to verify their design performance and to establish the reliability of the circulators, their drivers, and the essential service and control systems.

2.5.1.7 Reactor Core Program

The research and development program covering the reactor core will include the design, analysis, and testing of the reactor core and its components (fuel elements, hexagonal reflector elements, plenum elements, neutron sources, control rods, and reserve shutdown material). The work in this area is aimed at providing the development and verification needed for the core components, material properties, and design methods.

2.5.2 DEVELOPMENT AND DEMONSTRATION

Commercial HTGR plants, according to the commercialization plan adopted by Gas Cooled Reactor Associates (GCRA) would come on line in the 1990s at a rate providing about 20 GWe of capacity through the year 2000. The commercialization program has been developed by GCRA through the HTGR Commercialization Program Plan covering a 12-year commercialization period.

The scheme detailed in the program plan was structured for the first plant as a three-phase program, with discrete milestones and decision points at the completion of each phase. The three phases are

Phase I: Program Definition
Phase II: Plant Design and Licensing
Phase III: Plant Construction

In addition, a fourth "commercialization" phase was also included for the design and construction of follow-on plants. The program definition phase, now in progress, is directed to the definition of the technical, project, and business elements of the program plan.

The GCRA program plan includes both "generic" and "specific" technology-development and design-verification activities applicable to the steam-cycle, direct-cycle, and process-heat HTGRs. These activities include development and performance verification of materials, components, and systems; the performance of safety, reliability, and availability analyses; and the development and verification of analytical methods.

Technology transfer is also part of the HTGR development program. This is from operating experience with the Peach Bottom and Fort St. Vrain reactors and includes results from end-of-life examination of Peach Bottom and experience from LWRs. Technology developed in the French and German HTGR programs and those in other countries, such as Japan, will be evaluated, as will the gas-cooled reactors in Britain.

The existing requirements for each plant component for demonstration are discussed below.

2.5.2.1 Nuclear Fuel

For all of the HTGR fuel cycles based on uranium and thorium fuel materials, the technology is developed and demonstrated. Contemporary technology with a modest modification for application would be acceptable for all fuel cycles except the plutonium/thorium cycle. The large fuel-development programs carried out in the United States and Europe over the past 20 years on high-, low-, and medium-enrichment fuels is directly applicable. In the case of the plutonium/thorium cycle there would be a requirement for process development and characterization testing because plutonium fuels have not been fabricated or tested in commercial-scale equipment for HTGR applications. The plutonium/thorium cycle is judged to be in the category requiring "modest improvement in performance and modified configuration/application."

2.5.2.2 Reactivity-Control Systems

These are essentially the same as those currently in operation in the Fort St. Vrain reactor. No new configurations or significant size changes are involved.

2.5.2.3 Reactor Vessel

The layout and size of the PCRV are different from those of any other vessel, but no new technology or fabrication requirements are involved. The major constituent parts of the vessel--concrete, 3/4-inch steel plate, and insulation panels--are such that change in layout and size is not a significant technical factor. The design for the Delmarva Power & Light Company (DPL) project had reached the point where material procurement had been initiated.

2.5.2.4 Core-Support Structure

This system is very similar to that employed in the Fort St. Vrain reactor, with an increase in the number of components with no significant change in their size and with no new materials.

2.5.2.5 Reactor-Vessel Internals

The permanent reflector is not significantly different from the Fort St. Vrain design. However, the lateral restraint structure and the peripheral seal are configurations that have been developed, analyzed, and tested for the large HTGR plant. No new technology is involved. The design is essentially the same as that used in the DPL project.

2.5.2.6 Primary-Coolant Pumps

A primary-coolant-circulator prototype for the DPL and the lead plant has been on test for some time, and no further development is required.

2.5.2.7 Primary-Coolant Chemistry and Radiochemistry Control

For the 1,330-MWe high-enrichment fuel cycle, the chemistry is well understood and Fort St. Vrain is providing additional data. For the medium-enriched uranium/thorium cycle, the radiochemistry will have to be determined by analysis and testing of the new fuel. Current predictions are that no major problems are involved in using medium-enriched uranium fuel in the existing plant design.

2.5.2.8 Primary-System Heat Exchangers

The heat exchangers represent a scaling up from the Fort St. Vrain design but not from the DPL design, for which material procurement had been initiated. Helium-side conditions are less severe in the lead plant than in the DPL and Fort St. Vrain designs.

2.5.2.9 Reactor Instrumentation

No new instrumentation is required.

2.5.2.10 Emergency Core-Cooling/Safe-Shutdown System

The core auxiliary heat exchangers are of a configuration that has not yet been constructed, but the materials and heat-exchanger technologies are essentially the same as those for the main heat exchanger, which has been proved, and the design is similar to that of the DPL project.

2.5.2.11 Containment, Containment-Cleanup, and Effluent-Control Systems

The containment itself requires standard technology common to all reactor types, and the ventilation system has less severe requirements than do comparable systems for LWRs. Furthermore, all the hardware in the HTGR containment is standard equipment of proved design.

2.5.2.12 Plant-Protection System

The plant protection system has been developed from the Fort St. Vrain system, from which much operating experience is available. No significant development is involved.

2.5.2.13 Onsite Fuel-Handling, Storage, and Shipping Equipment

The fuel-handling machine is based on the design currently in use at Fort St. Vrain. Fuel storage and shipping equipment has been designed for the HTGR. Differences from the Fort St. Vrain equipment in the storage area are due to cost-optimization studies, and the Fort St. Vrain systems could, if required, be used instead of the revised design; however, no technology advance is involved.

2.5.2.14 Main Turbine

The amount of development required for the main turbine depends on the size of the plant and the number of turbines per plant. For instance, a single turbine with a 1,330-MWe gas-reheat plant will require some development, which has been evaluated with suppliers. Smaller (twin) machines could be used that do not require this development but with some penalty in generating cost.

2.5.2.15 Balance of Plant

No major technological advance is required in the balance of plant for the HTGR.

2.5.3 SUMMARY--STATUS OF RESEARCH AND DEVELOPMENT REQUIREMENTS

The research and development requirements for each of the above plant components are summarized in Table 2-17.

Table 2-17. Status of research and development requirements for 1,330-MWe lead-plant-HTGR design

Plant component	No new knowledge required	Contemporary technology with modified configuration/ application	Modest improvement in performance or size from present knowledge
Nuclear fuel		X	(a)
Reactivity-control systems	X		
Reactor vessel		X	
Core-support structure		X	
Reactor-vessel internals, including shielding, ducting, control-rod guides, baffles, etc.			X
Primary-coolant pumps and auxiliary systems	X		
Primary-coolant chemistry and radiochemistry control		X	(a)
Primary system heat exchangers			X
Reactor instrumentation	X		
Emergency core-cooling/ safe-shutdown system			X
Containment, containment-cleanup systems, and effluent-control systems		X	
Other accident-mitigating systems (i.e., plant protection systems)		X	
On-site fuel-handling, storage, and shipping equipment		X	
Main turbine		X	
Other critical components, if any			
Balance-of-plant components	X		

[a]For medium-enrichment fuel cycle.

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Chapter 3

MEDIUM-ENRICHMENT URANIUM-233/THORIUM, RECYCLE FUEL

3.1 DESCRIPTION

This reactor/fuel-cycle combination is a high-temperature gas-cooled reactor (HTGR) using recycled, denatured uranium-233 as 12% fissile spherical particles and thorium oxide fertile particles. The spent fuel is reprocessed to separate the fissile and fertile particles which are reprocessed separately. The uranium and plutonium are recovered, separated, and sent to secure storage. The fertile particles are reprocessed to recover the bred uranium-233 which is diluted with depleted uranium to 12% fissile and recycled to fabrication. The thorium recovered during reprocessing is placed in 10-year storage. Wastes from fuel fabrication and from reprocessing are sent to a geologic waste repository.

The generalized reactor performance and design data specifications are summarized in Chapter 1. Data on fuel management are given in Section 3.1.4.

3.1.1 FUEL MECHANICAL DESIGN

For a complete description of the bases of the fuel mechanical design, see Section 4.2.1 of Reference 1.

3.1.2 FUEL NUCLEAR DESIGN

For a complete description of the bases of the fuel nuclear design, see Section 4.3 of Reference 1.

3.1.3 FUEL THERMAL-HYDRAULIC DESIGN

For a complete description of the bases for the fuel thermal-hydraulic design, see Section 4.4 of Reference 1.

3.1.4 FUEL MANAGEMENT

Fuel-management information is given in Table 3-1. The fresh fuel and spent fuel are characterized in Table 3-2, which includes data on the heavy-element isotopic content for initial and equilibrium loadings and discharges. Fuel mass-flow data (charge and discharge) are given in Table 3-3. Fuel isotopic data (charge and discharge) are given in Table 3-4.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram (Figure 3-1) and are discussed in the following sections of Volume VII:

Fuel fabrication 7	Chapter 4
Reprocessing (Purex 1)	Section 5.1
Reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Depleted uranium storage	Section 6.4

Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

3.2 SAFETY CONSIDERATIONS

Safety considerations for this fuel cycle are identical with those described in Section 2.2.

3.3 ENVIRONMENTAL CONSIDERATIONS

The primary fissile isotope in the fuel cycle is uranium-233, with a small amount of uranium-235 present. Thorium is the predominant fertile isotope, yielding additional uranium-233 during operation. A small amount of plutonium is also produced from the uranium-238 used to denature the uranium-233. The core is similar to that used in the medium-enrichment uranium/thorium once-through cycle, the major difference being the use of 12% enriched uranium-233, rather than 20% enriched uranium-235, as the fissile material. The core dimensions, total fissile and fertile loadings, power density, and other parameters are likewise similar for the two cores. Plant parameters other than those for the core are identical.

Fission-product yields for this fuel are similar to yields for the once-through fuel, particularly for isotopes that are of concern in the environmental evaluation. Thus, the inventories of key isotopes in the core and potentially available for release to the coolant are similar, but not identical. Since the power densities, core temperatures, fuel-particle coatings, and fuel-element failure rates are similar for the two cores, the fractional release of the fission products to the coolant is expected to be approximately the same for both cores, and the concentrations of important fission products in the coolant would be similar. The remaining plant features that affect the transport of fission products to the environment are the same for both cycles. In addition, maintenance and refueling schedules would not differ significantly. Thus, the environmental effects of this cycle would be similar to those of the medium-enrichment uranium/thorium once-through cycle, which are described in Section 2.1.2.

3.4 LICENSING STATUS AND CONSIDERATIONS

The status and considerations for this fuel cycle are identical with those discussed in Section 2.4.

3.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The research and development programs required for the design, construction, and licensing of a commercial plant for this fuel cycle are identical with those outlined in Section 2.5.

Table 3-1. Parameters for the medium-enrichment uranium-233/thorium recycle fuel cycle[a]

Average capacity factor, %	75
Fuel form	Coated oxide or carbide particles
Fraction of core replaced annually	0.33
Enrichment-plant tails assay, %	0.2
Core power density, W/cm^3	7.1
Carbon-to-thorium ratio	
Initial core	275
Equilibrium reload	300
Fuel-rod diameter, cm	1.59
Average fuel temperature, °C	880
Maximum fuel temperature, °C	1,350
Core fuel loading, kg/GWe (initial core/equilibrium reload)	
Total heavy metal	42,300/13,762
Fissile	1,350/585 (242 net)[b]
Burnup, MWd/MT	
Average	48,000
Peak	60,500
Conversion ratio	
Beginning of life (initial core)	0.76
After equilibrium fuel loading	0.73
Average during equilibrium	0.77
Yellowcake requirements	None
Separative-work requirements	None
Annual discharge, kg/GWe	
Fissile plutonium	50
Total plutonium	75
Uranium-235	17
Bred uranium-233	327
Total uranium	4,519
Total thorium	8,450
30-year cumulative discharge,[c] kg/GWe	
Fissile plutonium	1,560
Total plutonium	2,310
Uranium-235	492
Bred uranium-233	10,690
Total uranium	146,000
Total thorium	297,000

[a]Fissile material is 12% enriched uranium-233; fertile material is thorium; annual refueling, 3-year cycle. An external source is required for uranium-233 makeup.

[b]Bred fissile material required annually from an external source.

[c]The 30-year cumulative discharge is the sum of 30 annual discharges plus the partially consumed heavy metal in the reactor at the end-of-plant life.

Table 3-2. Characterization of HTGR fresh and spent fuel for the medium-enrichment uranium-233/thorium recycle fuel cycle

Cycle, years	3
Refueling method	Batch
Refueling frequency	Annual
Fuel-assembly characteristics	
Type	Oxide and carbide
Weight, kg	119
Length, m	0.79
Core load mass, kg HM/GWe	42,300
Annual reload mass at 75% capacity factor, kg HM/GWe	13,762
Design burnup,[a] MWd/MT	48,000
Dose rate at 1 m in air after 90 days, rem/hr	~4,900

Heavy-metal element isotopic content[b]

Isotope	Fresh fuel element (kg) Initial	Fresh fuel element (kg) Equilibrium	Discharged fuel element (kg) Initial	Discharged fuel element (kg) Equilibrium
Thorium-232	7.81	7.14	7.46	6.83
Uranium-233	0.34	0.44	0.24	0.26
Uranium-234	--	0.018	0.039	0.058
Uranium-235	--	0.003	0.008	0.014
Uranium-236	--	2.0×10^{-4}	9.49×10^{-4}	0.002
Uranium-238	2.50	3.29	2.39	3.15
Neptunium-237	--	--	3.40×10^{-5}	4.60×10^{-5}
Plutonium-238	--	--	1.1×10^{-5}	1.25×10^{-5}
Plutonium-239	--	--	0.02	0.025
Plutonium-240	--	--	0.008	0.01
Plutonium-241	--	--	0.01	0.009
Plutonium-242	--	--	0.006	0.006

[a]Discharge batch average.

[b]Multiply by 1,332 (fuel elements per GWe) for the isotopic content in kilograms.

Table 3-3. Fuel mass flows[a] for the medium-enrichment uranium-233/thorium recycle fuel cycle

Segment	12	13	14	15	16	17	18	19	20	21
Region	C	A	B	C	A	B	C	A	B	C
Discharge time (yr)[b]	12.80	13.87	14.93	16.00	17.07	18.13	19.20	20.27	20.27	20.27
Thorium charged	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4
Uranium-235 loaded	4.7	4.6	4.6	4.6	4.6	4.6	4.6	4.6	4.6	4.6
Uranium-233 loaded	828.1	828.4	829.4	829.3	829.2	829.0	829.1	829.1	829.1	829.1
Total uranium loaded	7,023.0	7,025.8	7,033.9	7,033.1	7,032.1	7,030.8	7,031.2	7,031.4	7,031.6	7031.5
Total metal loaded	19,604.4	19,607.2	19,615.3	19,614.5	19,613.5	19,612.2	19,612.6	19,612.8	19,613.0	19,612.9
Thorium discharged	12,043.2	12,043.1	12,043.1	12,043.2	12,043.2	12,043.2	12,043.2	12,043.2	12,220.0	12,399.4
Bred uranium-233 discharged for recycle	286.4	286.4	286.4	286.4	286.4	286.4	286.4	286.4	236.0	148.5
Bred uranium-235 discharged for recycle	4.6	4.6	4.6	4.6	4.6	4.6	4.6	4.6	1.9	.4
Total bred uranium discharged for recycle	323.4	323.4	323.4	323.4	323.4	323.4	323.4	323.4	256.4	155.4
Uranium-233 discharged for credit	179.3	179.3	179.5	179.5	179.5	179.5	179.5	179.5	298.9	497.8
Uranium-235 discharged for credit	19.7	19.7	19.7	19.7	19.7	19.7	19.7	19.7	15.3	9.6
Total uranium discharged for credit	6,107.7	6,110.1	6,117.4	6,116.7	115.9	6,114.7	6,115.0	6,115.2	6,335.4	6,625.2
Total uranium discharged	6,431.1	6,433.5	6,440.8	6,440.1	6,439.3	6,438.1	6,438.4	6,438.6	6,591.8	6,780.6
Fissile plutonium retired	71.4	71.5	71.5	71.5	71.5	71.5	71.5	71.5	66.8	51.8
Total plutonium retired	106.9	106.9	107.0	107.0	107.0	106.9	107.0	107.0	93.9	66.6
Total metal discharged	18,581.1	18,583.5	18,590.9	18,590.3	18,589.4	18,588.2	18,588.5	18,588.8	18,905.7	19,246.6

[a]Mass flows are in kilograms.

[b]Original calculation was performed for an 80% capacity factor and a 1-year refueling interval. Data in this table were adjusted to a 75% capacity factor by using the following formula: discharge time = segment number x (0.80/0.75).

Table 3-3. Fuel mass flows[a] for the medium-enrichment uranium-233/thorium recycle fuel cycle (continued)

Segment	12	13	14	15	16	17	18	19	20	21
Region	C	A	B	C	A	B	C	A	B	C
Discharge time (yr)[b]	12.80	13.87	14.93	16.00	17.07	18.13	19.20	20.27	20.27	20.27
Thorium charged	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4	12,581.4
Uranium-235 loaded	4.7	4.6	4.6	4.6	4.6	4.6	4.6	4.6	4.6	4.6
Uranium-233 loaded	828.1	828.4	829.4	829.3	829.2	829.0	829.1	829.1	829.1	829.1
Total uranium loaded	7,023.0	7,025.8	7,033.9	7,033.1	7,032.1	7,030.8	7,031.2	7,031.4	7,031.6	7031.5
Total metal loaded	19,604.4	19,607.2	19,615.3	19,614.5	19,613.5	19,612.2	19,612.6	19,612.8	19,613.0	19,612.9
Thorium discharged	12,043.2	12,043.1	12,043.1	12,043.2	12,043.2	12,043.2	12,043.2	12,043.2	12,220.0	12,399.4
Bred uranium-233 discharged for recycle	286.4	286.4	286.4	286.4	286.4	286.4	286.4	286.4	236.0	148.5
Bred uranium-235 discharged for recycle	4.6	4.6	4.6	4.6	4.6	4.6	4.6	4.6	1.9	.4
Total bred uranium discharged for recycle	323.4	323.4	323.4	323.4	323.4	323.4	323.4	323.4	256.4	155.4
Uranium-233 discharged for credit	179.3	179.3	179.5	179.5	179.5	179.5	179.5	179.5	298.9	497.8
Uranium-235 discharged for credit	19.7	19.7	19.7	19.7	19.7	19.7	19.7	19.7	15.3	9.6
Total uranium discharged for credit	6,107.7	6,110.1	6,117.4	6,116.7	115.9	6,114.7	6,115.0	6,115.2	6,335.4	6,625.2
Total uranium discharged	6,431.1	6,433.5	6,440.8	6,440.1	6,439.3	6,438.1	6,438.4	6,438.6	6,591.8	6,780.6
Fissile plutonium retired	71.4	71.5	71.5	71.5	71.5	71.5	71.5	71.5	66.8	51.8
Total plutonium retired	106.9	106.9	107.0	107.0	107.0	106.9	107.0	107.0	93.9	66.6
Total metal discharged	18,581.1	18,583.5	18,590.9	18,590.3	18,589.4	18,588.2	18,588.5	18,588.8	18,905.7	19,246.6

[a]Mass flows are in kilograms.

[b]Original calculation was performed for an 80% capacity factor and a 1-year refueling interval. Data in this table were adjusted to a 75% capacity factor by using the following formula: discharge time = segment number x (0.80/0.75).

Table 3-4. HTGR mass-flow data for the medium-enrichment uranium-233/thorium recycle fuel cycle: equilibrium loadings at a 75% capacity factor normalized to 1,000-MWe reactor, annual refueling[a]

Isotope	Quantity[b] (kg/GWe)	
	Charged	Discharged
Fertile particle		
Thorium-232	8,827.6	8,450
Uranium-233	-	201
Uranium-235	-	3.2
Total uranium	-	226.9
Fission products[c]	-	~140
Fissile particle		
Uranium-233	581.9	125.9
Uranium-235	3.2	13.8
Total uranium	4,934.7	4,291.7
Plutonium fissile	-	50.2
Total plutonium	-	75.1
Fission products[c]	-	~550

[a]Factor $= \frac{1000\text{ MWe}}{1332\text{ MWe}} \times \frac{1\text{ year}}{1.07\text{ years}} = 0.7016$

[b]Data base; segment 15 from Table 3-3.

[c]Fission product quantities estimated.

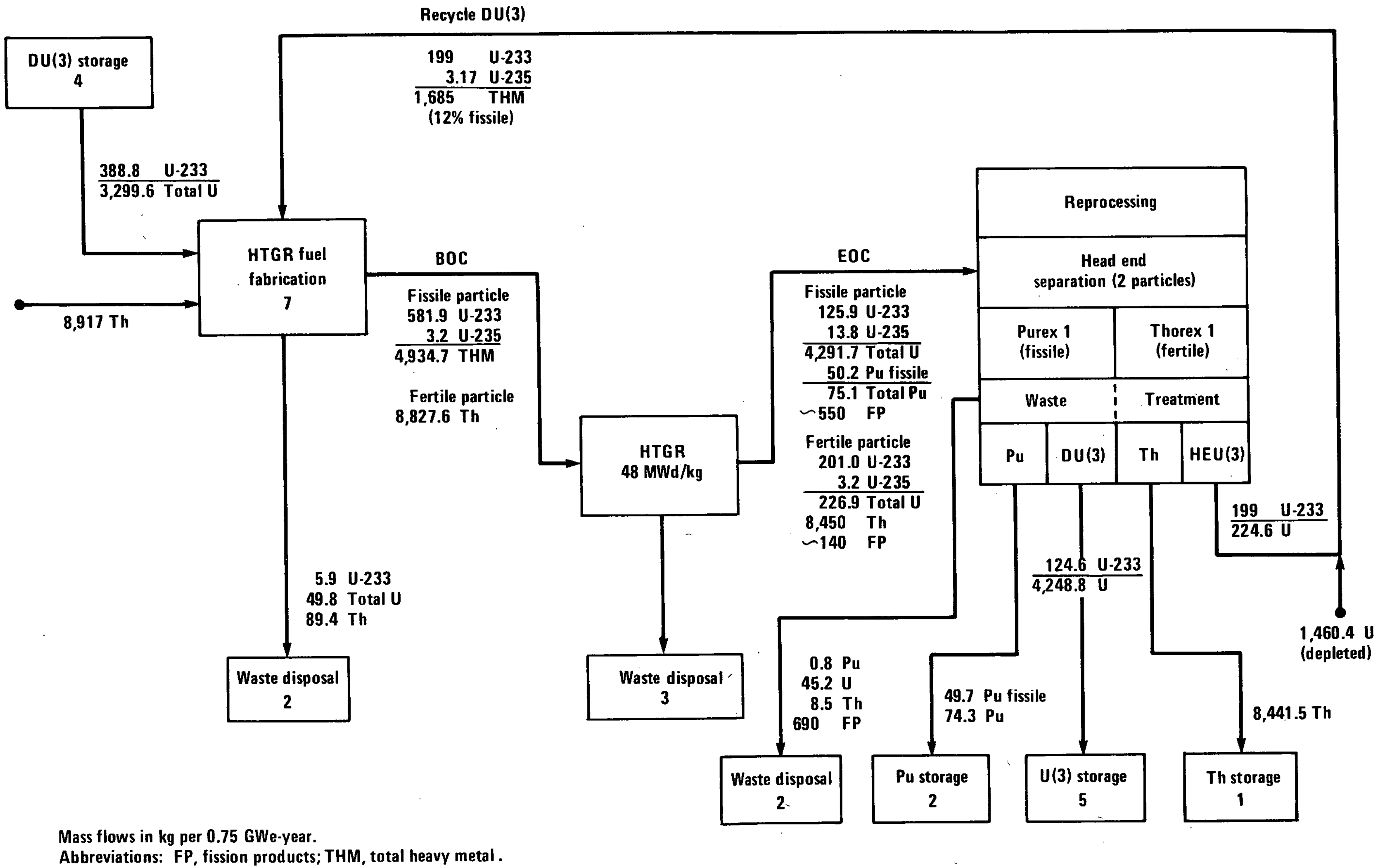


Figure 3-1. Material flow diagram for the HTGR denatured uranium-233/thorium fuel cycle.

REFERENCE FOR CHAPTER 3

1. General Atomic Company, General Atomic Standard Safety Analysis Report, GASSAR-6, GA-A13200, NRC Docket No. STN-50-535, February 1975.

Chapter 4

HIGH-ENRICHMENT URANIUM-235/THORIUM RECYCLE (SPIKED) FUEL CYCLE

4.1 DESCRIPTION

This reactor/fuel-cycle combination is a high-temperature gas-cooled reactor (HTGR) using high-enrichment fuel composed of three types of particles. The fissile particles are 93% uranium-235 for makeup and uranium-233 or uranium-235 for recycle mixed with thorium oxide fertile particles produced from new thorium. The spent fuel is reprocessed to separate the fissile and fertile particles. The recycled fissile particles of uranium-235 are retired. Recycle uranium-233 or uranium-235 particles and makeup uranium-235 fuel particles to be recycled are reprocessed to separate the plutonium, which is diverted to storage. The once-burned high-enrichment uranium-235 is recycled for one additional pass and then sent to secure storage. The fertile particles are reprocessed to separate the uranium-233 and the thorium. The uranium-233 produced from the reprocessing of fertile particles is mixed with the uranium-233 recovered from reprocessing the recycle fissile particle. All the recovered uranium-233 is recycled to refabrication; the recovered thorium is sent to 10-year interim storage. Wastes from reprocessing and from fuel fabrication are sent to a geologic waste repository. A radioactive spikant is added to uranium-233 and uranium-235 recovered in reprocessing during the final product homogenization before shipment to refabrication as recycle-fuel feed material.

The generalized reactor performance and design data specifications are summarized in Chapter 1. Data on fuel management are given in Section 4.1.4.

4.1.1 FUEL MECHANICAL DESIGN

For a complete description of the bases of the fuel mechanical design, see Section 4.2.1 of Reference 1.

4.1.2 FUEL NUCLEAR DESIGN

For a complete description of the bases of the fuel nuclear design, see Section 4.3 of Reference 1.

4.1.3 FUEL THERMAL-HYDRAULIC DESIGN

For a complete description of the bases for the fuel thermal-hydraulic design, see Section 4.4 of Reference 1.

4.1.4 FUEL MANAGEMENT

Fuel-management information is given in Table 4-1. The fresh fuel and spent fuel are characterized in Table 4-2, which includes data on the content of heavy-element isotopes for initial and equilibrium loadings and discharges. Fuel mass-flow data (charge and discharge) are given in Table 4-3. Reactor charge and discharge data for this fuel cycle are given in Table 4-4.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram of Figure 4-1 and are discussed in the following sections of Volume VII:

Enrichment	Chapter 3
Fabrication 7	Chapter 4
Reprocessing (Purex 1)	Section 5.1
Reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Depleted Uranium storage	Section 6.4
Uranium-235 storage	Section 6.6
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

4.2 SAFETY CONSIDERATIONS

Safety considerations for this fuel cycle are identical with those described in Section 2.2.

4.3 ENVIRONMENTAL CONSIDERATIONS

The nonradiological and radiological impacts of the HTGR plant using high-enrichment uranium-235/thorium recycle fuel are the same as those described in Section 2.3 for the medium-enrichment-once-through fuel cycle because the designs of the nuclear steam supply system and balance of plant are the same for both cases. The differences in fuel are not expected to change significantly the source term discussed in Section 2.3.6.1.

A possible difference in radiological impacts may be in occupational exposure because the spiked recycle characteristic of this fuel cycle could increase exposure in fresh-fuel-handling operations. On the other hand, the use of highly enriched fuel could decrease occupational exposure during plant maintenance because the highly enriched fuel would have less severe plateout activity than does medium-enriched fuel. The net effect cannot be quantified without some operational experience; it is not, however, expected to be significant.

4.4 LICENSING STATUS AND CONSIDERATIONS

The status and considerations for this fuel cycle are identical with those discussed in Section 2.4.

4.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The research and development programs required for the design, construction, and licensing of a commercial plant for this fuel cycle are identical with those outlined in Section 2.5.

Table 4-1. Parameters for the high-enrichment uranium-235/thorium recycle (spiked) fuel cycle[a]

Average capacity factor, %	75
Fuel form	Coated oxide or carbide particles
Fraction of core replaced annually	0.25
Enrichment-plant tails assay, %	0.2
Core power density, W/cm^3	7.1
Carbon-to-thorium ratio	
Initial core	180
Equilibrium reload	180
Fuel-rod diameter, cm	1.59
Average fuel temperature, ^{o}C	880
Maximum fuel temperature, ^{o}C	1,350
Core fuel loading, kg/GWe (initial core/equilibrium reload)	
Total heavy metal	45,750/12,200
Fissile	1,870/570
Burnup, MWd/MT	
Average	59,500
Peak	75,000
Conversion ratio	
Beginning of life (initial core)	0.65
After equilibrium fuel loading	0.72
Average during equilibrium	0.75
Yellowcake requirements, ST/GWe	
Initial core	508
Equilibrium annual	54
30-year total	2,281
30-year cumulative, net[b]	1,600
Separative-work requirements, 10^3 SWU/GWe	
Initial core	509
Equilibrium annual	54
30-year total	2,285
30-year cumulative, net[b]	1,600
Annual discharge, kg/GWe	
Fissile plutonium	1
Total plutonium	5
Uranium-235	79
Bred uranium-233	288
Total uranium	580
Total thorium	10,536

Table 4-1. Parameters for the high-enrichment uranium-235/thorium recycle (spiked) fuel cycle[a] (continued)

30-year cumulative discharge, kg/GWe[c]	
Fissile plutonium	42
Total plutonium	159
Uranium-235	2,810
Bred uranium-233	8,770
Total uranium	18,000
Total thorium	348,000

[a]Fissile material is 93% enriched uranium-235; fertile material is thorium; annual refueling, 4-year cycle. An external source is required for uranium-235 makeup.

[b]The 30-year cumulative net is equal to the 30-year total less a credit for fissile material recoverable at the end of life (five segments).

[c]The 30-year cumulative discharge is the sum of 30 annual discharges plus the partially consumed heavy metal in the reactor at the end-of-plant life.

Table 4-2. Characterization of HTGR fresh and spent fuel for the high-enrichment uranium-235/thorium recycle (spiked) fuel cycle

Cycle, years	4
Refueling method	Batch
Refueling frequency	Annual
Fuel-assembly characteristics	
Type	Oxide and carbide
Weight, kg	120
Length, m	0.79
Core load mass, kg HM/GWe	45,750
Annual reload mass at 75% capacity factor, kg HM/GWe	12,200
Design burnup,[a] MWd/MT	59,500
Dose rate at 1 m in air after 90 days, rem/hr	~4,900

Heavy-metal isotope content[b]

Isotope	Fresh fuel element (kg)		Discharged fuel element (kg)	
	Initial	Equilibrium	Initial	Equilibrium
Thorium-232	11.02	11.49	10.32	10.87
Uranium-232	--	2.21×10^{-4}	1.38×10^{-4}	2.30×10^{-4}
Uranium-233	--	0.285	0.26	0.31
Uranium-234	0.004	0.10	0.045	0.107
Uranium-235	0.47	0.30	0.059	0.080
Uranium-236	0.002	0.05	0.08	0.094
Uranium-238	0.029	0.028	0.024	0.021
Neptunium-237	--	--	0.007	0.009
Plutonium-238	--	--	0.004	0.003
Plutonium-239	--	--	9.5×10^{-4}	8.7×10^{-4}
Plutonium-240	--	--	3.4×10^{-4}	4.2×10^{-4}
Plutonium-241	--	--	4.0×10^{-4}	3.8×10^{-4}
Plutonium-242	--	--	--	--

[a]Discharge batch average.
[b]Multiply by 993 (fuel elements per GWe) for the isotope content in kilograms.

Table 4-3. Fuel mass flows (kg) for the high-enrichment uranium-235/thorium recycle (spiked) fuel cycle: core segments 1 through 13

Segment	1	2	3	4	5	6	7	8	9	10	11	12	13
Region	A	B	C	D	A	B	C	D	A	B	C	D	A
Discharge time (yr)	1.60	2.67	3.73	4.80	5.87	6.93	8.00	9.07	10.13	11.20	12.27	13.33	14.40
Thorium charged	15,194.5	14,832.0	14,832.0	14,564.6	15,194.5	14,832.0	14,832.0	14,564.6	15,194.5	14,832.0	14,832.0	14,564.6	15,194.5
Bred U-233 recycled	0.0	0.0	0.0	0.0	0.0	221.6	283.0	320.2	335.0	343.1	358.6	365.3	363.1
Bred U-235 recycled	0.0	0.0	0.0	0.0	0.0	1.4	4.1	7.7	11.7	9.6	16.2	20.3	24.2
Total bred uranium recycled	0.0	0.0	0.0	0.0	0.0	239.2	318.4	378.0	407.5	407.1	450.5	473.9	486.2
Total U-233 recycled	0.0	0.0	0.0	0.0	0.0	221.6	283.0	320.2	335.0	343.1	358.6	365.3	363.1
Total U-235 recycled	0.0	0.0	0.0	0.0	0.0	299.3	178.8	114.6	76.6	148.1	63.8	70.9	72.2
Total uranium recycled	0.0	0.0	0.0	0.0	0.0	646.7	615.2	609.6	597.9	739.8	562.9	592.5	599.1
U-235 makeup	676.9	660.8	660.8	648.8	1000.7	335.0	351.9	335.3	369.3	289.5	345.5	329.7	353.6
Total uranium makeup	726.9	709.6	709.6	696.8	1074.6	359.7	377.9	360.0	396.5	310.8	371.0	354.0	379.7
Total uranium loaded	726.9	709.6	709.6	696.8	1074.6	1006.4	993.2	969.5	994.5	1050.6	934.0	946.5	978.9
Total metal loaded	15,921.4	15,541.6	15,541.6	15,261.4	16,269.1	15,838.4	15,825.2	15,534.1	16,189.0	15,882.6	15,766.0	15,511.1	16,173.4
Thorium discharged	14,863.6	14,300.0	14,096.5	13,646.4	14,392.0	14,014.1	14,015.4	13,761.7	14,355.1	14,012.5	14,012.2	13,760.0	14,356.0
Bred U-233 discharged for recycle	227.3	290.2	328.4	343.5	351.9	367.8	374.6	372.4	388.7	381.7	383.5	378.5	393.7
Bred U-235 discharged for recycle	1.4	4.2	7.9	12.0	9.9	16.6	20.9	24.8	28.5	27.0	32.0	34.7	37.4
Total bred uranium discharged for recycle	245.3	326.5	384.6	417.9	417.5	462.0	486.0	498.7	530.4	517.1	538.2	543.2	569.9
Total U-233 discharged for recycle	227.3	290.2	328.4	343.5	351.9	367.8	374.6	372.4	388.7	381.7	383.5	378.5	393.7
Total U-235 discharged for recycle	306.9	183.4	117.5	78.5	151.9	65.5	72.7	74.0	82.3	69.2	82.4	83.0	89.5
Total uranium discharged for recycle	663.2	631.0	625.1	613.3	758.7	577.4	607.7	614.5	657.6	616.7	657.2	656.9	692.2
U-235 retired	0.0	0.0	0.0	0.0	0.0	44.8	26.7	16.4	10.0	21.2	7.3	7.8	7.4
Total uranium retired	0.0	0.0	0.0	0.0	0.0	177.5	155.6	143.0	130.8	211.6	70.9	74.7	71.2
Total uranium discharged	663.2	631.0	625.1	613.3	758.7	754.9	763.3	767.5	788.4	828.3	728.1	731.6	763.4
Total fissile plutonium discharged	1.2	1.4	1.6	1.8	2.6	2.6	2.5	2.3	2.2	2.8	1.6	1.6	1.7
Total plutonium discharged	2.1	3.5	5.4	7.6	9.5	10.2	9.3	8.3	7.9	10.0	6.1	6.3	6.6
Total metal discharged	15,528.9	14,934.5	14,727.0	14,267.2	15,120.3	14,779.1	14,738.0	14,527.5	15,151.4	14,850.8	14,746.4	14,497.9	15,126.0

Table 4-3. Fuel mass flows (kg) for the high-enrichment uranium-235/thorium recycle (spiked) fuel cycle: core segments 14 through 27 (continued)

Segment	14	15	16	17	18	19	20	21	22	23	24	25	26	27
Region	B	C	D	A	B	C	D	A	B	C	D	A	B	C
Discharge time (yr)	15.47	16.53	17.60	18.67	19.73	20.80	21.87	22.93	24.00	25.07	26.13	26.13	26.13	26.13
Thorium charged	14,832.0	14,832.0	14,564.6	15,194.5	14,832.0	14,832.0	14,564.6	15,194.5	14,832.0	14,832.0	14,564.6	15,194.5	14,832.0	14,832.0
Bred U-233 recycled	379.0	372.1	373.9	369.0	383.8	378.2	378.3	373.6	388.3	382.5	382.5	376.9	391.5	385.5
Bred U-235 recycled	27.8	26.3	31.2	33.8	36.4	38.8	37.8	40.5	42.2	43.9	45.0	44.3	45.9	47.0
Total bred uranium recycled	517.2	504.1	524.7	529.6	555.6	559.1	554.8	560.6	583.4	534.5	589.7	580.1	603.5	602.5
Total U-233 recycled	379.0	372.1	373.9	369.0	383.8	378.2	378.3	373.6	388.3	382.5	382.5	376.9	391.5	385.5
Total U-235 recycled	80.3	67.5	80.4	80.9	87.2	81.7	86.8	87.5	93.6	90.5	95.5	92.2	98.7	94.3
Total uranium recycled	641.2	601.3	640.1	640.5	674.9	659.4	668.5	669.1	701.4	690.6	704.2	658.3	722.2	708.4
U-235 makeup	296.7	334.9	317.9	344.1	308.4	331.4	312.2	341.9	304.0	320.0	313.5	340.3	302.9	319.7
Total uranium makeup	318.6	359.7	341.4	369.5	331.1	355.8	335.3	367.1	326.5	343.6	336.7	365.4	325.3	343.3
Total uranium loaded	959.8	961.0	982.1	1,010.0	1,006.0	1,015.2	1,003.7	1,036.2	1,027.8	1,034.2	1,040.8	1,053.7	1,047.5	1,051.7
Total metal loaded	15,791.8	15,793.0	15,546.7	16,204.5	15,838.0	15,847.2	15,568.3	16,230.7	15,859.8	15,866.2	15,605.4	16,248.2	15,879.5	15,883.7
Thorium discharged	14,014.6	14,016.4	13,765.8	14,363.0	14,022.2	14,023.7	13,772.0	14,368.9	14,027.2	14,028.3	13,776.5	14,574.8	14,426.8	14,628.4
Bred U-233 discharged for recycle	387.8	388.0	383.2	398.2	392.3	392.3	386.6	401.5	395.4	395.2	389.4	403.6	399.4	393.8
Bred U-235 discharged for recycle	39.8	38.9	41.6	43.3	45.0	46.2	45.4	47.1	48.2	49.0	49.4	48.1	48.3	48.1
Total bred uranium discharged for recycle	573.5	569.0	575.0	598.4	589.5	604.8	595.0	619.0	618.0	621.7	618.2	626.1	623.3	616.0
Total U-233 discharged for recycle	387.8	388.0	383.2	398.2	392.3	392.3	386.6	401.5	395.4	395.2	389.4	403.6	399.4	393.8
Total U-235 discharged for recycle	83.8	89.0	89.8	96.0	92.8	98.0	94.5	101.2	96.7	100.3	100.0	135.0	170.6	251.6
Total uranium discharged for recycle	676.3	685.6	686.2	719.4	708.3	722.2	705.9	740.8	726.6	736.3	730.8	778.1	798.1	863.5
U-235 retired	8.2	6.5	7.8	7.6	8.2	7.0	8.1	7.6	8.6	7.8	8.5	12.6	21.6	30.2
Total uranium retired	78.4	61.5	73.5	70.3	75.5	63.5	71.8	68.3	74.2	68.6	71.7	73.3	88.9	90.1
Total uranium discharged	754.7	747.1	759.8	789.6	783.9	785.7	777.7	809.1	800.7	802.9	802.6	851.3	887.1	953.6
Total fissile plutonium discharged	1.6	1.6	1.7	1.7	1.7	1.7	1.7	1.8	1.8	1.7	1.8	1.6	1.3	.9
Total plutonium discharged	6.4	6.1	6.7	7.0	7.0	7.0	7.0	7.5	7.5	7.6	7.9	5.4	3.4	1.6
Total metal discharged	14,775.7	14,769.7	14,532.2	15,159.6	14,813.1	14,816.4	14,556.7	15,185.5	14,835.4	14,838.8	14,586.9	15,431.6	15,317.2	15,583.6

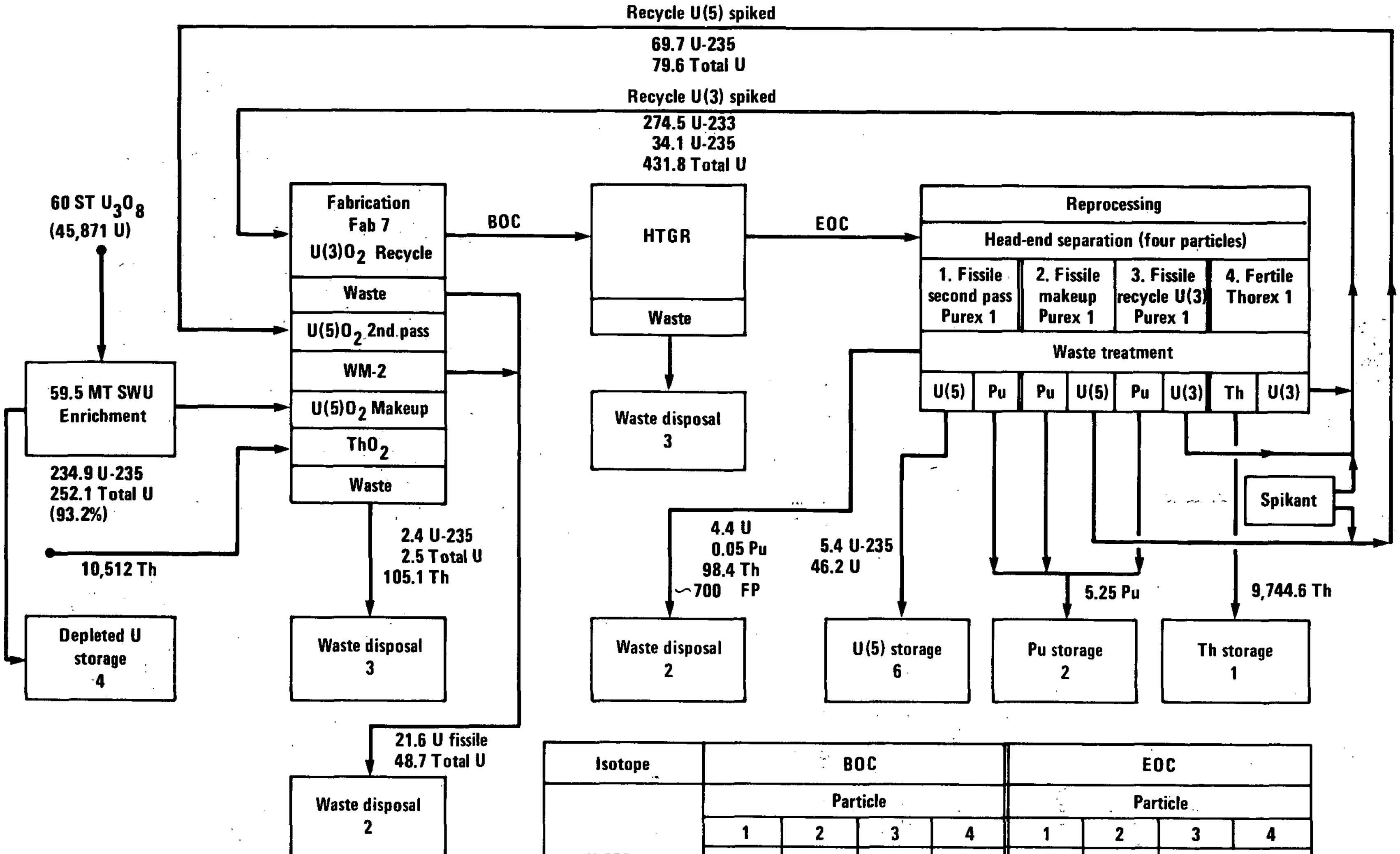
Table 4-4. HTGR mass-flow data for the high-enrichment uranium-235/thorium recycle (spiked) fuel cycle: equilibrium loadings at a 75% capacity factor[a,b]

Isotope	Quantity (kg/GWe)	
	Charged	Discharged
Thorium-232	10,407	9,843
Uranium-233	265.4	277.3
Uranium-235	317	110.3
Uranium-total	707.3	563.3
Plutonium, fissile	--	1.2
Plutonium, total	--	5.3
Total heavy metal	11,114.3	10,411.6
Fission products	--	~700[c]

[a]There are 993 fuel elements per GWe (4-year fuel life).

[b]Normalized from data for a 1,332-MWe reactor on a 1.07-year cycle. Data from Table 4-3 for year 20.8 for beginning of cycle and year 25.07 for end of cycle.

[c]Fission products estimated as difference between charge and discharge quantities.



Isotope	BOC Particle 1	BOC Particle 2	BOC Particle 3	BOC Particle 4	EOC Particle 1	EOC Particle 2	EOC Particle 3	EOC Particle 4
U-233			265.4				277.3	
U-235	59.3	232.5	27.2		5.5	70.4	34.4	
Total U	70.7	249.6	392.3		46.7	80.4	436.2	
Pu fissile					—	1.2	—	
Total Pu					—	5.3	—	
Thorium				10,407				9,843
FP						~700		

Notes:

1. Mass flows in kg per GWe-yr.
2. NASAP/INFCE data base, years 21.33 and 25.6 normalized from a 1,332-MWe reactor, 1.07-year cycle.
3. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products.

Figure 4-1. Material flow diagram for the HTGR with high-enrichment uranium-235/thorium recycle (spiked) fuel cycle.

REFERENCE FOR CHAPTER 4

1. General Atomic Company, General Atomic Standard Safety Analysis Report, GASSAR-6, GA-A13200, NRC Docket No. STN-50-535, February 1975.

Chapter 5

HIGH-ENRICHMENT URANIUM-233/THORIUM RECYCLE (SPIKED) FUEL CYCLE

5.1 DESCRIPTION

This reactor/fuel-cycle combination is a high-temperature gas-cooled reactor (HTGR) using self-spiked, high-enrichment uranium-233 fissile particles and thorium fertile particles. The spent fuel is reprocessed to separate the thorium and the uranium-233. All recovered uranium-233 is recycled to refabrication, where it is mixed with makeup uranium-233 from a secure storage facility for the manufacture of fissile particles. The fertile particles are fabricated from new thorium. The recovered thorium is sent to 10-year interim storage. Wastes from reprocessing and from fuel fabrication are sent to a geologic waste repository.

The generalized reactor performance and design data specifications are summarized in Chapter 1. Data on fuel management are given in Section 5.1.4.

5.1.1 FUEL MECHANICAL DESIGN

For a complete description of the bases of the fuel mechanical design, see Section 4.2.1 of Reference 1.

5.1.2 FUEL NUCLEAR DESIGN

For a complete description of the bases of the fuel nuclear design, see Section 4.3 of Reference 1.

5.1.3 FUEL THERMAL-HYDRAULIC DESIGN

For a complete description of the bases for the fuel thermal-hydraulic design, see Section 4.4 of Reference 1.

5.1.4 FUEL MANAGEMENT

Fuel-management information is given in Table 5-1. The fresh fuel and spent fuel are characterized in Table 5-2, which includes data on the content of heavy-element isotopes for initial and equilibrium loadings and discharges. Fuel mass-flow data (charge and discharge) are given in Table 5-3. Reactor charge and discharge data are given in Table 5-4.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram of Figure 5-1 and are discussed in the following sections of Volume VII:

Fuel fabrication 7	Chapter 4
Reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

Table 5-1. Parameters for the high-enrichment uranium-233/thorium recycle (spiked) fuel cycle[a]

Average capacity factor, %	75
Fuel form	Coated oxycarbide or oxide particles
Fraction of core replaced annually	0.25
Enrichment-plant tails assay, %	0.2
Core power density, W/cm^3	5.0
Carbon-to-thorium ratio	
Initial core	150
Equilibrium reload	150
Fuel-rod diameter, cm	1.59
Average fuel temperature, °C	880
Maximum fuel temperature, °C	1,350
Core fuel loading, kg/GWe (initial core/equilibrium reload)	
Total heavy metal	78,600/19,940
Fissile	2,072/630 (83 net)[b]
Burnup, MWd/MT	
Average	35,500
Peak	44,730
Conversion ratio	
Beginning of life (initial core)	0.98
After equilibrium fuel loading	0.91
Average during equilibrium	0.92
Yellowcake requirements	None
Separative-work requirements	None
Annual discharge, kg/GWe	
Fissile plutonium	0.3
Total plutonium	3
Uranium-235	78
Bred uranium-233	491
Total uranium	850
Total thorium	18,400
30-year cumulative discharge, kg/GWe[c]	
Fissile plutonium	7
Total plutonium	70
Uranium-235	2,240
Bred uranium-233	15,930
Total uranium	26,300
Total thorium	608,000

[a]Fissile material is enriched uranium-233; fertile material is thorium; annual refueling, 4-year cycle. An external source is required for uranium-233 makeup.

[b]Bred fissile material required annually from an external source.

[c]The 30-year cumulative discharge is the sum of 30 annual discharges plus the partially consumed heavy metal in the reactor at the end-of-plant life.

Table 5-2. Characterization of HTGR fresh and spent fuel for the high-enrichment uranium-233/thorium recycle (spiked) fuel cycle

Cycle, years	4
Refueling method	Batch
Refueling frequency	Annual
Fuel-assembly characteristics	
Type	Oxide
Weight, kg	122
Length, m	0.79
Core load mass, kg HM/GWe	78,600
Annual reload mass at 75% capacity factor, kg HM/GWe	19,940
Design burnup,[a] MWd/MT	35,500
Dose rate at 1 m in air after 90 days, rem/hr	~4,900

Heavy-metal isotope content[b]

Isotope	Fresh fuel element (kg)		Discharged fuel element (kg)	
	Initial	Equilibrium	Initial	Equilibrium
Thorium-232	13.48	13.48	13.48	12.99
Uranium-232	--	2.21×10^{-4}	4.90×10^{-5}	2.3×10^{-4}
Uranium-233	0.31	0.39	0.38	0.37
Uranium-234	0.094	0.17	0.126	0.155
Uranium-235	0.032	0.057	0.040	0.056
Uranium-236	0.010	0.045	0.018	0.049
Neptunium-237	--	--	6.0×10^{-4}	0.005
Plutonium-238	--	--	7.6×10^{-5}	0.0017
Plutonium-239	--	--	3.3×10^{-6}	1.73×10^{-4}
Plutonium-240	--	--	4.1×10^{-7}	6.4×10^{-5}
Plutonium-241	--	--	-	3.8×10^{-5}
Plutonium-242	--	--	-	1.0×10^{-5}

[a]Discharge batch average.
[b]Multiply by 993 (fuel elements per GWe) for the isotope content in kilograms.

Table 5-3. Fuel mass flows (kg) for the high-enrichment uranium-233/thorium recycle (spiked) fuel cycle

Segment	1	2	3	4	5	6	7	8	9	10	11	12	13	14
Region	A	B	C	D	A	B	C	D	A	B	C	D	A	B
Discharge time (yr)	1.07	2.13	3.20	4.27	5.34	6.40	7.47	8.54	9.61	10.67	11.74	12.81	13.88	14.94
Thorium charged	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7
Total U-235 makeup	62.4	62.4	62.4	62.4	78.5	76.1	74.9	74.2	76.9	89.1	89.6	90.0	90.2	92.0
U-233 makeup	628.0	628.0	628.0	628.0	789.3	773.8	757.0	738.8	750.5	759.4	760.5	762.0	763.0	759.7
Total uranium makeup	886.0	886.0	886.0	886.0	1,113.6	1,104.4	1,091.6	1,075.7	1,101.7	1,159.4	1,165.4	1,170.5	1,173.2	1,178.3
Total uranium loaded	886.0	886.0	886.0	886.0	1,113.6	1,104.4	1,091.6	1,075.7	1,101.7	1,159.4	1,165.4	1,170.5	1,173.2	1,178.3
Total metal loaded	26,342.7	26,342.7	26,342.7	26,342.7	26,570.3	26,561.1	26,548.3	26,532.4	26,558.4	26,616.1	26,622.1	26,627.2	26,629.9	26,635.0
Thorium discharged	25,193.5	24,939.7	24,692.8	24,451.1	24,472.7	24,480.4	24,484.5	24,486.8	24,489.9	24,494.4	24,497.0	24,499.3	24,501.2	24,503.1
Total U-233 discharged for recycle	625.3	626.5	628.6	630.2	666.0	667.1	666.3	664.5	668.7	672.0	673.7	675.4	676.9	677.5
Total U-235 discharged for recycle	61.3	61.9	63.3	65.0	80.1	80.6	80.8	80.7	83.2	92.4	93.1	93.6	93.8	95.0
Total uranium discharged for recycle	895.1	907.9	921.0	932.8	1,029.4	1,035.6	1,037.3	1,036.3	1,052.0	1,091.9	1,097.6	1,102.2	1,105.2	1,111.9
Total uranium discharged	895.1	907.9	921.0	932.8	1,029.4	1,035.6	1,037.3	1,036.3	1,052.0	1,091.9	1,097.6	1,102.2	1,105.2	1,111.9
Total fissile plutonium discharged	0.0	0.0	0.1	0.1	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.3	0.3	0.3
Total plutonium discharged	0.1	0.4	0.8	1.4	1.7	1.8	2.0	2.1	2.3	2.6	2.7	2.8	2.9	3.1
Total metal discharged	26,088.7	25,848.0	25,614.7	25,385.3	25,503.8	25,517.8	25,523.8	25,525.2	25,544.2	25,588.9	25,597.4	25,604.3	25,609.4	25,618.0

Table 5-3. Fuel mass flows (kg) for the high-enrichment uranium-233/thorium recycle (spiked) fuel cycle (continued)

Segment	15	16	17	18	19	20	21	22	23	24	25	26	27	28
Region	C	D	A	B	C	D	A	B	C	D	A	B	C	D
Discharge time (yr)	16.01	17.08	18.15	19.21	20.28	21.35	22.42	23.48	24.55	25.62	26.68	26.68	26.68	26.68
Thorium charged	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7	25,456.7
Total U-235 makeup	101.2	101.9	102.5	102.5	103.4	110.1	110.7	111.3	111.4	111.4	116.2	116.7	117.4	117.5
U-233 makeup	765.8	767.4	770.5	770.6	767.4	772.4	773.7	776.8	777.7	773.7	778.4	779.4	782.3	783.2
Total uranium makeup	1,221.4	1,226.9	1,233.4	1,234.4	1,235.5	1,268.8	1,273.2	1,279.7	1,281.4	1,278.5	1,305.3	1,308.8	1,315.3	1,316.9
Total uranium loaded	1,221.4	1,226.9	1,233.4	1,234.4	1,235.5	1,268.8	1,273.2	1,279.7	1,281.4	1,278.5	1,305.3	1,308.8	1,315.3	1,316.9
Total metal loaded	26,678.1	26,683.6	26,690.1	26,691.1	26,692.2	26,725.5	26,729.9	26,736.4	26,738.1	26,735.2	26,762.0	26,765.5	26,772.0	26,773.6
Thorium discharged	24,506.4	24,508.2	24.509.9	24,511.2	24,512.5	24,514.9	24,516.2	24,517.5	24,518.5	24,519.3	24,521.1	24,752.2	24,985.3	25,220.1
Total U-233 discharged for recycle	680.0	681.3	682.9	683.9	684.2	686.1	687.1	688.5	689.4	689.2	690.9	706.0	725.9	750.7
Total U-235 discharged for recycle	101.6	102.2	102.7	102.7	103.2	107.9	108.3	108.9	108.9	108.8	112.2	113.8	115.6	116.7
Total uranium discharged for recycle	1,142.0	1,146.5	1,151.8	1,152.8	1,156.0	1,179.4	1,183.0	1,187.3	1,188.9	1,189.3	1,208.1	1,228.6	1,254.8	1,282.7
Total uranium discharged	1,142.0	1,146.5	1,151.0	1,152.8	1,156.0	1,179.4	1,183.0	1,187.3	1,188.9	1,189.3	1,208.1	1,228.6	1,254.8	1,282.7
Total fissile plutonium discharged	0.3	0.3	0.3	0.3	0.3	0.4	0.4	0.4	0.4	0.4	0.4	0.2	0.1	0.0
Total plutonium discharged	3.5	3.6	3.7	3.7	3.9	4.3	4.4	4.4	4.5	4.6	5.0	3.1	1.5	0.4
Total metal discharged	25,651.9	25,658.3	25,664.6	25,667.7	25,672.4	25,698.6	25,703.6	25,709.2	25,711.9	25,713.2	25,734.1	25,983.9	26,241.6	26,503.2

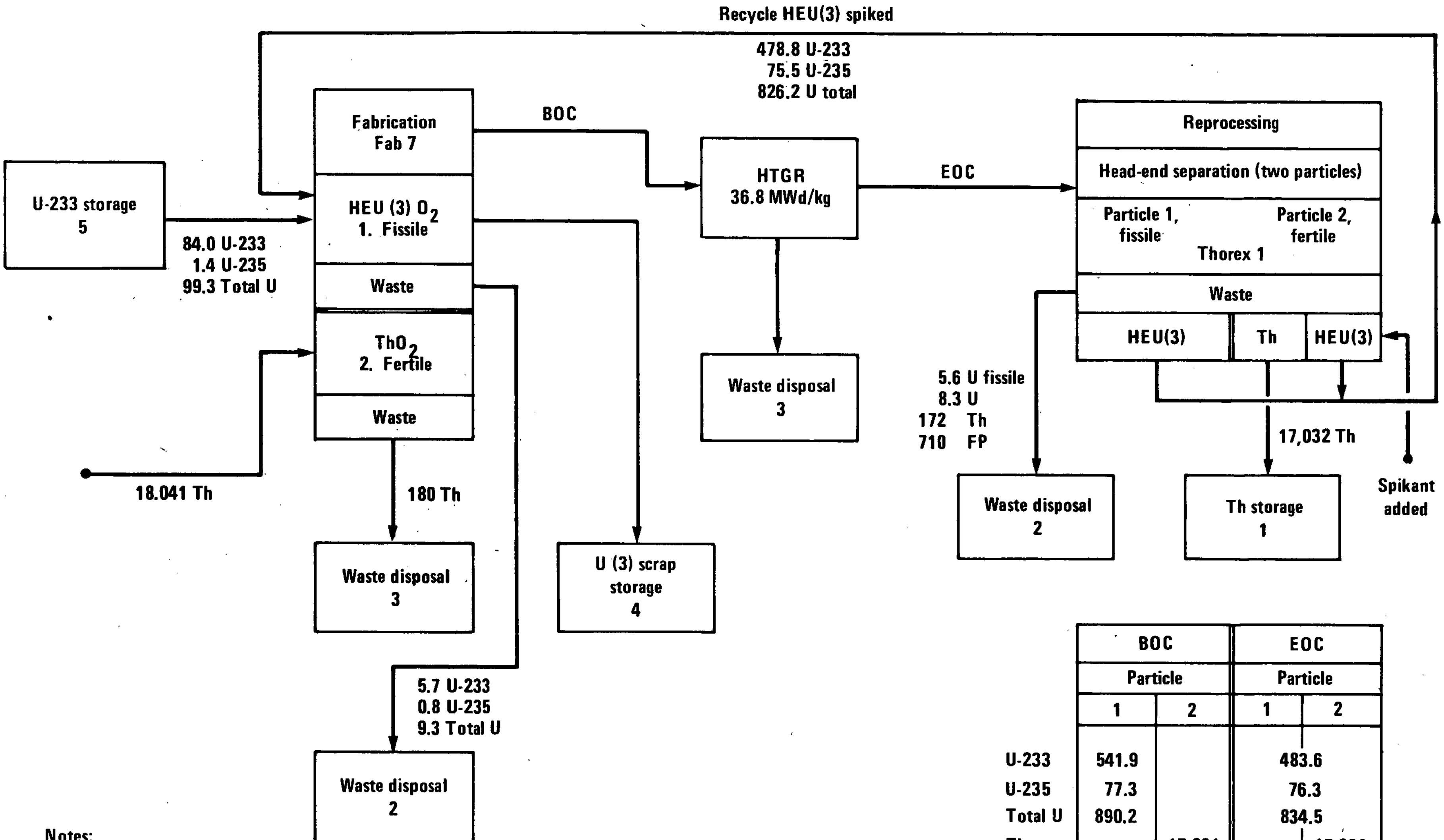
Table 5-4. HTGR mass-flow data for the high-enrichment uranium-233/thorium recycle (spiked) fuel cycle: equilibrium loadings at a 75% capacity factor[a,b]

Isotope	Quantity (kg/GWe)	
	Charged	Discharged
Thorium-232	17,861	17,204
Uranium-233	514.9	483.6
Uranium-235	77.3	76.3
Uranium, total	890.2	834.5
Fission products		710[c]

[a]There are 993 fuel elements per GWe (4-year fuel life).

[b]Normalized from data for a 1,332-MWe reactor operating on a 1.07-year cycle. Data from Table 5-3 for years 21.35 and 25.62 (beginning and end of cycle, respectively).

[c]Fission products estimated as the difference between charge and discharge quantities.



	BOC		EOC	
	Particle		Particle	
	1	2	1	2
U-233	541.9		483.6	
U-235	77.3		76.3	
Total U	890.2		834.5	
Th		17,861		17,204
FP			~710	

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Data normalized from a 1,332-MWe reactor, 1.07-year cycle; NASAP/INFCE data base years 20.8 and 25.07.
3. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products.

Figure 5-1. Material flow diagram for the HTGR with high-enrichment uranium-233/thorium recycle (spiked) fuel cycle.

5.2 SAFETY CONSIDERATIONS

Safety considerations for this fuel cycle are identical with those described in Section 2.2.

5.3 ENVIRONMENTAL CONSIDERATIONS

The comments made in Sections 2.3 and 3.3 are valid for this reactor/fuel-cycle combination with one possible exception. The core power density for this cycle is lower (5 vs. 7.1 W/cm^3) and the carbon-to-thorium ratio is higher (500 vs. 300 for equilibrium reload). Both of these aspects would lead to a better retention of fission products in the core because of the lower fuel temperatures and the increased retention capacity of the graphite. Although this difference has not been quantitatively assessed it can be concluded that the radiological environmental impacts of this reactor/fuel-cycle combination can be less severe than those of the medium-enriched uranium-233/thorium recycle case described in Chapter 3. The nonradiological impact is the same.

5.4 LICENSING STATUS AND CONSIDERATIONS

The status and considerations for this fuel cycle are identical with those discussed in Section 2.4.

5.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The research and development program required for the design, construction, and licensing of a commercial plant for this fuel cycle are identical with those outlined in Section 2.5.

REFERENCE FOR CHAPTER 5

1. General Atomic Company, General Atomic Standard Safety Analysis Report, GASSAR-6, GA-A13200, NRC Docket No. STN-50-535, February 1975.

Chapter 6

GAS-TURBINE HIGH-TEMPERATURE GAS-COOLED REACTOR

6.1 INTRODUCTION

A program to design and develop a commercial gas-turbine high-temperature gas-cooled reactor (HTGR) power plant has been under way at the General Atomic Company for several years with support from the Department of Energy (DOE), the utilities, and manufacturing companies. The approach is based on proceeding from the current HTGR technology base through a comprehensive development program for power-conversion system components of a dry-cooled nuclear demonstration plant that would be replicated in follow-on commercial-plant designs.

The development and utilization of a helium-turbine power-conversion system operating in a direct cycle on the hot helium delivered by the HTGR core have been shown to be technically feasible and to have substantial advantages. International preliminary design studies and development work have been in progress since 1970. The work in the United States has been done by General Atomic, by the Power Systems and the Pratt & Whitney Divisions of United Technologies Corporation, and by the Gas Turbine Projects Division of the General Electric Company; it has been supported by the DOE, the utilities, and the manufacturers. In Europe, the work is being conducted under the High-Temperature Helium Turbine (HHT) project led by General Atomic affiliates, Hochtemperatur-Reaktorbau and Kernforschungsanlage, with major industrial participation, and support by the Federal Republic of Germany and by the Swiss government. Cooperation between the General Atomic and HHT projects was initiated in 1973 and is currently conducted under an exchange agreement.

The gas-turbine industry has already established as state of the art for heavy-duty gas turbines the level of temperatures, unit frame size, and much of the basic technology needed for high-temperature helium turbines suitable for use with advanced gas-turbine HTGRs.

The gas-turbine HTGR offers major improvements in plant simplification, lower capital cost, increased efficiency, and waste-heat rejection. Heat is rejected either by economical dry-cooling towers, combined wet and dry cooling, or optionally by a low-temperature secondary Rankine power cycle (binary-cycle plant) that generates additional power with subsequent wet and dry or wet-cooling heat rejection. With an 850°C turbine-inlet temperature, the dry-cooled plant will have a 40% efficiency and the binary cycle a 48% efficiency.

The gas-turbine HTGR plant combines the existing HTGR core with closed-cycle helium-turbine power-conversion loops that operate on the reactor-coolant helium. The power-conversion loops (PCLs) are integrated into the prestressed-concrete reactor vessel (PCRV) for both safety and economic reasons: this design eliminates the necessity of providing burst protection for large external metallic pressure vessels and ducts. The PCRV is located inside a containment building that, together with the PCRV, incorporates safety features to limit loss of primary coolant and to limit missile damage in the event of failures in the turbomachinery, shaft seals, generator, heat exchangers, and other components.

The gas-turbine HTGR system differs from the steam-cycle HTGR system described in Chapter 1 as follows:

1. The six helium circulators and steam generators are replaced by three turbocompressors, recuperators, and precoolers all still inside the PCRV and secondary containment.
2. The steam piping system penetrating the PCRV and secondary containment is replaced by the turbocompressor rotating shaft coupled to the shaft of the electric generator located inside the containment. Thus the steam piping system and steam turbine are eliminated.
3. The gas-turbine HTGR helium-coolant core inlet and outlet temperatures are higher by 170° and 130°C, respectively. A minor change in fuel pin arrangement keeps the peak fuel temperature at about the same level as for the steam cycle plant.
4. If the binary-cycle option is used, the plant would contain an ammonia turbogenerator building. The cycle would convert heat in the water piped from the precooler to electricity, providing about 25% of the total plant output.

The perceived advantages of the gas-turbine HTGR are the same as those of the steam-cycle HTGR. However, the following unique advantages are attributed to the gas-turbine design:

1. The gas-turbine HTGR rejects its waste heat at a high temperature, making dry cooling economically feasible and thus allowing plant siting in arid areas or in areas with limited water supplies. In addition, the high rejection temperature permits the addition of an ammonia bottoming cycle that increases plant efficiency from 40% to 48%, thus further conserving fuel resources.
2. The load-following capability of the gas-turbine HTGR is better than that of a steam plant. In addition, the drop-load-recovery characteristic allows load following as rapid as 80% in 5 seconds, a feature unique to the gas-turbine HTGR concept.
3. The modular approach associated with multiple power-conversion loops ensures a high capability for part-load operation.
4. The direct-cycle concept eliminates the complex secondary systems needed in a steam plant since power conversion is in the primary system.
5. The high-pressure-differential high-temperature heat exchangers (steam generators) of a steam plant are replaced with low-pressure-differential, low-temperature (less than 1,000°F) heat exchangers in the gas-turbine HTGR. In addition, a heat-exchanger leak will result in a primary-to-primary leak only in the recuperator and a leak from the primary to the closed secondary loop circulating-water system only in the precooler.

The disadvantages of the gas-turbine HTGR as compared with a steam plant are as follows:

1. Large rotating mass in the primary system (the primary system is protected by a containment ring that is an integral part of the turbomachinery structure)
2. Potential oil ingress from the bearing-lubrication system (protected against by redundant seal system with backup scavenge pumps and "last-chance" oil baffles)
3. Maintenance of contaminated turbomachinery (considered in the turbomachinery design, installation, and tooling to minimize plant downtime and personnel exposure)

6.2 GENERAL DESCRIPTION

6.2.1 PLANT LAYOUT

The conceptual plot plan shown in Figure 6-1 shows the general layout of buildings and dry-cooling towers for a twin 3,000-MWt gas-turbine HTGR plant. The reactor service building and fuel-storage facilities are shared by the two reactor units. Each unit has a separate control building and safety-related auxiliaries. A runway system is provided for turbomachinery and generator handling. Space is allocated on the plot plan for an ammonia-turbine building should the binary-cycle option be selected.

Based on the utilization of an existing 3,000-MWt core design, the gas-turbine HTGR embodies three power-conversion loops, each rated at 1,000 MWt. The simplified isometric diagram of the reactor and primary system (Figure 6-2) shows the core, turbomachinery, heat exchangers, and entire helium inventory enclosed in the PCRV. The isometric view illustrates the integrated approach for the gas-turbine plant; changes in the major components (particularly the precooler) made since Figure 6-2 was prepared are discussed below.

The main cycle parameters for the nonintercooled plant are given in the simplified loop diagram (Figure 6-3). As shown in this diagram, each loop includes a single-shaft gas turbine, a recuperative gas-to-gas heat exchanger, and a precooler (gas-to-water exchanger) for cycle heat rejection. As shown in the plan view of the prestressed-concrete reactor vessel (Figure 6-4), the three power-conversion loops are located symmetrically around and below the central core cavity. The three turbomachines are oriented in a delta arrangement and the heat exchangers are installed in vertical cavities within the PCRV sidewalls, two for each loop. This orientation of the major components results in a minimum reactor-vessel diameter, this being economically important since the vessel is the single most costly item in the plant. The elevation views through the PCRV shown in Figures 6-5 and 6-6 illustrate the helium-gas flow path within the primary system. The components are connected by large internal ducts inside the prestressed-concrete reactor vessel. The horizontal turbomachine cavities are located directly below their loop heat exchangers. The recuperator is positioned directly above the turbine exhaust, and the precooler is above the compressor inlet. A summary of the main features of the gas-turbine HTGR power plant is given in Table 6-1.

6.2.2 POWER-CONVERSION-LOOP COMPONENTS

6.2.2.1 Helium Turbomachine

Preliminary design of the turbomachinery for the gas-turbine HTGR plant has been done by the Power Systems Division and the Pratt and Whitney Aircraft Division of United Technologies Corporation. A simple and rugged arrangement consisting of a single-shaft, direct-drive turbomachine was chosen for the gas-turbine HTGR. A simplified cross section of the 400-MWe, 60-Hz machine is shown in Figure 6-7; the main features are outlined in Table 6-2. The design and high-performance predictions for this machine reflect the influence of technology from demonstrated advanced-technology industrial gas turbines. The 400-MWe helium turbomachine has 18 compressor stages (for a pressure ratio of 2.5 with a gas of low molecular weight) and 8 turbine stages. The rotor is of welded construction. Welded rotors have a long, successful history in Europe for both gas and steam turbines. With the 60,800-kg (67-ton) rotor supported on two journal bearings (with state-of-the-art loading and peripheral speed), the overall length of the machine is 11.3 meters (37 feet). The overall diameter is 3.5 meters (11.5 feet). The overall machine weighs 276,800 kg (305 tons).

Rotor burst protection is incorporated into the machine design in the form of containment rings around the rotor-bladed sections of the compressor and turbine (Figure 6-7). Man-access cavities are provided in the PCRV for inspection and limited maintenance work on the journal bearings, which are of the multiple, tilting-pad, oil-lubricated type. The spaces in which the bearings are located are isolated from the main cycle working fluid by shielding (purged gas from the purification system is used to give an acceptable radiological environment for man access). The drive to the generator is from the compressor end of the turbomachine, and the thrust bearing is located outside the reactor vessel to facilitate inspection and maintenance.

For a single-shaft helium turbomachine with a net power output of 400 MWe, the rotating section is compact and is substantially smaller than an equivalent air-breathing machine because of the high degree of pressurization (particularly at the turbine exit) and because the enthalpy drop in the helium turbine is many times greater (i.e., increased specific power). The external dimensions of the 400-MWe helium-gas turbine are similar to those of an air-breathing, advanced, open-cycle industrial gas turbine in the 100-MWe range. The fact that the helium turbine (particularly the rotor assembly and casings) is comparable in size with existing machines substantiates the claim that conventional fabrication methods and facilities can be used.

The turbomachinery is coupled to an all-water-cooled generator that is located inside the containment building to eliminate shaft penetration of the containment.

6.2.2.2 Heat Exchangers

Tubular construction was selected for both the recuperator and precooler in the gas-turbine plant. The main reason for this selection was that it represents the only type of construction that has been proved to have the structural integrity needed for long-life electrical utility power service.

Initially, straight-tube axial-counterflow configurations were selected for both the recuperator and precooler, and this is reflected in the isometric of the primary system shown in Figure 6-2. The current recuperator in the reference plant design is of straight-tube design and embodies a modular assembly having many heat-transfer elements. For this gas-to-gas heat exchanger, inspection and repair are done at the module level. The present recuperator configuration is shown in Figure 6-8.

In the plant layout shown in Figure 6-5, the helical precooler design is shown installed in the PCRV. The helical precooler configuration is shown in Figure 6-9. Heat-exchanger dimensions and weights are given in Table 6-3.

A ground rule for the heat exchangers is that they must be designed to operate for the full life of the plant. Both units will be lowered into the PCRV cavities by a system of hydraulic jacks during construction; they are expected to remain in place during the life of the plant. In both exchanger designs, provision is made for replacement and for maintenance and repair. In the case of a failed heat-transfer element (i.e., a module in the case of the recuperator and a tube in the helical precooler assembly), plugging will be performed from outside the reactor vessel.

Even though the single-phase working fluids (helium and water) can realize relatively high heat-transfer coefficients, large surface areas are necessary because of the high thermal conductance requirements associated with the large heat-transfer rates. However, the modest metal temperatures and internal pressure differentials, compared with modern steam generators, permit the use of code-approved lower grade alloys of reduced cost. The ferritic materials selected for both exchangers have been used extensively in industrial and nuclear-plant heat exchangers. Though the exchanger assemblies are large, state-of-the-art manufacturing methods can be used, and the modular approach in the case of the recuperator eases the fabrication, handling, and assembly. The overall size and weight of both the recuperator and precooler are similar to those of contemporary steam generators. Transport, handling, and installation techniques developed for these units will be applicable to the heat exchangers for the gas-turbine HTGR.

6.2.3 CYCLE PARAMETERS

The plant performance is based on International Standards Organization (ISO) day conditions of 15°C (59°F) and assumes heat rejection to the atmosphere via a natural-draft dry-cooling tower. Figure 6-3 is the cycle diagram for the 3,000-MWt gas-turbine HTGR. Table 6-4 gives cycle conditions around the loop for the dry-cooled cycle. If an ammonia bottoming cycle is added, plant efficiency increases to 47.9%. Table 6-5 gives the cycle conditions around the loop for the binary cycle.

6.2.4 CORE AND FUEL FEATURES FOR THE GAS-TURBINE HTGR

The gas-turbine HTGR is designed to accommodate the same basic core design as the steam-cycle HTGR plant. The same fuel-cycle alternatives are available for the two plant designs. The primary differences in core design and performance characteristics are related to the temperatures of the helium coolant entering and leaving the core.

The average core-coolant exit temperature is 850°C (1,560°F) for the gas-turbine HTGR and 692°C (1,280°F) for the steam-cycle plant. The core-inlet temperatures are 500°C (930°F) and 318°C (605°F) for the gas-turbine and steam-cycle designs, respectively. These coolant temperature differences would result in an increase in peak fuel temperature of about 140°C for a common fuel-element design. A fuel-element variation being evaluated for the gas-turbine HTGR design uses a fuel-rod array of 10 rows across the element radius, the same as the Fort St. Vrain fuel element (216 fuel rods per element). In contrast, the steam-cycle HTGR large-plant design has been based on an 8-row fuel element (132 fuel rods per element). Because of the 10-row element, peak fuel temperatures in the gas-turbine HTGR are the same as those for the steam-cycle HTGR with an 8-row element. The tradeoff for using the 10-row element is represented by the fabrication cost for a larger number of fuel rods and modestly higher core pressure drop.

The basic core parameters are given in Table 6-6. The 3,000-MWt core contains 534 standard fuel columns and 91 control fuel columns with 120° symmetry. Each fuel column consists of 8 fuel elements for a total of 5,000 fuel elements in the core. The fuel element and the control element are of hexagonal prism shape and their designs are identical to the Fort St. Vrain elements. The control element contains a hole for the small control rod in addition to holes for the control rod pair and the reserve shutdown system.

The core is controlled during normal operation with small control rods (SRCs) located in each control column. The SRCs are operated in three banks, where a bank corresponds to a fuel age segment. This means that each bank is uniformly distributed throughout the core, which minimizes power perturbations due to insertion of the control rod pairs.

6.2.5 GAS-TURBINE HTGR AUXILIARY SYSTEMS

Systems directly related to the primary coolant chemistry and coolant discharge are the same for both the gas-turbine HTGR and the steam-cycle HTGR including the helium purification system, the gas waste system, etc. The following are the auxiliary systems that are unique to the reactor turbine system of the gas-turbine HTGR.

6.2.5.1 Valve Hydraulic Supply System

There are four valves arranged in a split-flow bypass configuration in the primary conversion loop (Fig. 6-10). The trim, safety, and primary bypass valves function to control core-turbine bypass flow between the core inlet and the turbine outlet in each loop; the attemperation valve controls flow between the compressor exit and the turbine exit. The trim valve makes fine adjustments of turbine speed and load and is of particular use when synchronizing with the grid. The primary bypass valve can be operated in two modes: (1) it can be modulated by the plant control system for plant load control, or (2) it can be operated as a safety bypass valve in an open/close mode by a separate actuator as part of the safety bypass valve system that is included in the plant-protection system (PPS). The safety valve, used primarily for turbine overspeed/overpressure protection, is actuated by the plant-protection system and is operated in an open/close mode. This valve cannot be used for load control. The attemperation valve is used to mix cold compressor discharge helium with cool turbine exhaust helium, thereby minimizing thermal shock to the power conversion loop components during transients, specifically the recuperator.

These four valves are supplied with hydraulic fluid from the valve hydraulic supply system (1 per valve). Each system consists of hydraulic pumps, accumulators, pressurizers, and controls. The system operates at 1,500 psi to 2,500 psi as a function of which valve is served. The accumulators allow for a safe shutdown of a turbomachine through actuation of the bypass valve system in the case of a loss of power to the hydraulic system or a failure of the hydraulic system.

6.2.5.2 Rotating Machinery Service System

a. Turbomachinery Turning Gear System

Because of the length and weight of the turbomachinery rotor, a turning gear is required. In addition, during the low-speed turning gear operation, a shaft jacking pump must be utilized to lift the rotor hydraulically to avoid bearing damage since the shaft speed is not adequate to create a hydraulic wedge in the bearings.

b. Turbomachinery Lubrication and Buffer System

The turbomachinery radial bearings and bearing housings are serviced by the lubrication and buffer system. This system provides lubrication to the No. 1 and No. 2 bearings and buffer helium to the bearing housing shaft seals.

c. Main Shaft Penetration Seal Oil System

This system provides seal oil at 1,010 psia to the multiple floating ring seal that forms the seal in the turbomachine cavity plug around the drive shaft between the turbomachinery and the generator.

d. Generator, No. 3 Load Bearing, and Thrust Bearing Lubrication System

The generator, the No. 3 load bearing, and thrust bearings are lubricated from a common oil system. The sump for this system must be located near and below the generator because a gravity oil-return system is utilized. As in the case of the turbomachinery, the generator rotor requires jacking oil pumps to lift the rotor hydraulically during low-speed operation of the generator.

e. Generator Deionized-Water System

The generator water-cooling system provides separate water-cooling systems to the rotor, stator, and air gap cooling passages.

6.2.6 PLANT-PROTECTION, CONTROL, AND DATA-ACQUISITION SYSTEMS

6.2.6.1 Plant-Protection System

The plant-protection system includes all of the equipment from and including the sensors to the input terminals of the actuated devices that are involved in providing actions that lead to a function that provides protection to the public.

The plant-protection system prevents any unacceptable releases of radioactivity that could constitute a hazard to the health and safety of the public by initiating actions to protect the fission-product barriers and to limit the release of radioactivity if failures occur in the barriers. To accomplish these functions, the PPS systems provide the following:

1. Initiation of rapid reduction in power level following reactivity excursions, loss of adequate core cooling, or other events in order to minimize the damage to fuel coating and preserve the integrity of the primary coolant system boundary (PCSB) (reactor trip system)
2. Limit the quantity of water that can leak into the PCRV following failures in the precooler in order to minimize damage to the fuel and protect the integrity of the PCRV (precooler isolation and dump system)[a]
3. Prevent any damage to the PCSB that might result from turbomachine failure at excessive speeds (main loop shutdown system)
4. Initiate auxiliary core cooling following the loss of effective main loop cooling in order to preserve the integrity of or minimize the damage to the fuel coating and/or the PCSB (core auxiliary cooling system (CACS) initiation system)

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[a]In the event of a leak in a precooler, the plant-protection system and detection instrumentation protect against the release of primary coolant by isolating the precooler and dumping one-half of its water inventory to a surge tank.

5. Limit the maximum PCRV internal pressure in order to preserve the integrity of the PCSB (main-loop-shutdown system, reactor trip system, and safety bypass valve system)
6. Prevent simultaneous withdrawal of more than one control rod pair in order to restrict the possible reactivity excursions that can be initiated by control rod withdrawals (single control rod withdrawal interlock)

Table 6-7 presents a summary of the PPS protective functions. The table describes each protective function, the signals that initiate each function, the purpose of each function, and remarks concerning the system actions and/or interfaces involved in these protective functions.

6.2.6.2 Plant Control System (PCS)

Figure 6-11 illustrates the plant model used in control studies and analyses. Turbine speed and electrical load are regulated by a bypass valve in each power conversion loop, which bypasses helium from the reactor inlet to turbine discharge. Load is controlled by this regulation in combination with automatic reactor outlet temperature control and manually initiated helium inventory control. Load control by helium inventory or reactor outlet helium temperature control offers improved part-load efficiency relative to the use of bypass valve control. Reactor outlet helium temperature is regulated by the adjustment of control rods to regulate reactor power. The optimum combination of these modes of control will be determined as the plant design is developed.

a. Plant Control System Description

The PCS is designed to regulate reactor power and to control electrical load, turbine speed, temperature of the helium delivered to the turbine, and thermal transients experienced by the PCL and reactor components.

The PCS gives the plant the capability of continuous operation under fully automatic control at any point between 100% and 25% rated load. In addition, the PCS provides automatic load-following control capabilities for the various rates of electrical load changes.

To perform the PCS functions, several plant variables require manipulation by closed-loop controllers. These are:

1. Turbine-inlet temperature
2. Electrical power and turbomachine--generator shaft speed
3. High-pressure recuperator exit temperature[a] and low-pressure recuperator inlet temperature[a]
4. Compressor surge margin

Figures 6-11 and 6-12 show the location of each manipulated variable and the load-following part of the PCS, respectively.

The control system operates the reactor control rods, producing reactivity changes to control reactor power and turbine-inlet temperature. In addition, the

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[a]Active during bypass valve operation and for component protective action.

control system operates the trim and primary bypass valves in each loop to control turbomachine shaft speed variations in response to electrical load fluctuations, and the attemperation valve in each loop to control thermal transients in PCL components. These valves are controlled independently in the three loops.

Desired electrical power (E_d) is the primary input to the control system; from this demand, the scheduled turbine-inlet temperature is computed. Both of these quantities are then used to compute reactor power and control bypass valve system (CBVS) setpoints. Below full power, turbine-inlet temperature is nominally scheduled in a manner that will allow a minimum of 10% of full electrical load to be picked up by actuating the CBVS. Reactor outlet temperature or helium inventory control may be used to maintain high plant efficiency below full power conditions.

The reactor neutron flux (F) and the valve setpoints have been obtained for steady-state conditions over the full operating range of the plant. The inclusion of these setpoints as feed-forward signals provides anticipatory control and, therefore, rapid response to changes in load demand. The reuglation of the closed-loop temperature and load/speed controllers is limited in such a manner that no major system transient can be caused by a failure of one of these controllers.

Helium inventory change to increase part-load efficiency is currently designated as a manual operation. The automatic controls remain compatible with this manual option.

There are three automatic control loops and two supplementary control functions which are described in more detail below.

b. Turbine Inlet Temperature Control

The average inlet temperature of the turbines of the operating PCLs is controlled throughout the normal load range by adjustment of reactor power via the turbine-inlet temperature controller, which provides a command signal to the reactor neutron flux controller. The neutron-flux controller adjusts the position of the control rods to vary reactor power and, thus, the heat transferred to the helium.

The temperature control loop consists of a proportional-plus-integral-plus-deviation controller with limited output. The limits have been chosen to prevent control-system-induced power transients from causing any unintentional reactor trip.

The flux controller provides commands to the rod control system to regulate control rod position. The controller maintains the neutron flux as measured by an average of up to six out-of-core neutron detectors to within a prescribed tolerance about either a locally adjusted setpoint or a remotely controlled setpoint provided by the turbine-inlet temperature controller. The flux controller consists of an on-off type of element with hysteresis.

In addition, the neutron-flux controller issues a runback signal to the rod control system to provide automatic shim action on several rods whenever a large load reduction occurs. The PCS initiates rod insertion whenever the reactor average flux exceeds the setpoint by more than 10%. The runback control output will not reset until the deviation is reduced to 4%.

c. Electrical Power and Turbine Speed Control

The control system uses the primary bypass valves to provide the coarse control necessary to establish an operating point for large load changes. The trim valves are used to provide fine control for load and speed regulation about the established operating point. Actuation of either the primary bypass valves or the trim valves causes partial diversion of helium from the core inlet plenum to the low-pressure recuperator inlet, thus reducing turbine drive by reducing the turbine pressure ratio, and consequently, the turbine flow. The turbine-inlet temperature control subsequently operates to adjust the bypass control to its maximum level at reduced loads.

Gains and limiters in the controller are set to limit excursions about the setpoints to values compatible with 10% step load changes.

d. Attemperation Control

Thermal transients experienced by the PCL and reactor components are controlled in each loop throughout the normal load range by the attemperation controller. The controller manipulates the high-pressure recuperator exit and low-pressure recuperator-inlet temperature to a demanded value that is a programmed function of average turbine-inlet temperature. Control is accomplished by actuation of the attemperation valve, diverting helium flow from the compressor exit to the turbine exit. The controller forces the sum of the two measured temperatures to a demanded value.

The temperature demand signal is designed to hold the attemperation valve closed under normal operating conditions. The command signal is nominally rate-limited to $1^{\circ}F/sec$ to control the rate of change of temperature that components experience. The remainder of the loop consists of a proportional-plus-integral controller with limiters to prevent integrator saturation.

e. No-Load Turbomachine Speed Control

Direct control of turbomachine speed in each loop is required for plant startup or shutdown, controller manual or automatic shutdown, synchronization, and overspeed protection. In these instances, turbomachine speed is controlled by the no-load speed controller. The controller commands actuation of the primary bypass and trim valves to maintain speed at a demanded value. This demanded value may be a fixed setpoint, as in the event of loss of load with return to idle, or it may be a programmed ramp profile for purposes such as plant startup.

f. Surge Margin Control

The surge margin controller in each loop prevents reduction of compressor surge margin below a setpoint. Control is accomplished by actuation of the attemperation valve. Opening of this valve increases the compressor surge margin. The measurement of surge margin for the control is not feasible in terms of measuring and processing "real" parameters. Direct measurement of the compressor-inlet pressure and pressure rise, however, can be translated into pressure ratio and related to surge margin for control purposes.

g. Component Operational Protection

In addition to accommodating the plant system perturbations that result from normal load changes, the PCS acts to provide component operational protection by detecting out-of-bound parameters and initiating actions to limit conditions imposed on the system during loop trip or electrical load rejection. Under these conditions, reactor power and helium flow are regulated to minimize any temperature transients imposed on the PCL and reactor components.

On detection of conditions that could lead to a requirement for an overspeed protection or a main loop trip, the PCS initiates a reconfiguration of the control mode and a modification of control system demand levels; thus, the no-load speed controller prevents the turbomachine rotational speed from increasing to a point that would result in actuation of the PPS overspeed protection function.

The PCS will assist in any PPS-initiated actions to minimize system requirements. Proper PPS operation, however, does not depend on any part of the PCS.

h. Startup and Shutdown Operation

The PCS provides proper management of the systems required for normal plant and loop startup and shutdown. Manually initiated and automatically sequenced commands are issued to the turbine speed, turbine-inlet temperature, and attemperation controllers to perform the startup and shutdown functions.

Startup involves motoring of the generator through a static frequency converter from zero speed up to a speed (approximately 950 to 1,000 rpm) where the turbomachine is self-sustaining. Motoring of generator through the static frequency converter (SFC) may also be used to extend normal main loop cooling beyond the point where the afterheat generation and temperature of the core have become insufficient for self-sustained operation. The operation of the SFC is limited, however, to low-speed operation at reduced helium inventory based on the limited power capability of the station frequency converter.

6.2.6.3 Plant Data Acquisition, Processing, and Display System

The data acquisition, processing, and display system is a dual-computer-based interface between the plant instrumentation and the plant operator. Redundancy of computers and critical peripheral equipment is used for maximum availability.

This system converts certain instrument signals to engineering units, tests for alarm conditions, and provides visual and audible alarms, periodic logs, point trending, sequence-of-event recording, post-trip review, and displays of various operator information and procedural instructions on multicolor cathode ray tubes. Various applications programs are executed in the system computers to provide operational or plant-performance information. Categories of these applications programs are:

Core-reactivity status
Core temperature and power distribution
Heat balance
On-line control rod calibration
Plant-performance calculations
Operator guides
Condition monitoring of all PCL components

Table 6-1. Main features of the closed-cycle gas-turbine HTGR plant[a]

Power-plant life, year	40
Plant availability, %	80
Core thermal rating, MWt	3,000
Efficiency with dry cooling, %	40
Efficiency with ammonia bottoming cycle, %	48

[a]Reference design based on--

a. Integrated direct cycle plant
b. Prismatic core, thermal rating, 3,000 Mwt
c. MEU fuel
d. Reactor core power density, 6.8 W/cc
e. Nonintercooled cycle with high degree of recuperation
   P_{max} = 1,150 psia
   T_{max} = 1,562°F (850°C)
   R_{comp} = 2.5
   R_{recup} = 0.90
f. Turbomachine rating, 400 MWe
g. Water-cooled and insulated liners throughout
h. PCRV central core cavity: diameter, 129 ft; height, 116 ft
i. Delta turbomachine position
j. CACS--3 x 100% units
k. Two-bearing turbomachine (single turbine inlet duct)
l. Man-access provision to bearing cavity areas
m. Straight tube, modular recuperator
n. Helical bundle precooler
o. Dry-cooled plant
p. Cycle adaptable to waste heat rankine bottoming plant
q. Emphasis placed on gas flow path simplicity and minimization of primary system pressure loss
r. Parameters and plant layout based on optimization study
s. State-of-the-art technology

Table 6-2. Details of 400-MWe (60-Hz) single-shaft helium-gas turbine

Parameter	Compressor	Turbine
Number of stages	18	8
Hub diameter, in. (mm)		
First stage	62.0 (1,575)	66.6 (1,691)
Last stage	62.0 (1,575)	62.6 (1,590)
Tip diameter, in. (mm)		
First stage	71.9 (1,826)	76.5 (1,943)
Last stage	68.3 (1,735)	86.0 (2,184)
Hub-to-tip ratio, first/last stage	0.86/0.91	0.87/0.73
Blade height, in. (mm)		
First stage	4.95 (126)	4.95 (126)
Last stage	3.15 (80)	11.7 (297)
Blading adiabatic efficiency, %	89.8	91.8
Overall machine length, ft (m)	37 (11.3)	
Machine outer diameter, ft (m)	11.5 (3.5)	
Rotor weight, tons (kg)	67 (60,800)	
Stator and case weight, tons (kg)	238 (216,000)	
Total machine weight, tons (kg)	305 (276,800)	
Speed of rotation, rpm	3,600	
Type of rotor construction	Welded	
Turbine blade material	Nickel-base alloy (IN 100)	
Rotor burst shield	Integral part of machine structure	
Journal bearing man-access	For inspection and limited maintenance	
Bearing details		
Number of journal bearings	2	
Type of journal bearings	5 pad, tilting pad, oil lubricated	
Thrust-bearing type	8 pad, tilting pad, double acting	
Thrust-bearing location	External to PCRV	

Table 6-3. Heat exchanger details for the gas-turbine HTGR (400-MWe loop rating)

Heat Exchanger	Recuperator	Precooler
Plant loop rating, MWt	1000	
Thermodynamic cycle	Nonintercooled	
Matrix type	Plain tubular	Externally finned tubes
Flow configuration	Axial counterflow	Multipass cross counterflow
Construction	Modular	Helical bundle
Heat transfer rate, MWt	918	581
LMTD, °F (°C)	76.5 (42.5)	54.9 (30.5)
Effectiveness	0.898	0.972
Water outlet temperature, °F (°C)	--	270 (132)
Helium $\Delta P/P$, %	2.82	0.75
Tube outer diameter, in. (mm)	0.4375 (11.1)	1.125 (28.6)
Tube wall thickness, in. (mm)	0.045 (1.14)	0.113 (2.87)
Maximum metal temperature, °F (°C)	960 (516)	351 (177)
Pressure differential, psi (bar)	656 (45.2)	265 (18.3)
Material type	Ferritic, 2.25Cr-1Mo	Low-alloy steel (0.5 Cr)
Modules/exchanger	83	1
Tube/module	804	832
Effective tube length, ft (m)	40 (12.2)	41 (12.5)
Surface area exchanger, ft^2 (m^2)	305,730 (28,400)	238,000 (22,110)
Cavity diameter, ft (m)	19.5 (5.95)	16.5 (5.03)
Thermal power density, MWt/m^3	5.4	3.3
Heat flux, W/cm^2	3.2	2.6
Overall length, ft (m)	67 (20.4)	65 (19.8)
Assembly diameter, ft (m)	18.5 (5.63)	15 (4.6)
Approximate weight, kg (tons)	726,000 (800)	435,600 (480)
ISI repair level	Module	Individual tubes
Assembly location	Shop	
Shipping mode	Barge	
ASME code class	Section VIII	

Table 6-4. Major performance parameters for a 3,000-MWt dry-cooled gas-turbine HTGR plant

Parameter	Pressure (psia)	Temperature (°F)	Flow/loop (10^6 lb/hr)
Reactor inlet	1,128.8	926.9	4.312
Reactor outlet	1,120.7	1,562.1	4.312
Duct inlet	1,120.7	1,562.1	4.312
Duct outlet	1,115.2	1,562.0	4.312
Turbine inlet	1,115.2	1,560.2	4.319
Turbine outlet	476.6	995.9	4.482
Duct inlet	476.6	993.2	4.504
Duct outlet	473.7	993.2	4.504
Recuperator hot inlet	473.7	993.2	4.481
Recuperator hot outlet	467.5	433.8	4.481
Duct inlet	467.5	433.6	4.504
Duct outlet	466.7	433.6	4.504
Precooler inlet	466.7	433.3	4.493
Precooler outlet	461.8	79.0	4.493
Duct inlet	461.8	79.0	4.504
Duct outlet	460.0	79.0	4.504
Compressor inlet	460.0	80.1	4.522
Compressor outlet	1,150.0	346.2	4.522
Duct inlet	1,150.0	346.2	4.317
Duct outlet	1,141.8	346.1	4.317
Recuperator cold inlet	1,141.8	346.1	4.317
Recuperator cold outlet	1,132.1	927.0	4.317
Duct inlet	1,132.1	927.0	4.317
Duct outlet	1,128.8	927.0	4.317
Overall plant eficiency	39.7%		
Net plant electrical power	1,191 MWe		

Table 6-5. Performance parameters for a 3,000-MWt binary-cycle gas-turbine HTGR

Loop component	Pressure (psia)	Temperature (°F)	Flow per loop (lb/hr)
Reactor inlet	1,124	966	13,783,000
Reactor outlet	1,115	1,562	13,783,000
Duct inlet	1,115	1,562	4,594,000
Duct outlet	1,108	1,562	4,594,000
Turbine inlet	1,108	1,560	4,601,000
Turbine outlet	480	1,025	4,647,000
Duct inlet	480	1,023	4,671,000
Duct outlet	477	1,022	4,671,000
Recuperator hot inlet	477	1,022	4,647,000
Recuperator hot outlet	470	519	4,647,000
Duct inlet	470	518	4,671,000
Duct outlet	469	518	4,671,000
Precooler inlet	469	518	4,659,000
Precooler outlet	462	153	4,659,000
Duct inlet	462	153	4,671,000
Duct outlet	460	153	4,671,000
Compressor inlet	460	154	4,690,000
Compressor outlet	1,150	456	4,690,000
Duct inlet	1,150	456	4,599,000
Duct outlet	1,139	456	4,599,000
Recuperator cold inlet	1,139	456	4,599,000
Recuperator cold outlet	1,128	966	4,599,000
Duct inlet	1,128	966	4,599,000
Duct outlet	1,124	966	4,599,000

Combined Plant

Primary plant output	1,081 MW
Secondary plant output	378 MW
Auxiliary power:	
Primary plant	11.0 MW
Secondary plant	11.9 MW
Net output	1,436 MW
Plant efficiency	47.88

Table 6-6. Basic core parameters for 3,000-MWt gas-turbine HTGR

Parameter	Value
Thermal power, MWt	3,000
Power density, kW/l	6.8
Number of axial zones	4
Number of fuel elements	5,000
Number of fuel elements/column	8
Number of fuel columns	
Standard	534
Control	90
Core height, m	6.3
Effective core diameter, m	8.5

Table 6-7. Summary of protective functions of the plant-protective system

Protective Function	Initiating Condition	Purpose	Remarks	System Action/ Interfaces
Reactor trip	High reactor-power-to-flow ratio at high flow	Prevent damage to core and PCRV internals following a power excursion or loss of flow	Primary (primary means designated to provide principal protection against a condition)	1. Drop all control rods 2. Initiate PCS load reduction (not required for safety) 3. Initiate main loop shutdown on "high primary coolant pressure" only (internal pressure relief)
Reactor trip	High reactor trip at low flow	Prevent damage to core and PCRV internals following a power excursion	Primary	
Reactor trip	High reactor flux during low power testing	Prevent damage to core and PCRV internals following a power excursion	Primary	
Reactor trip	High helium temperature at the turbine inlet	Maintain integrity of primary coolant pressure boundary and prevent damage to core in the event of power-to-flow mismatches following a power excursion or loss of flow	Diverse backup for reactor trip on high reactor-power-to-helium-flow ratio	
Reactor trip	High primary coolant pressure	Limit primary coolant pressure	Primary	
Reactor trip	High containment radiation level	Prevent damage to core and PCRV internals following reactor depressurization into the containment	Primary	

Table 6-7. Summary of protective functions of the plant-protective system (continued)

Protective Function	Initiating Condition	Purpose	Remarks	System Action/ Interfaces
Reactor trip	High containment pressure	Prevent damage to core and PCRV internals following a reactor depressurization into the containment	Diverse backup to high containment radiation level	
Reactor trip	Loss of preferred bus voltage	Prevent damage to core and PCRV internals following loss of preferred power	Primary	
Reactor trip	Two or more main loop shutdown signals	Prevent damage in upper plenum and prevent reaching the high reactor power-to-helium-flow limit	Single loop shutdown cannot result in excessive temperature in the upper plenum	
Reactor trip	Manual reactor trip	Allow reactor trip at operator's discretion		
CACS initiation	Low plant helium flow	Prevent damage to core and PCRV internals following loss of primary coolant flow	Primary	1. Initiate main loop shutdown 2. Commence startup of all CACS loops
CACS initiation	Manual CACS initiation	Allow CACS initiation at operator's discretion		
Main loop shutdown	High PCL exit temperature	Prevent damage to upper plenum thermal barrier	Primary	1. Trip the SBVS 2. Initiate nonsafety reactor power setback (not required for safety)
Main loop shutdown	High turbomachine speed	Limit peak turbomachine speed to within turbomachinery design limits	Primary	1. Trip the SBVS 2. Initiate nonsafety reactor power setback (not required for safety)

Table 6-7. Summary of protective functions of the plant-protective system (continued)

Protective Function	Initiating Condition	Purpose	Remarks	System Action/ Interfaces
Main loop shutdown	CACS initiation	To allow proper functioning of CACS		
Main loop shutdown	Precooler isolation and dump	Prevent damage to PCL components	Followup action[a]	
Main loop shutdown	Detection of PCS main loop trip failure	Discretionary loop shutdown	Followup action[a]	
Main loop shutdown	Manual loop shutdown	To allow shutdown at operator's discretion		
Main loop shutdown--all loops	Isolation and dump of both halves of any precooler	Prevent dryout of precooler and consequent overtemperature of thermal barrier/liner and damage to turbomachinery		
Precooler isolation and dump	High activity in pre-cooler water outlet line	Limit fission-product release following a failure in a precooler	Primary	1. Close isolation valves (one-half precooler) 2. Open dump valves (one-half precooler) 3. Initiate main loop trip
Precooler isolation and dump	Manual precooler isolation dump	Allow precooler isolation and dump at operator's discretion		
Single control rod pair withdrawal interlock	Detection of outward command to two or more control rod pairs	Prevent simultaneous withdrawal of two or more control rod pairs		Block motor controller output to rod drive motors

[a]Action designated to minimize transient effects on plant systems/hardware.

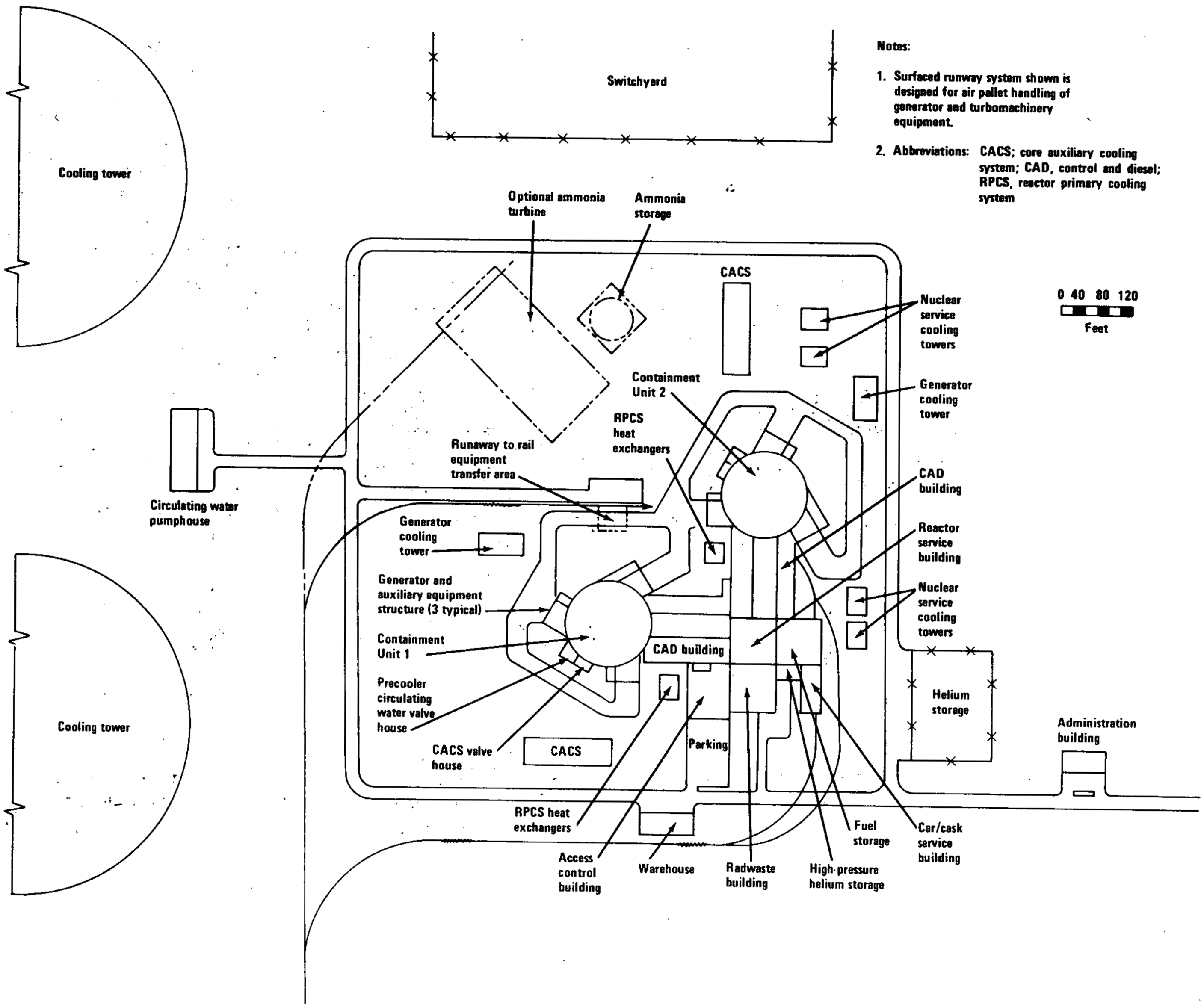


Figure 6–1. Plot plan for a gas-turbine HTGR plant with twin 3,000-MWt reactors.

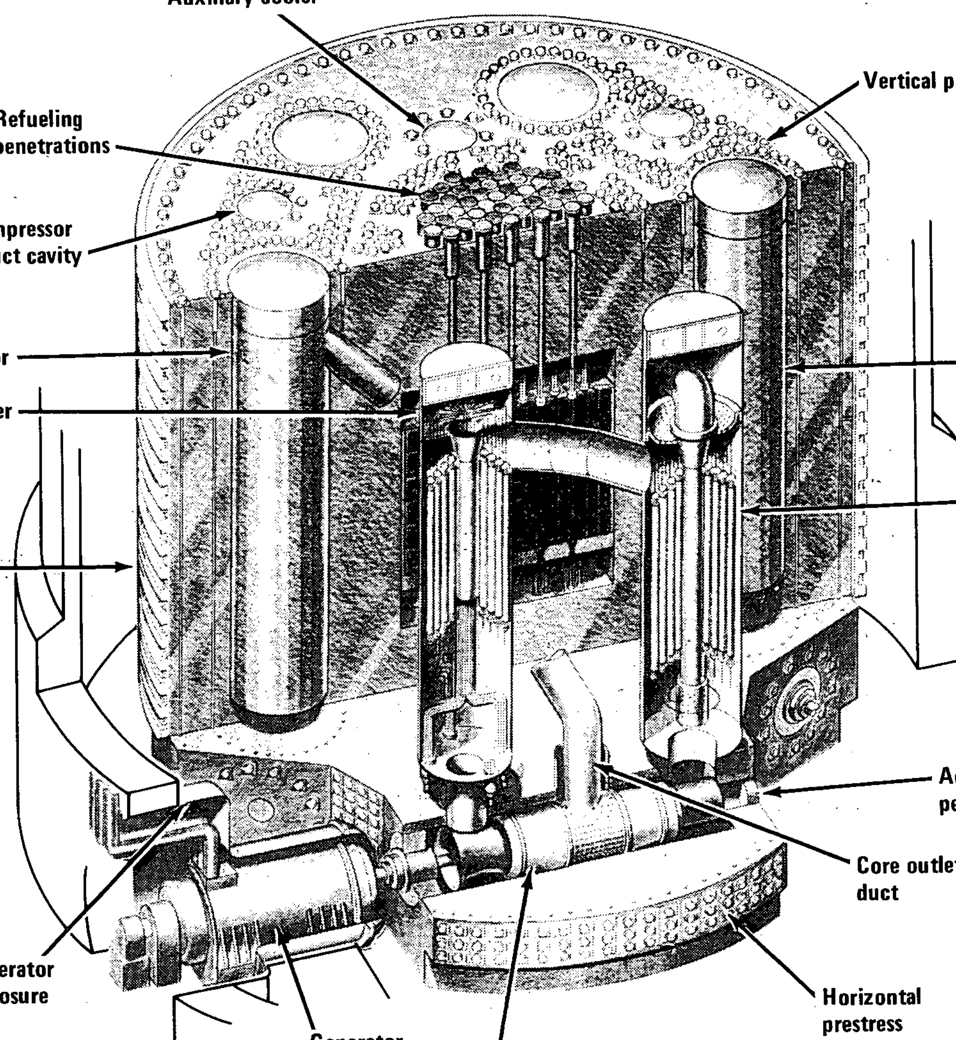


Figure 6-2. Integrated gas-turbine HTGR with 3,000-MWt reactor core and three power-conversion loops.

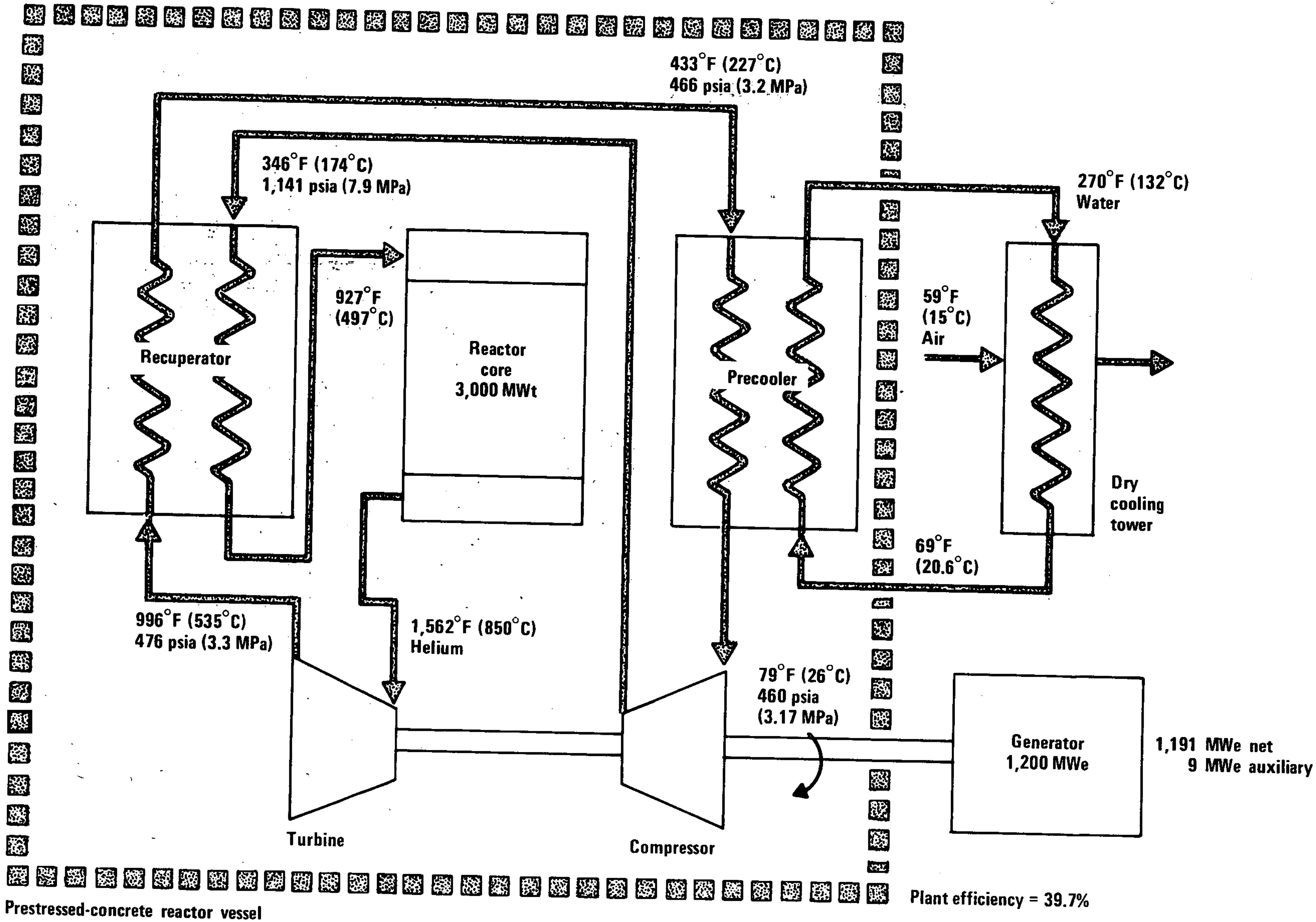


Figure 6-3. Cycle diagram of 3,000-MWt gas-turbine HTGR with dry cooling (ISO rating conditions).

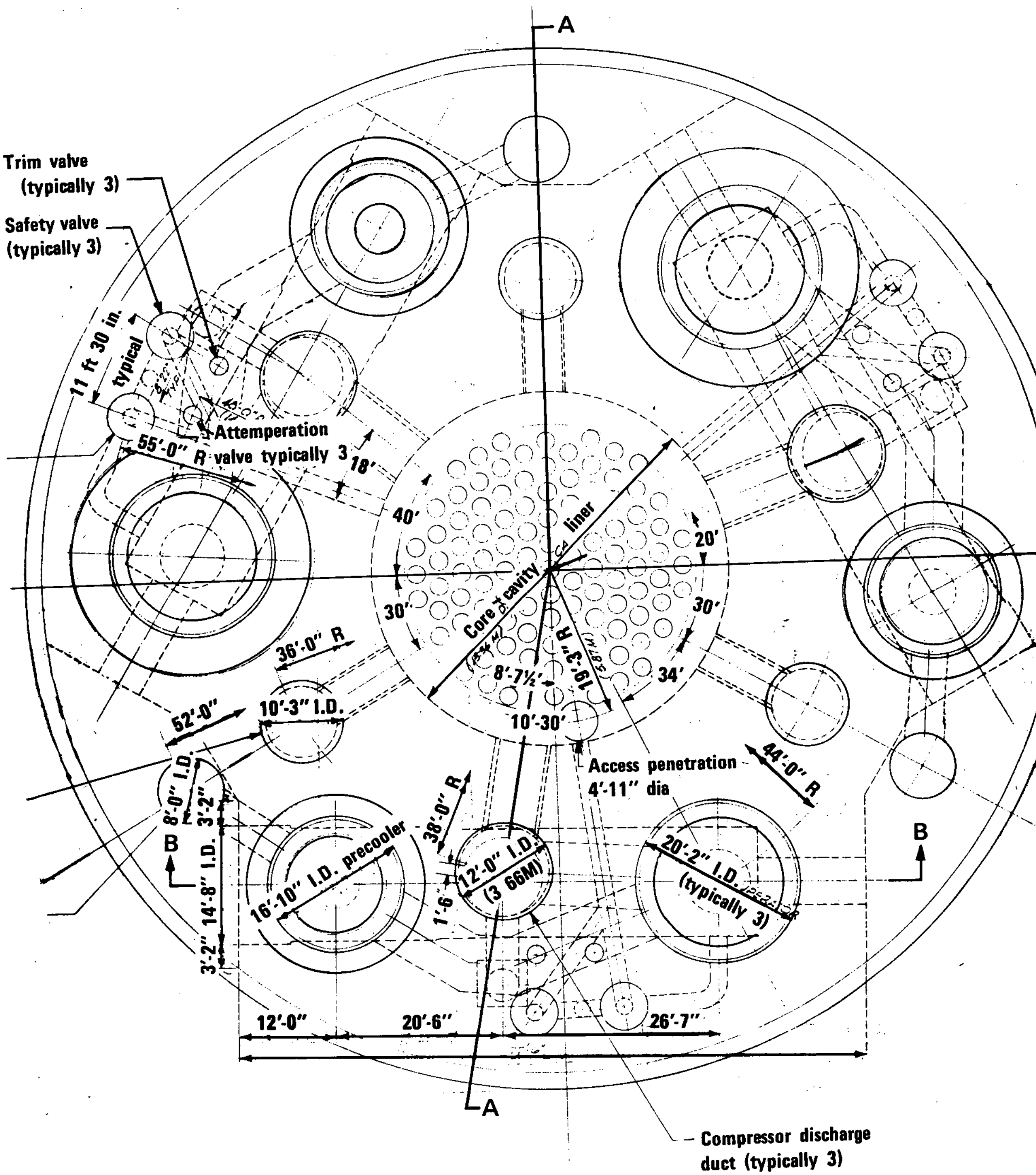


Figure 6-4. Plan view of the prestressed-concrete reactor vessel.

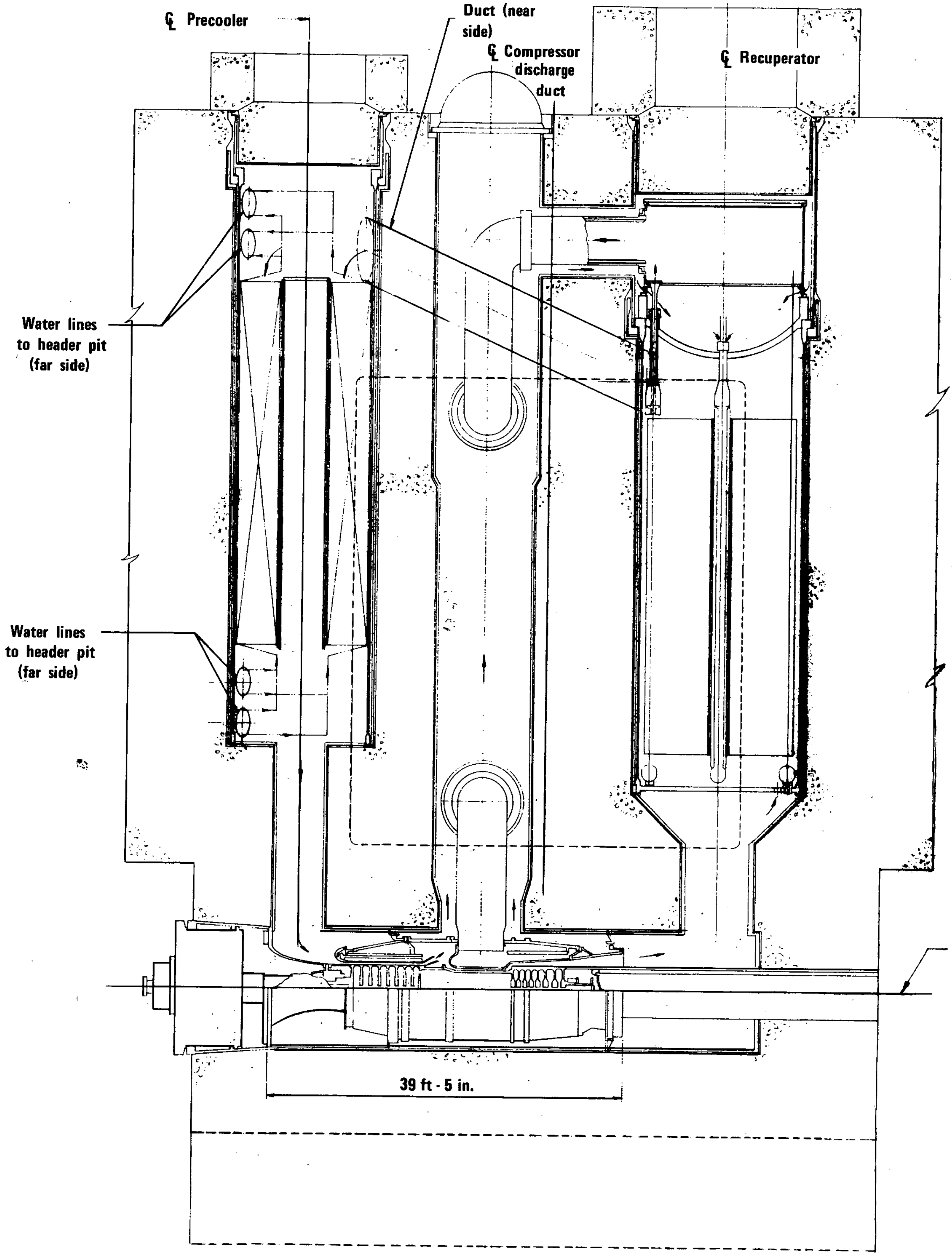


Figure 6-5. Prestressed-concrete reactor vessel, three-loop gas-turbine HTGR, elevation view (section B-B of Figure 6-4).

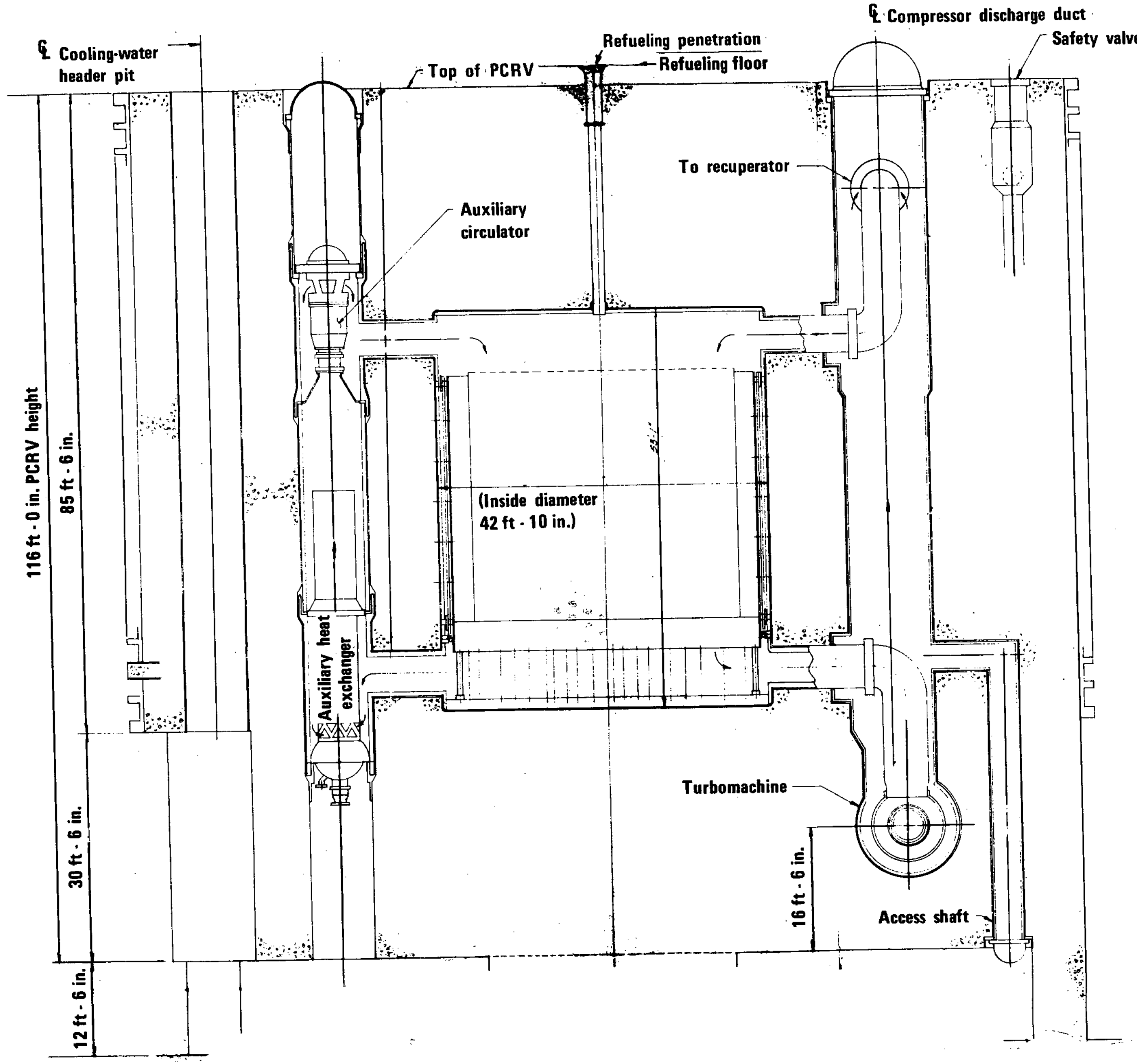


Figure 6-6. Prestressed-concrete reactor vessel, three-loop gas-turbine HTGR, elevation view (section A-A of Figure 6-4).

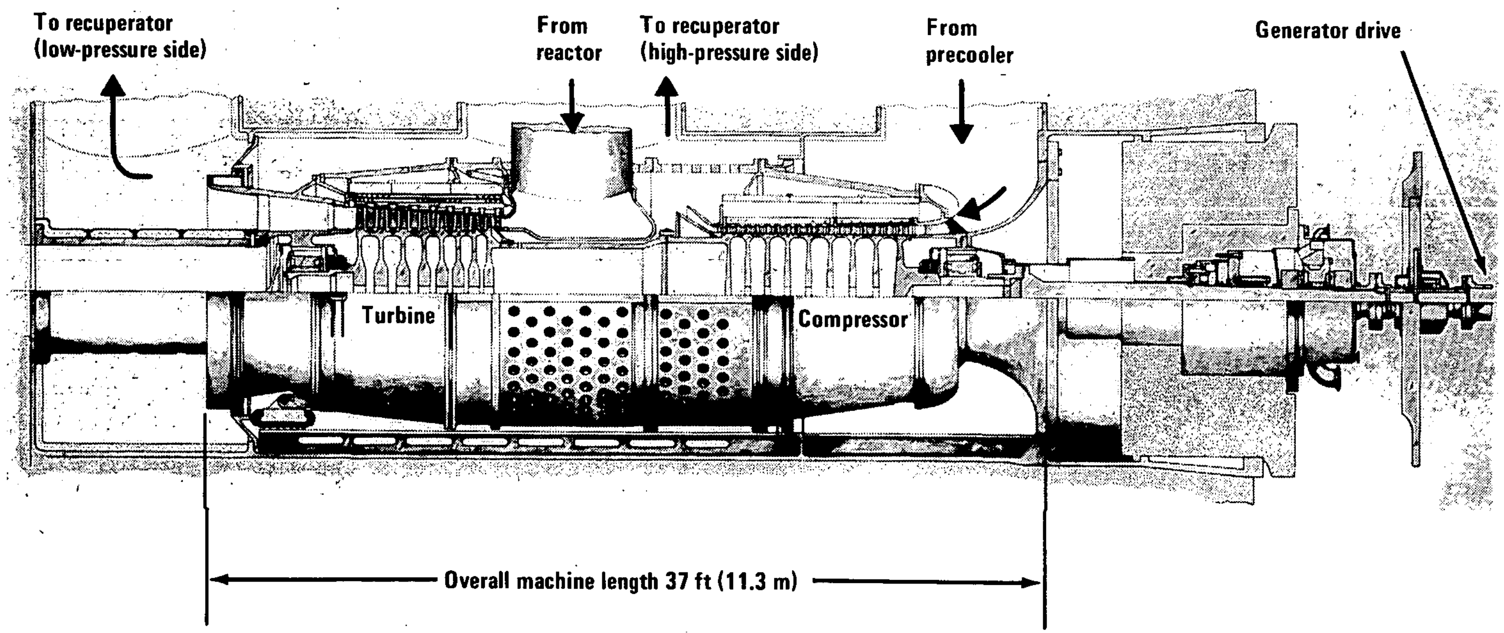


Figure 6-7. Diagram of a 400-MWe single-shaft helium turbomachine for a gas-turbine HTGR plant.

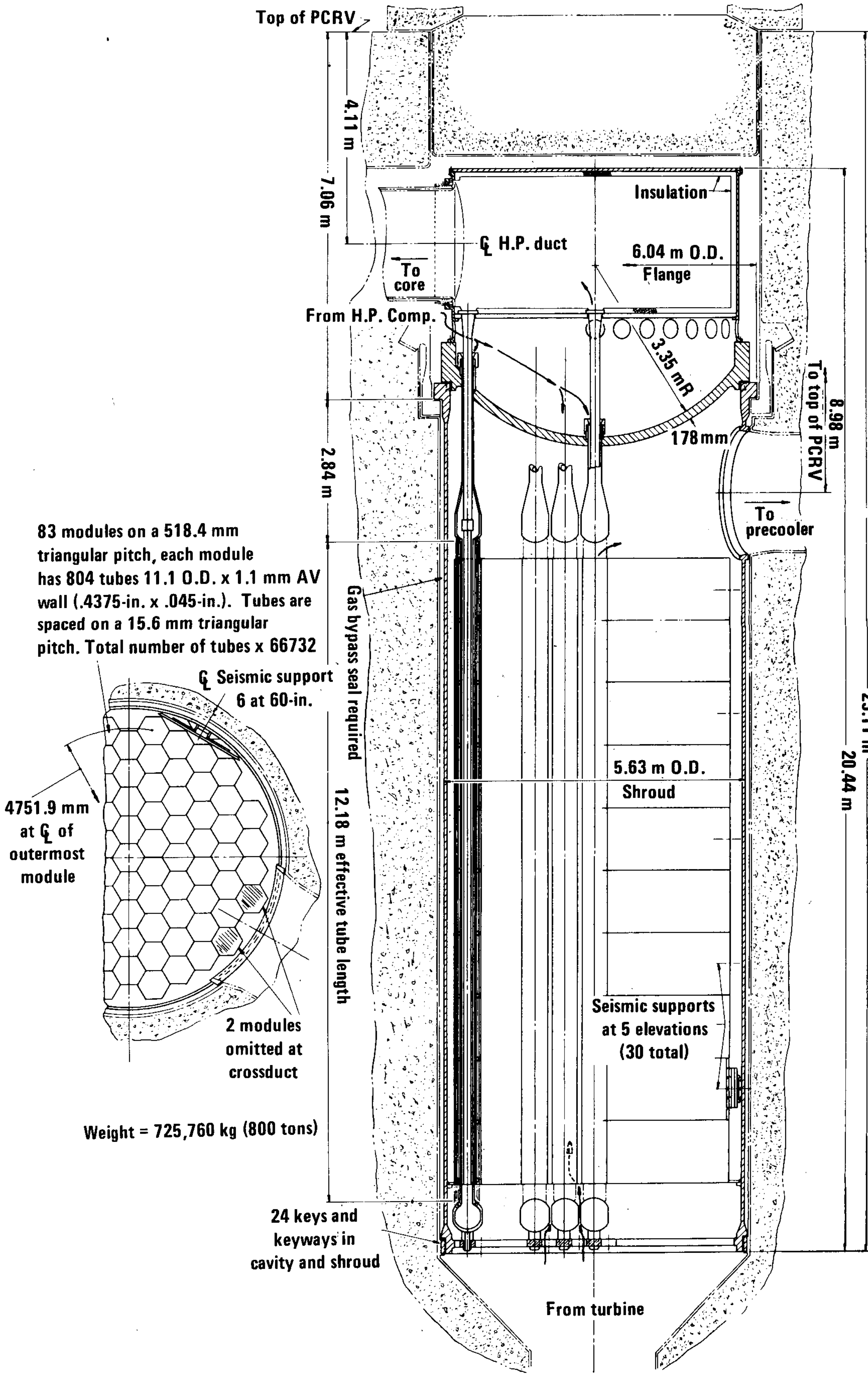


Figure 6-8. Gas-turbine HTGR recuperator configuration.

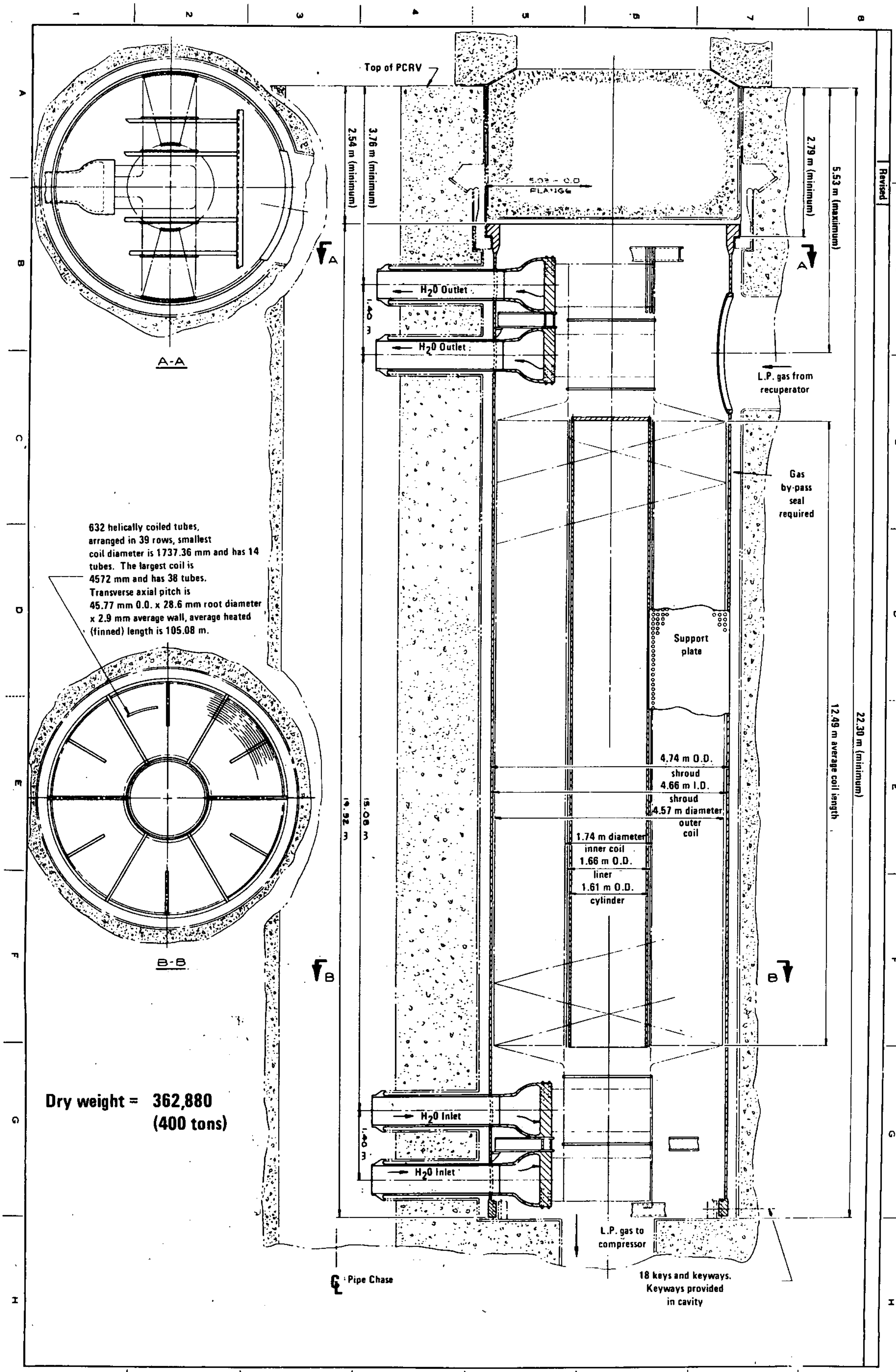


Figure 6-9. Gas-turbine HTGR precooler configuration.

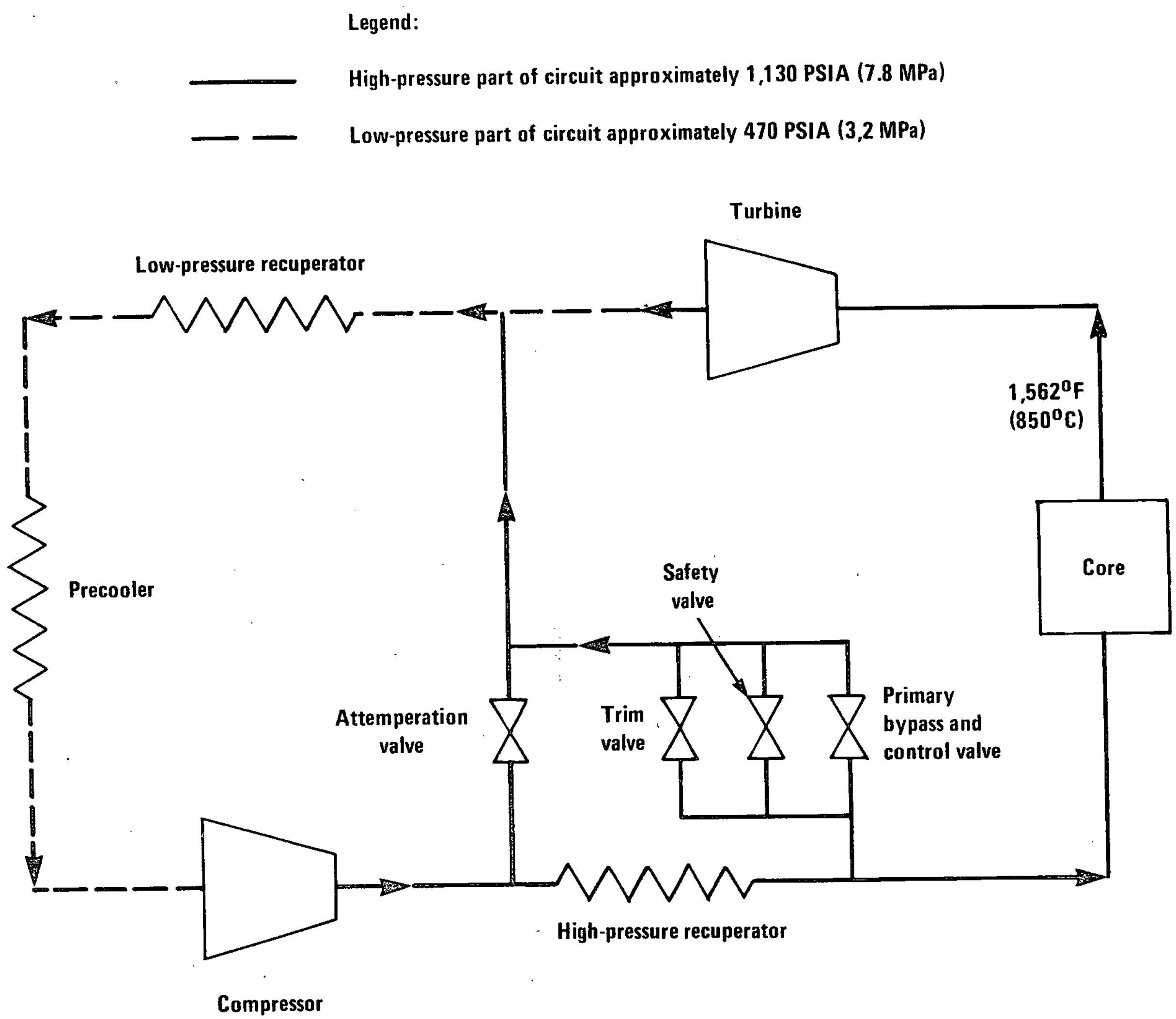
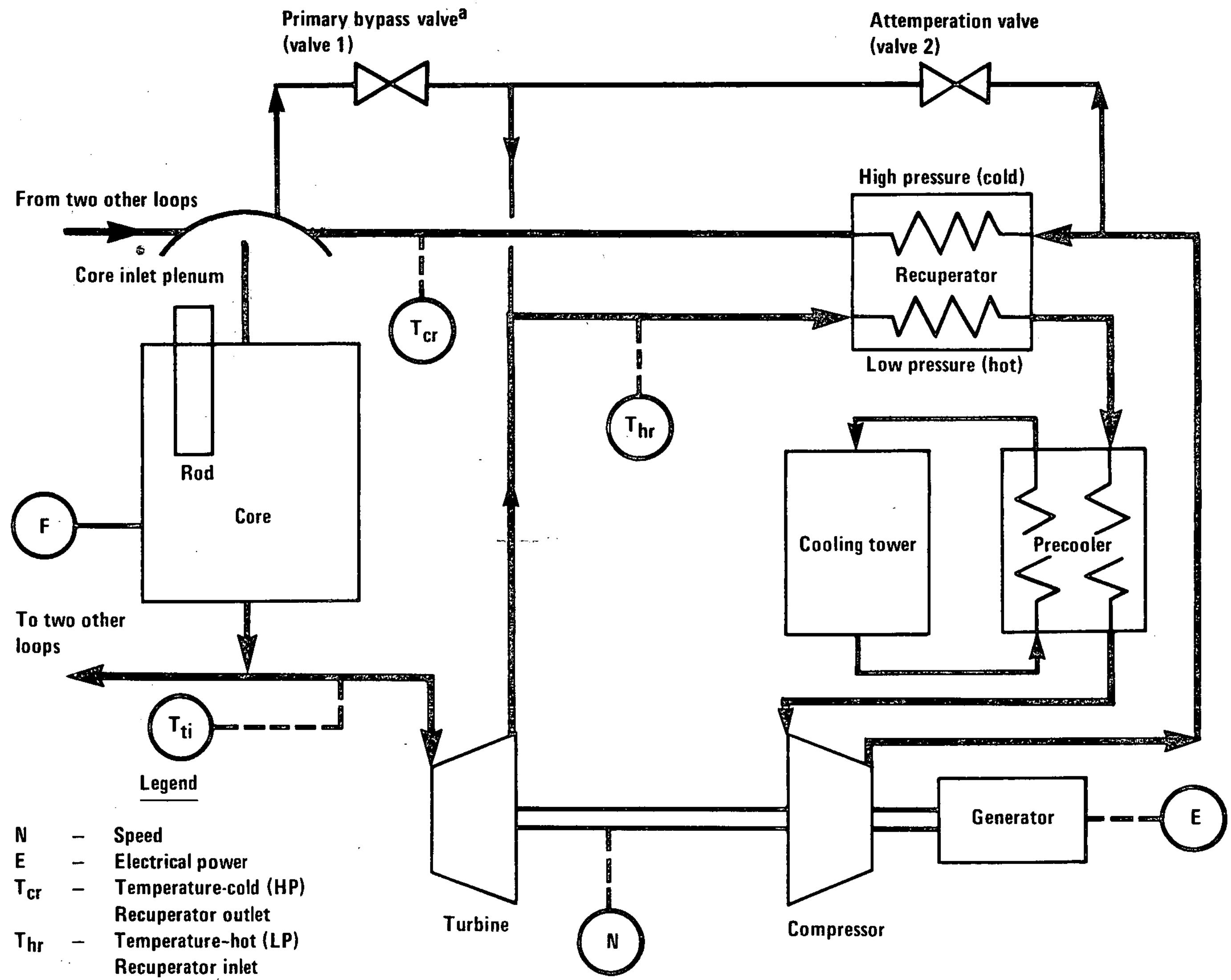


Figure 6-10. Simplified control valve diagram for gas-turbine HTGR power plant.



Legend

N – Speed
E – Electrical power
T_{cr} – Temperature-cold (HP) Recuperator outlet
T_{hr} – Temperature-hot (LP) Recuperator inlet
T_{ti} – Temperature-turbine inlet
F – Neutron flux

[a] The safety valve and the trim valve are at this schematic position also.

Figure 6-11. Plant schematic showing location of major plant parameters.

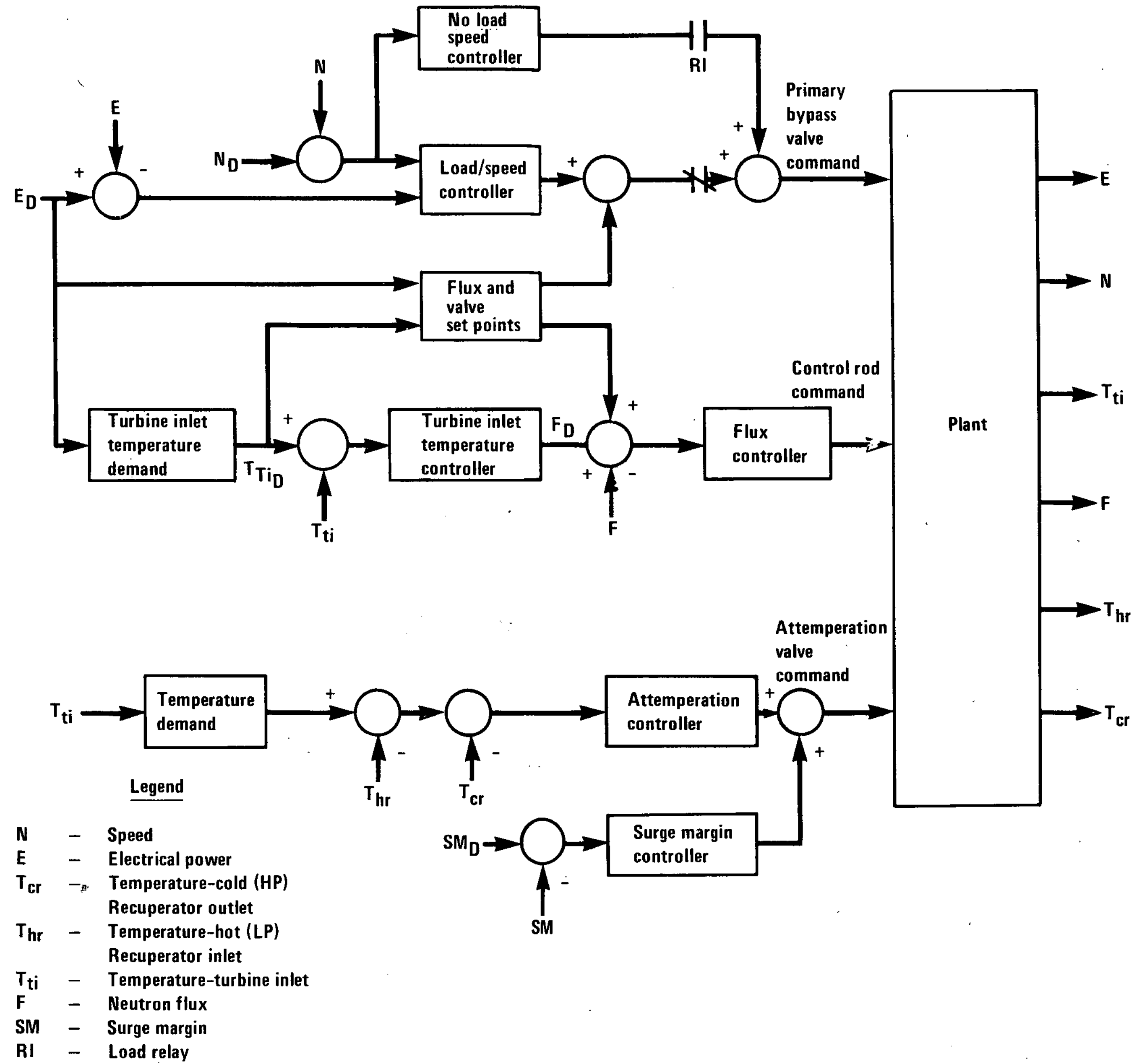


Figure 6-12. Plant control system.

6.3 SAFETY CONSIDERATIONS

The gas-turbine HTGR has a number of important features in common with the steam-cycle HTGR. The most important of these are the use of the PCRV, the prismatic graphite core with encapsulated fuel particles, and the use of three independent auxiliary cooling loops. Safety features such as the control-rods, reserve-shutdown system, and the liner-cooling system are essentially identical. In addition, the three major inherent safety features of the steam-cycle HTGR are inherent in the gas-turbine HTGR:

1. The large mass of graphite in the fuel and reflector blocks gives the core a very high heat capacity. This feature protects against rapid changes in core temperature and is highly beneficial in limiting the consequences of design-basis accidents.
2. The helium coolant does not cause reactivity changes as its density varies.
3. The enclosure of the entire reactor-coolant system within a high-integrity PCRV minimizes the possibility of a rupture in the coolant boundary.

The control and protection systems in a gas-turbine HTGR are significantly different from those of the steam-cycle HTGR, but the increased simplicity of the gas-turbine concept may lead to enhanced overall plant safety. (See Section 6.2.6.)

The systems safety philosophy on which the HTGR has been based is formulated in a way that makes it applicable to both the steam-cycle and the gas-turbine designs. Safety-related design criteria for individual components such as recuperator, precooler, turbomachines, PCRV shaft seals, and other equipment unique to the gas-turbine plant remain to be determined.

The safety classification of the heat exchangers inside the PCRV of the gas-turbine HTGR has not yet been determined. This subject is the object of the ongoing design effort, which includes a comprehensive safety evaluation of the plant.

6.3.1 STEAM-CYCLE HTGR ISSUES APPLICABLE TO THE GAS-TURBINE HTGR

In September 1978, the U.S. Nuclear Regulatory Commission (NRC) submitted to the DOE questions on eight topics related to the proposed lead-plant design for a commercial steam-cycle HTGR. These questions, and the answers by General Atomic Company, are discussed in Section 2.4.2 of Volume IV. Although the questions and answers were formulated specifically for the steam-cycle HTGR, much of the information is applicable to the gas-turbine HTGR.

This section lists the eight topics addressed in the NRC steam-cycle HTGR questions and briefly discusses the applicability of the answer to the gas-turbine HTGR.

<u>Graphite as structural material</u>. The response in Section 2.4.2 is directly applicable. The effect of the higher temperatures in the gas-turbine HTGR will have to be taken into account.

<u>Core seismic response</u>. The response in Section 2.4.2 is directly applicable.

<u>Fuel transient response</u>. Much of the response in Section 2.4.2 is applicable to gas-turbine HTGR fuel. The temperature coefficient for the gas-turbine HTGR will be stronger because of the higher average temperature of the graphite.

In-service inspection and testing. Criteria for in-service inspection of the gas-turbine HTGR will be based on the considerations used to establish the requirements for the steam-cycle HTGR. These requirements are given in the proposed Section XI, Division 2, of the ASME Boiler and Pressure Vessel Code.

Low-probability accidents. A comprehensive study of low-probability accidents for the HTGR has not yet been performed. The parts of the answer that pertain to control-rod ejection, core drop, and depressurization in the steam-cycle HTGR should be generally applicable to the gas-turbine design. The answers concerning research programs, gas-cooled-reactor experience, and nonprobabilistic criteria are applicable as well.

Containment requirements. The criteria for containment-design requirements are essentially the same for the steam-cycle and for the gas-turbine concepts. However, the differences in primary-coolant inventories and other operating characteristics must be taken into account for the gas-turbine HTGR containment design.

Primary-system integrity. Even though many of the components internal to the PCRV are not the same, the design considerations for the primary-coolant systems of both concepts are essentially the same.

Emergency core-cooling provisions. The core auxiliary cooling systems for the two concepts are essentially the same. The capacities may differ because of the different flow paths for the two concepts. The requirement for containment back-pressure has not yet been fully investigated for the gas-turbine HTGR.

6.3.2 SAFETY ASPECTS OF THE GAS-TURBINE HTGR

In addition to the HTGR generic issues discussed above, the gas-turbine HTGR has a number of features that lead to some new safety and licensing questions. The most significant of these are discussed below.

6.3.2.1 Shaft-Seal Failure

The turbomachine/generator shaft penetrates the primary-coolant-system boundary. Failure of the seal because of machine or shaft malfunction can potentially cause a rapid depressurization of the PCRV. Design features must be incorporated to ensure that such accidents have acceptably low probability of occurrence.

6.3.2.2 Internal Pressure-Equilibration Accidents

Failure of internal components such as the turbomachines or recuperators can cause rapid pressure equilibration inside the PCRV. These pressure pulses/transients are much more severe than those associated with the most rapid postulated reactor-vessel depressurization for the HTGR steam-cycle. Pressure-equilibration accidents postulated for the gas-turbine HTGR place stringent design requirements on reactor-vessel internals and dictate component designs that may be different from those of the steam-cycle plant.

In order to determine the consequences of pressure-equilibration accidents, it is necessary to define, model, and verify the failure phenomena. This in turn depends on experimental data related to failure, as well as experimental or other data that verify the modeling tools. These modeling tools will include a computer code that describes the transient behavior of the compressible-fluid flow after the accident.

Considerable effort has been expended by the General Atomic Company to develop computer programs for the analysis of the transient thermal-fluid behavior of the primary-coolant system. One such program, TUBE, was developed specifically to analyze the local consequences of rapid pressure transients. The TUBE program can represent a segment of the primary-coolant system in considerable detail, accounting for shock effects as well as bends, contractions, and expansions. Considerable insight into the local pressure history associated with this type of accident can be gained by use of the TUBE program. Eventually, the analysis of these accidents must be performed with a program that models the entire primary-coolant system. Application of the RATSAM program to the gas-turbine HTGR is being studied. The ability of RATSAM to model accidents in the gas-turbine HTGR must be validated against experimental data and/or by comparison with computer programs developed elsewhere.

6.3.2.3 Turbomachine Failures

In addition to causing rapid pressure transients, turbomachine failures can create missiles, against which protection must be provided. The steam cycle also has the potential for internal missiles generated by circulator failures, but the magnitude of the missile problem for the gas turbine is larger. Analysis of failure consequences has proceeded at General Atomic and United Technologies Corporation as part of the conceptual design of a turbine-rotor burst shield.

6.4 ENVIRONMENTAL CONSIDERATIONS

The environmental assessment of the gas-turbine HTGR concept was based on a comparison with the steam-cycle HTGR concept (Section 2.3).

6.4.1 NONRADIOLOGICAL EFFECTS

Among the nonradiological effects, the major difference between the two concepts is in station water use. The gas-turbine concept, because of the higher reject temperatures, offers the potential for using dry cooling for rejecting heat, thus making the plant site virtually independent of water supply. The average consumption of water for the steam-cycle HTGR using evaporative cooling towers is approximately 5,300 gpm for a 1,000-MWe plant. The use of dry cooling also reduces the chemical waste volume associated with evaporative cooling.

In other nonradiological effects, such as land use and heat dissipation, the two concepts are similar. Waste heat from both the steam-cycle and the gas-turbine HTGRs is lower than that from the reference LWR by about 24%.

6.4.2 RADIOLOGICAL EFFECTS

The bases for the radiological effects is the fission-product release from the core. Detailed calculations of the source term have not been performed at present. However, preliminary calculations[a] indicate that scaling factors of 1.0 and 3.0 can be applied to the steam-cycle HTGR core releases for gaseous and plateable isotopes, respectively, to estimate the source term for the gas-turbine HTGR. The reason for the higher releases in plateable isotopes is the higher fuel temperatures in the gas-turbine HTGR for a common fuel-element design.

[a]Private communication between David Hanson (General Atomic Company) and A. Papadopoulos (NUS Corporation).

With the above scaling factors, the gaseous effluents released during the normal operation of the gas-turbine HTGR plant should be the same or lower than those of a steam-cycle HTGR plant, assuming that similar gaseous-waste-processing systems are used in the two concepts. It should be noted that the gas-turbine HTGR does not have a main condenser, and this source term for gaseous effluents is eliminated.

The source for liquid effluents is the plateout radioactivity, which may find its way to the environment from component-decontamination operations. Since the plateout activity is scaled up by a factor of 3, liquid radiological effluents are expected to increase proportionally, assuming similar liquid-processing systems for the two concepts.

A unique operation for the gas-turbine HTGR is the periodic remote removal and decontamination of the turbomachinery. It is estimated that each machine will be removed for maintenance every 6 to 7 years. The effluents from the decontamination operation, possibly including parts of the turbomachinery (i.e., turbine blades), will be in the form of solid waste. This solid waste will be in addition to that specified in Section 2.3.6.4 for the steam-cycle HTGR. The radioactivity generated from each turbomachinery decontamination operation is shown in Table 6-8.

The turbomachinery maintenance operations will also increase occupational exposure over that expected from the steam-cycle HTGR. The disassembly and maintenance will be performed remotely, thus minimizing the additional exposure. It is estimated[a] that the incremental increase will be about 6 man-rem per operation per machine. The impact of this increase on the total occupational exposure (approximately 52 man-rem/yr) is not great.

[a]Private communication between David Hanson (General Atomic Company) and A. Papadopoulos (NUS Corporation).

Table 6-8. Isotopic radioactivity generated from turbomachinery decontamination

Isotope	Activity (Ci)
Silver-110m	69.5
Antimony-125	1.84
Tellurium-129m	4.22
Tellurium-129	2.62
Cesium-134	28.6
Cesium-137	18.5
Barium-137m	17.3

[a]Data from the General Atomic Company; a 100-day decay is assumed between removal and decontamination.

Note: Philosophy is to have a spare turbomachine to minimize plant downtime associated with turbomachine maintenance.

6.5 LICENSING STATUS AND CONSIDERATIONS

A Preliminary Safety Information Document (PSID) was submitted by the General Atomic Company on July 1, 1975 (Ref. 1). The NRC returned the first Request for Additional Information (RAI) on round 1 questions on December 15, 1975; the second-round RAI was returned on April 26, 1976. By mid-1976, however, funding and manpower limitations resulted in the termination of significant activity on answering the RAIs or further dialogue with the NRC.

6.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

General Atomic has the prime responsibility to ensure that the necessary research, development, and testing programs are carried out and to support the reactor turbine system design. Other organizations will carry out some parts of the total program as follows:

1. U.S. commercial organizations through direct subcontract from General Atomic
2. U.S. commercial organizations or national laboratories through direct contract from the DOE, coordinated within the national HTGR program based on data needs identified by General Atomic
3. Swiss and German organizations on cooperative studies under the Umbrella Agreement

Recommendations have been made by General Atomic to the HHT project management that a project work statement be implemented covering the planning of program test requirements. This effort would be conducted in FY-80 and would address the following:

1. Testing requirements
2. Survey of existing test facilities
3. Definition of new test facilities
4. Definition of which country would own and operate specific facilities and how they would be shared between program participants

Supporting this development effort will be the operational experience gained at Fort St. Vrain under the General Atomic-Fort St. Vrain surveillance program. The cooperative effort with the Europeans will provide operating data from existing European gas reactors.

In addition to the research and development programs listed in Section 2.5.1, the gas-turbine HTGR system will require work to verify the design, development, and performance of the turbomachinery, recuperator, precooler, shaft and penetration seals, control valves, turbomachine hot duct, the PCRV, and reactor internals. Results of a research and development program for the gas-turbine HTGR that has been in progress for several years have been presented in several progress reports by the General Atomic Company (Refs. 2 through 10).

6.6.1 REACTOR VESSEL

The reactor vessel for the gas-turbine HTGR plant bears a close resemblance to the PCRV used in the steam-cycle HTGR. The reactor core cavity is centralized in the PCRV, and the vertically positioned heat exchangers are installed in side-wall cavities. Although the technology is contemporary, the vessel geometry configuration in the reactor-vessel bottom head is different for the gas-turbine HTGR since horizontal cavities are necessary for the turbomachines. In the vicinity of the turbomachine cavities horizontal cross tendons are necessary in addition to the vertical tendons. Liner and closure features are nearly identical with those used in the steam-cycle HTGR, but modifications to the thermal barrier are necessary because of the rapid pressure equilibration rates and high sound-power levels. The operating pressure is higher than that of the steam-cycle plant (although less than that in the gas-cooled fast reactor). This, in conjunction with the aforementioned geometrical differences, necessitates model testing.

6.6.2 REACTOR-VESSEL INTERNALS

The reactor internals (including shielding, ducting, control-rod drives, and baffles) are classed in the category of components requiring modest improvement in performance or size from present (steam-cycle HTGR) knowledge. The reactor internals bear a very close similarity to those for the steam-cycle plant, but the control rods and drives, for example, are affected by the thicker top head of the PCRV (required by the higher operating pressure). The higher reactor-inlet temperature of the gas-turbine HTGR will affect the design of the reactor internals and the materials of construction.

6.6.3 PRIMARY-SYSTEM HEAT EXCHANGERS

The technology for the gas-turbine HTGR heat exchangers is regarded as contemporary. The operating environments (temperature and internal pressure differential) are less severe than for the steam-cycle design, and existing code-approved alloys are used. The precooler operates with a maximum metal temperature of less than 400°F and is thus free from creep effects. The large surface-area requirements necessitate compact surface geometries, but the tubular surface geometries and fabrication methods are regarded as state-of-the-art technology. Large tubular units of the types selected have been built and operated successfully in fossil-fired closed-cycle gas-turbine plants in Europe.

6.6.4 OTHER PLANT ACCIDENT-MITIGATING SYSTEMS

In the area of plant-protection systems there are noticeable differences from the steam-cycle HTGR. To prevent turbine overspeed, for example, a compressor-bypass valve is necessary; this, in conjunction with other valves in the primary system, is used for plant control and protection. The plant-protection system is regarded as requiring a modest improvement over the steam cycle. An external (PCRV) pressure-relief valve is not necessary in the gas-turbine HTGR plant because there are essentially two levels of pressure within the reactor vessel. Thus the relief function can be done internally within the primary circuit and will eliminate any concern over a relief valve failing in the open position since the coolant will not be lost.

6.6.5 HELIUM GAS TURBINE

The helium turbine is a unique component for the gas-turbine plant, but its development is considered to be within the state of the art. All of the aerodynamic, thermodynamic, and structural technology from open-cycle industrial-gas-turbine practice is applicable. The turbine inlet temperature of 1,562°F is modest compared with current industrial gas turbines, and this permits the use of existing nickel-base alloys (uncooled blades). Areas in the turbomachine requiring extensive development are the bearings and seals. Helium turbomachines have been built and operated in

Europe with rotor sizes of 300-MWe equivalent rating (i.e., Oberhausen). (Because of the lower operating pressure of the fossil-fired Oberhausen 2 helium-turbine power plant the actual output is 50 MWe. The high helium volumetric flow in this plant results in component sizes representative of a nuclear gas turbine with a rating of about 300 MWe).

6.6.6 COMPONENT TESTING PROGRAM

An extensive program of component testing is planned. In addition to tests with scaled compressor and turbine rigs, the following components or parameters will be tested:

1. Bearings
2. Buffering and shaft seals
3. Lubricating system
4. Welded-rotor burst
5. Containment-ring integrity
6. Flow distribution
7. Spin test
8. Sound-pressure-level attenuation techniques.

6.6.7 CONTROL VALVES

As mentioned above, the plant control and protection system does include helium-bypass valves. These valves are integrated into the primary systems and are installed in cavities in the top head of the PCRV. Three (of the four) valves operate at an elevated temperature (reactor-inlet gas) in an environment of dry helium. While large, these valves are amenable to state-of-the-art design and fabrication technologies. Because of their important role in the plant control system, an extensive development is planned to ensure a high degree of reliability.

6.6.8 HOT DUCT

The duct from the reactor outlet to the turbomachine inlet is subject to a combination of high temperature, high velocities, high sound-pressure level, pressure fluctuation, and fission-product plateout. Failure of this component could cause a severe transient resulting in core and turbomachinery damage. Recognizing this, an extensive hot duct test program is planned. The German high-temperature helium test loop may be utilized for this testing.

The testing of the hot duct will address two specific areas--the thermal barrier and the entire hot duct including the thermal barrier. Tests will include:

1. Thermal barrier material/design thermal tests
2. Static structural tests
3. Internal flow resistance tests
4. Flow-induced vibration tests
5. Acoustic vibration tests
6. Full-scale hot-flow tests

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APPENDIX A

U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

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[a]$\geq$20% U-235 in uranium, $\geq$12% U-233 in uranium, or plutonium.
[b]Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.
[c]IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countries for weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

Fuel Material Type	Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6)	
	Mixed[a]	Mechanically attached[b]
PuO_2, $HEUO_2$ powder or pellets[c]	1,000/kgHM	10,000/kgHM
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[c]	100/kgHM	10,000/kgHM
LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[a]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c]HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

Fuel material type	Minimum 2-year radiation level, (rem/hr at 1 m)	Process	Minimum initial radiation level, (rem/hr at 1 m)	References
PuO_2, $HEUO_2$ powder or pellets	1,000/kgHM	Co-60 addition	1,300/kgHM	2, 3, 5, 6
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	Co-60 addition	130/kgHM	2, 3, 5, 6
		Fission product addition (Ru-106)	400/kgHM	2, 3, 5, 6
LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4
LMFBR or GCFR fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4

REFERENCES FOR APPENDIX A

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5. Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use, IRT-378-R, (The Spiking of Special Nuclear Materials as a Safeguards Measure, Vol. 2, BNL File No. 5.9.1 for Vol. 1).

6. E. A. Straker, Material Radiation Criteria and Nonproliferation, SAI-01379-50765, Science Application, Inc., January 8, 1979.

APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission
PSEID, Volume IV, High-Temperature Gas-Cooled Reactors

Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

RESPONSES TO GENERAL COMMENTS

1. Regarding the NRC request to reduce the number of reactor concepts and fuel-cycle variations, the Nonproliferation Alternative Systems Assessment Program (NASAP) set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and tradeoffs has been determined and the findings are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each preliminary safety and environmental information document (PSEID). Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and reference classified contractor reports.

RESPONSES TO SPECIFIC QUESTIONS

Question 1

It will be necessary to establish explicit licensing criteria for the gas-turbine high-temperature gas-cooled reactor (HTGR) as a portion of its construction permit review. Many of the criteria will be based on HTGR criteria used in past licensing actions; however, it will be necessary to review and to reestablish the use of these criteria in terms of current requirements and to develop additional criteria to meet the unique aspects of the gas-turbine design. The objective of these criteria will be to ensure that a level of safety comparable with other commercial reactors is achieved. Means for establishing such criteria (in descending order of desirability) are (a) direct adoption of existing criteria (e.g., IEEE criteria and applicable Regulatory Guides), (b) adoption of existing criteria where necessary discrepancies can be justified, and (c) the development of new criteria to meet the unique aspects of the design. Preliminary criteria development during the pre-application review phase is desirable in order to guide the conceptual and preliminary design activities and to anticipate areas that will require increased attention during the construction permit review stage. We appreciate that General Atomic has been active in HTGR criteria development in the past and is presently active in developing criteria for structural graphite and in-service inspection.

One aspect that has not yet been explored is the contribution to criteria development by the Federal Republic of Germany under its cooperative agreement for the development of the gas-turbine HTGR. We are generally aware of some of the differences in criteria between the Federal Republic of Germany and the United States, but have not considered how such differences might be manifested in either the design of the gas-turbine HTGR or in its licensing criteria. We are interested in the potential effect of these differences with particular regard to in-service inspection and testing, seismic design, and requirements for redundancy and diversity of engineered safety features. Please discuss how you expect these criteria differences to influence the design and licensing criteria of the gas-turbine HTGR in the United States. If there are other criteria differences you believe are significantly different, please discuss these also (e.g., design-basis accidents, containment-system-design bases, and primary system integrity).

Response

All past licensing experience for HTGRs in the United States and in the Federal Republic of Germany have utilized existing safety criteria for light-water reactors (LWRs).

In the United States, these criteria are given in 10 CFR 50, Appendix A, General Design Criteria, and in Germany, by the "Sicherheitskriterien fuer Kernkraftwerke," published by the Bundesministerium des Innern (BMI). A comparison of these two sets of safety criteria for LWRs has been made by others. In general, the comparison shows that the criteria differ to a much lesser degree than do the respective acceptance standards in the two countries.

Similar agreement is expected in the development of safety criteria specific for high-temperature reactors (HTRs) both in the United States and the Federal Republic of Germany. An official draft of the German HTR Safety Criteria has been prepared by BMI, by supporting Technische Ueberwachungs Vereine agencies, and by the Gesellschaft fuer Reaktorsicherheit for distribution in August 1979 to the states and licensing experts of the Federal Republic of Germany. A licensing topical report is in preparation by General Atomic that will request NRC review of proposed changes to the General Design Criteria (GDC) to make them specifically HTGR criteria. Thus, there has been a recent effort to review the German BMI draft HTR criteria, compare them with the existing GDC, and briefly describe the respective acceptance standards. A full report will be available within a few months; however, present expectations are that the gas-turbine HTGR will be designed and licensed based upon the U.S. criteria.

The BMI draft of HTR safety criteria represents a reworking of the existing German LWR criteria for specific HTR design features and is also a partial updating of the criteria to incorporate new developments and knowledge. It was purposely worded to be consistent with new German gas-cooled reactor concepts, such as the high-temperature helium turbine, as well as such previous concepts as the THTR-300. In general, the criteria are consistent with current U.S. design practice and approach for both steam-cycle and gas-turbine concepts. Thus, any differences in licensing requirements in the United States and the Federal Republic of Germany will be based largely on acceptance standards, such as the German equivalent to U.S. regulatory guides and standard review plans, and not on the criteria themselves. Specific HTR-acceptance standards have not yet been developed in detail.

Major features of the BMI draft criteria are summarized below.

a. For the primary coolant boundary, a distinction is made between the primary coolant pressure-bearing (PCPB) and nonpressure bearing (liner) parts; these have different safety requirements. Requirements for the PCPB are thermal protection and monitoring, consideration of external influences on the outside of the PCPB, and periodic testing. The liner does not have these requirements.
b. For the core and reactivity criteria, as for LWRs, two shutdown systems, one of which can maintain cold shutdown, are required. Inherent characteristics of the reactor can be used in the hardware design.
c. For afterheat removal, a main, nonsafety system is required which must be available for the large majority of plant shutdowns. A core auxiliary cooling system is required which must consider frequencies of accidents, potential air or steam ingress, and minimum containment backpressure (specific safety margins are not stipulated). Common parts of the two systems are not precluded if reliability is maintained and if the parts are testable. The (N-1) redundancy rule is sufficient for maintenance operations where the loop under repair can be restored in time, considering inherent plant characteristics; otherwise (N-2) redundancy is required.
d. The containment function may be met with filtered vented confinement concepts if dose exposure limits are maintained during accidents. Atmospheric-cleanup or heat-removal systems in containment are not required. Periodic pressure and leaktightness tests are required for the containment structure and penetrations. Isolation valve requirements are similar to those in the GDC, except that positions of valves must be shown in the control room. A unique requirement was added providing an internal barrier to protect the structure and groundwater from liquids on the inside of buildings.
e. Instruments are divided into "event" and "consequence" categories with all output displayed in the control room and in an emergency control station

(required as a backup). "Event" instruments must have redundant recording and an uninterruptible power source.

f. Specific redundancy requirements, whether (N-1) or (N-2), are not specified for diesel generators. As opposed to U.S. criteria, separate redundant switchyards (as well as electrical supply lines) are required.

g. Specific requirements were added to radiation criteria for stationary activity-measuring devices with recorded output and indication and alarms in the control room; portable devices are also required. The ALARA concept is specified for accidents as well as normal operation. Specific emphasis is placed on personnel exposure during maintenance and component replacement operations.

h. For external effects, the German HTR and LWR criteria require design consideration of human-related (aircraft crash, sabotage) events as well as natural phenomena (earthquakes, storms). Requirements by the Federal Republic of Germany for seismic design are basically the same as those in the United States.

i. Regarding testability, a separate criterion was written specifying component testability as befits safety importance. Exceptions are allowed when additional requirements on design and quality control are satisfied. Consequences of failure of nontestable components must be limited.

Question 2

From our meeting with General Atomic on February 27, 1979, we understand alternatives to the reference design for the gas-turbine HTGR presented at this meeting are being considered. Please identify the nature of these alternative concepts, with emphasis on those design features most likely to affect the finality of our safety and licensing review of the reference design. If possible, indicate the degree of "firmness" that can be attached to the current reference design and estimate when decisions will be final on the incorporation or exclusion of significant alternatives.

Response

The present gas-turbine HTGR concept is a two-loop, 800-MWe plant. The first plant is intended to be replicable. With the exception of layout of the PCRV, this plant is similar to the three-loop plant described in the gas-turbine HTGR preliminary safety information document. An alternative currently under consideration is the intercooled compressor vis-a-vis the referenced nonintercooled concept. We know of no significant safety differences between these two concepts. The choice between these two concepts is to be made September 1979. Consideration is being given not only to technical factors but also to economic ones, and to the influence of the cooperation with the German/Swiss high-temperature helium turbine project which favors intercooling for its potentially higher efficiency of 2 percentage points.

Question 3

What ground acceleration value is deemed a practical maximum for the gas-turbine design? What physically limits the gas-turbine HTGR to this value?

Response

A seismic analysis of a representative gas-turbine plant will be performed in FY-80. The results of this analysis will be used to establish the seismic design requirements of the various plant components. These components include the PCRV, fuel-handling equipment, core, heat exchangers, piping, etc. The selection of ground acceleration in the design is associated with the cost. Presently, the gas-turbine HTGR is being designed for a general envelope of soil sites at a ground seismic level of 0.15/0.30 g (operating-basis earthquake/safe-shutdown earthquake (OBE/SSE)). The seismic response for the most severe soil site and base mat design is limiting. This g level covers most of the sites in the United States except for the California coast and a few others. Many sites and plant base mat designs, however, are not as severe as the envelope limit, and higher g levels than 0.15/0.30 can be used in those cases. For example, a soft soil site with some plant embedment will allow ground g levels above 0.15/0.30 g and still not be as severe as a harder soil site at the 0.15/0.30 g level. Above the 0.15/0.30 g level, each site will be evaluated separately to see if the seismic response of the plant fits within the envelope. Gas Cooled Reactor Associates has requested that an evaluation be performed at higher acceleration levels (0.2/0.4 g) at some specific sites. A detailed answer would require complete designs for each proposed g level and site.

Question 4

There are no explicit criteria directly applicable to the design, construction, and inspection of the turbine-compressor unit that we are presently aware of. Indicate to what extent existing codes may be adopted, such as the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and comment on the applicability of NRC documents that may afford guidance. A list of NRC documentation that may be useful in this regard follows:

- Standard Review Plan 5.4.1.1, "Pump Flywheel Integrity (PWR)"
- Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity"
- Standard Review Plan 4.4, "Thermal and Hydraulic Design" (material pertaining to flow oscillations, loose parts, vibrations, load-following maneuvers, part-loop operation)
- Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors"
- Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles"
- General Design Criterion No. 4
- Standard Review Plan 3.5.13, "Turbine Missiles"
- Standard Review Plan 3.5.3, "Barrier Design Procedures"
- Standard Review Plan 10.2, "Turbine Generator"
- Standard Review Plan 10.2.3, "Turbine Disc Integrity"

Response

Based upon discussions with United Technology Corporation (UTC), it has been determined that there is presently no code entirely applicable to the turbomachine; however, many existing codes, standards, and guides have sections that have applicability to the turbomachine. Design activity to date by UTC has utilized standards for FAA certification including FAA 33 Airworthiness Standards: Engines. The UTC FT-50 industrial gas turbine was designed to meet requirements of the Pacific

Gas and Electric 5.0 specification. These have also been used as a guide for conceptual design activity of the HTGR turbomachine. ANSI B31.1 is considered for piping. The ASME Boiler and Pressure Vessel Code (Section VIII) Divisions I and II are considered for material selections and stress allowables. Also considered in any future code development would be the post-spin-test ultrasonic inspection provisions of paragraphs NB2540, NB2545, and NB2546 of Section III of the ASME Code (applying to the pressurized-water reactor circulating-water-pump flywheels).

General Design Criterion 4 is applicable, as is Regulatory Guide 1.68 (perhaps with modifications). Regulatory Guide 1.14 is not specifically applicable but certain requirements, or analogous ones, will be adopted. Regulatory Guide 1.115 is applicable in principle, but the placement of the turbomachine necessitates reliance on barriers to provide protection for essential systems.

Question 5

Tabulate the thermal and mechanical limits established or being considered for normal, transient, and accident plant conditions for the fuel, control rods, structural graphite, ceramic materials, metals, and any other component of the core, the primary system, or the primary system boundary deemed safety related. Identify which of these limits have been established by past HTGR licensing actions, which limits are to be established during gas-turbine HTGR licensing reviews or topical report reviews, and which limits are to be confirmed by research and testing programs.

Response

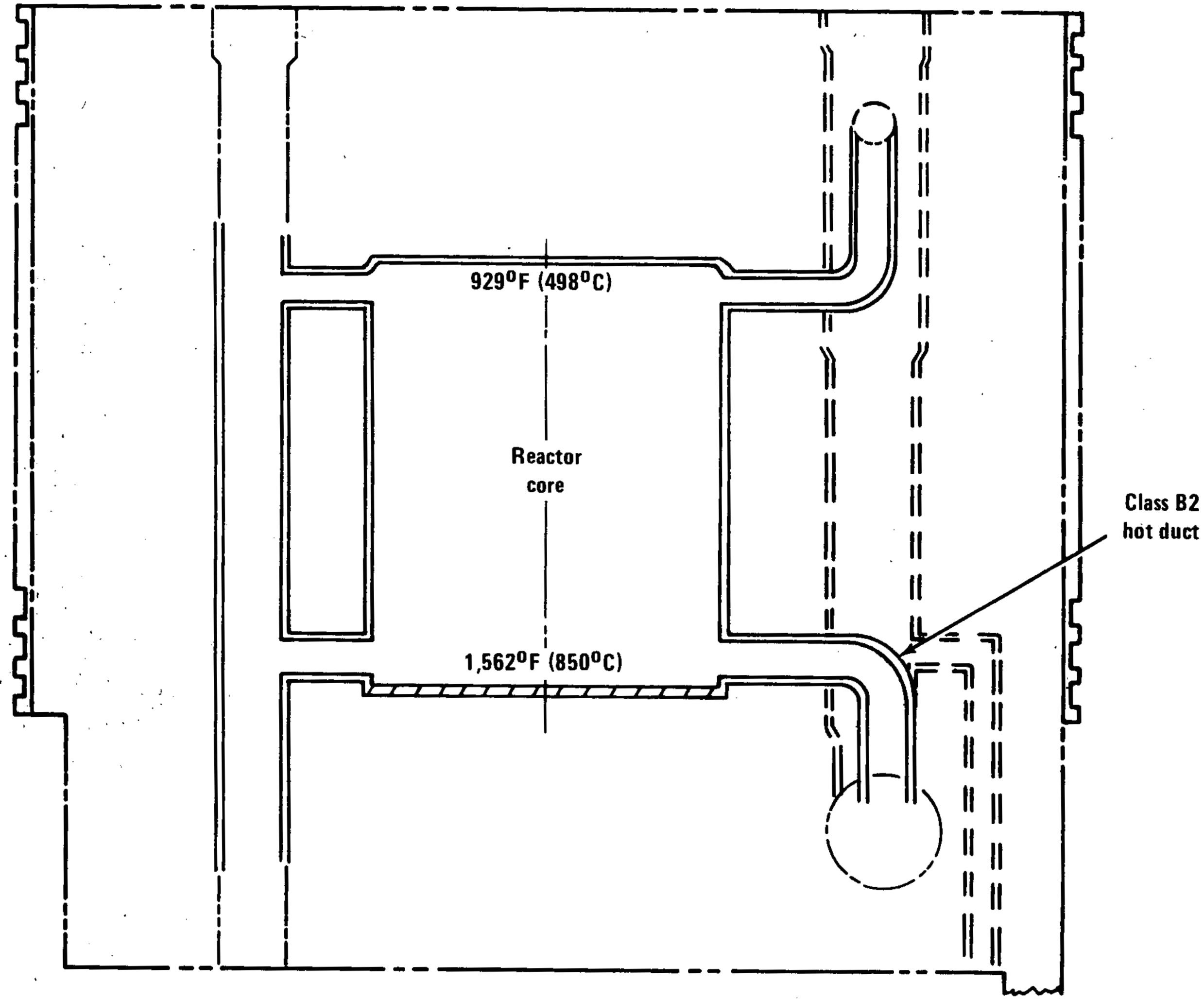
Thermal and mechanical limits are given below for the thermal barriers, reactor internal components, liners, penetrations, closures, the PCRV, the reactor core, and control rods.

a. Thermal Barrier

Four classes of thermal barrier are used in the HTGR. These classes of thermal barrier are indicated in Figure B-1 as a function of location in the primary coolant loop. Figure B-2 shows a typical Class A, Class B1, and Class B2 thermal barrier arrangement; the design concept for each of these three classes is the same, only the materials are changed. Figure B-3 is an elevation view of a Class C thermal barrier. This concept is significantly different from the other concepts in that hard ceramics are used.

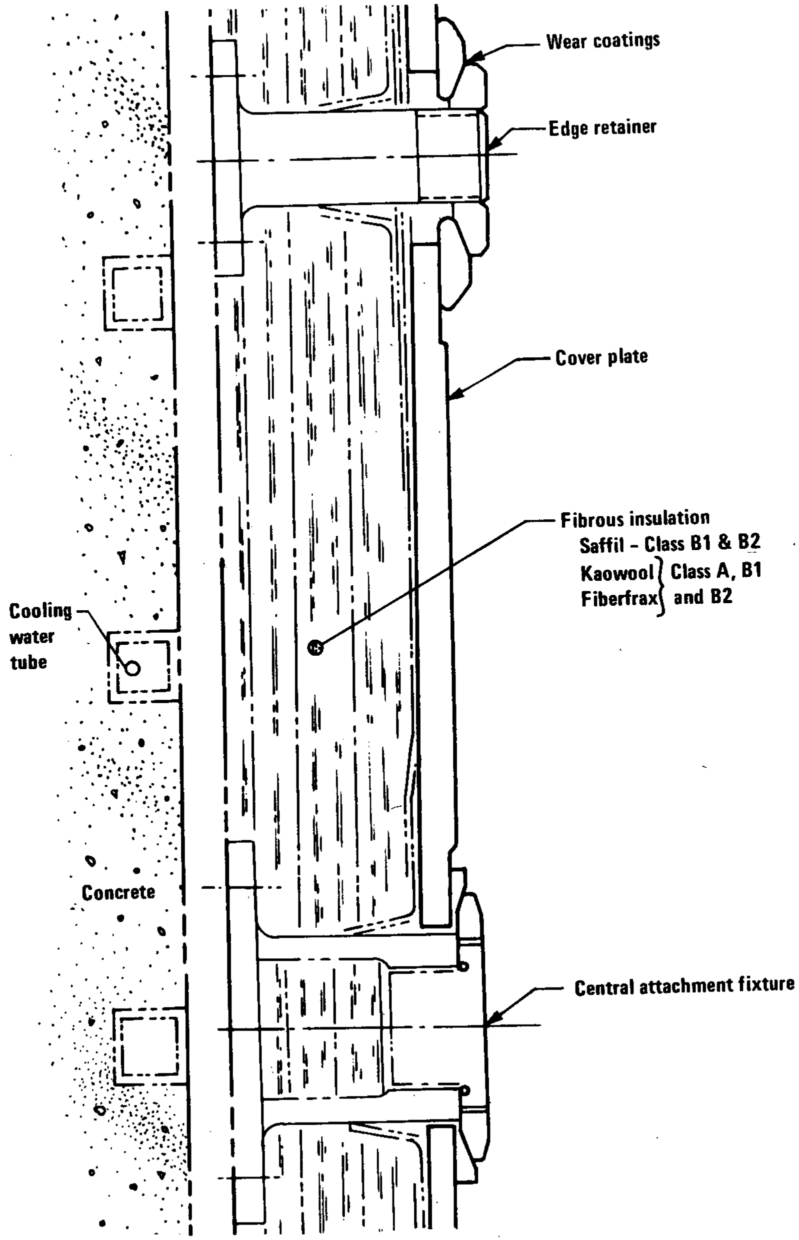
Tables B-1, B-2, B-3, and B-4 give the temperature limits and structural limits for the Class A, B1, B2, and C thermal barrier structural components, respectively. The structural limits for Class A, B1, and B2 thermal barrier apply to metallic components. As indicated in Figure B-2, carbon-carbon is a candidate material for Class B2 thermal barrier. This material is not a metal and not a typical ceramic. Therefore, if this material is used, a new structural criteria will need to be developed. These limits are to be confirmed by research and testing programs.

The structural limits for Class C thermal barrier apply to ceramic components; the metal parts are covered by Table B-2.



		Temperature limit (normal operation-300,000 hours)
Class	A	700°F (371°C)
Class	B1	1,000 - 1,300°F (538°C - 704°C)
Class	B2	1,669°F (909°C)
Class	C	1,730°F (943°C)

Figure B-1. Gas-turbine HTGR thermal barrier classes.



Section A-A

Candidate materials for coverplates and attachment fixtures:		
Class A	Class B1	Class B2
Carbon steel	2¼ Cr-1 Mo steel Type 316SS IN 800II Hastelloy-X	IN 100 IN 713 LC IN 738 Carbon-carbon

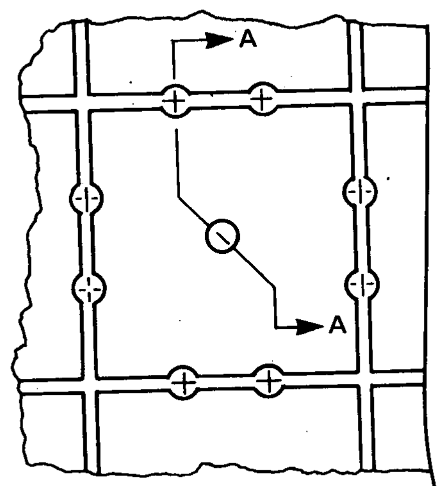


Figure B-2. Typical class A and class B thermal barrier arrangement.

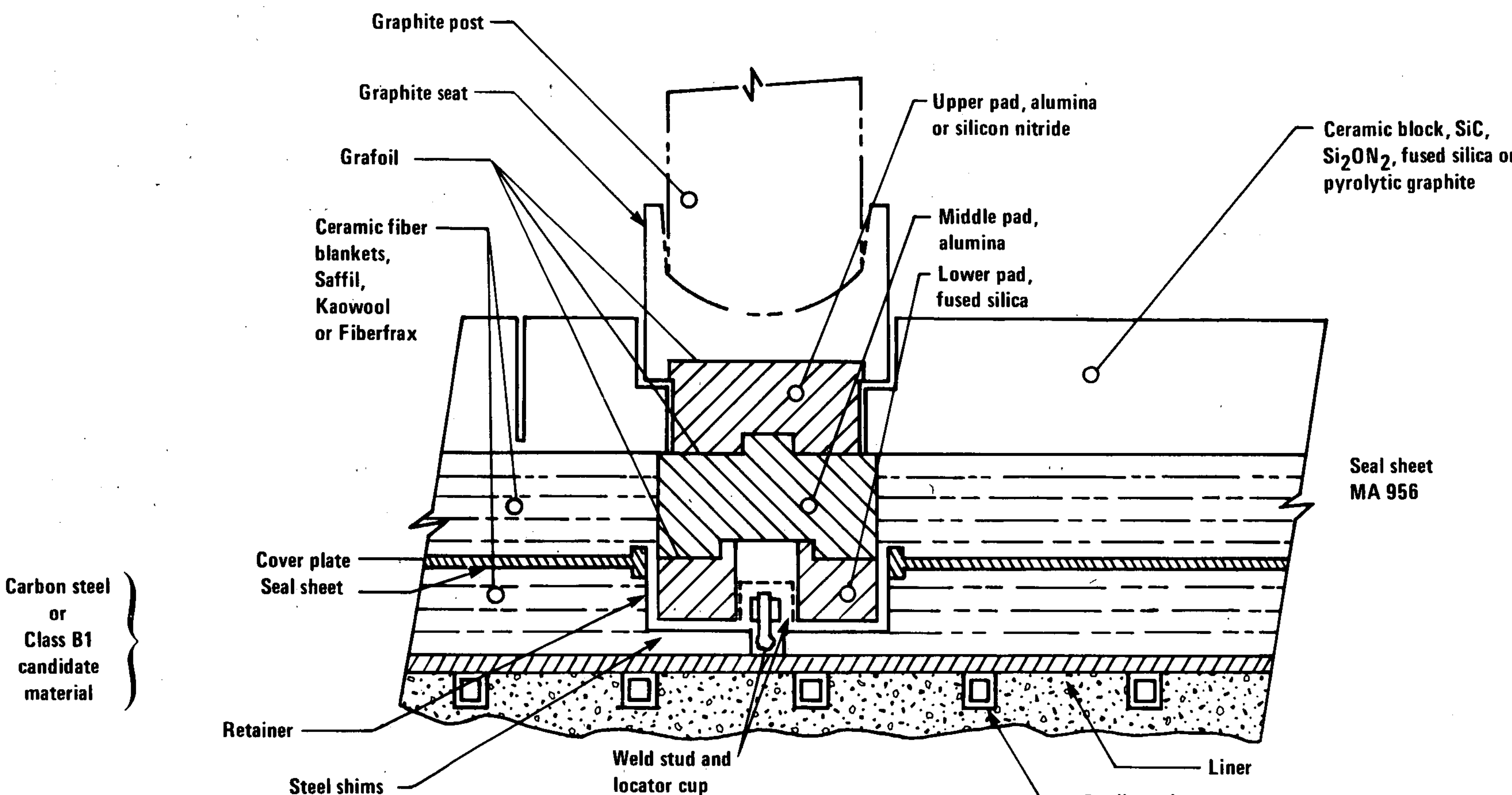


Figure B-3. Elevation view of bottom head thermal barrier.

Table B-1. The temperature (T) limits and structural limits controlling the design of the class A thermal barrier for the gas-turbine HTGR

Plant condition	Limiting temperature[a]	Loading condition	Design stress allowables[a] T $\leq 700^{\circ}F$	$>700^{\circ}F$
Normal and upset	$700^{\circ}F$	Primary membrane (P_m) plus bending (P_b)	1.5 S_m	-
		$P_m + P_b$ + secondary stresses (Q)	3.0 S_m	-
		Fatigue loading	S_a	-
Emergency	$900^{\circ}F$ for 10 hours	$P_m + P_b$	2.25 S_m	1.8 S_m or K_t S_t
Faulted	$1,100^{\circ}F$ for 1 hour	Primary stresses due to rapid depressurization	2.25 S_m or no damage or deformation which would interfere with safe shutdown of the reactor	-

[a]Limits are to be established during the gas-turbine HTGR licensing reviews or topical report reviews.

Abbreviations (as defined for the ASME Boiler and Pressure Vessel Code):

S_m = Time-independent design stress intensity.
S_t = Time-dependent design stress intensity.
S_a = Design allowable stress in fatigue.

Table B-2. The temperature (T) limits and structural limits controlling the design of the class B1 thermal barrier for the gas-turbine HTGR

Plant condition	Limiting temperature[a]	Loading condition	Design stress allowables[a] $T \leq 800^\circ F$	Design stress allowables[a] $>800^\circ F$
Normal and upset	1,000 to 1,300°F[b] for 300,000 hours	Primary membrane (P_m) plus bending (P_b)	1.5 S_m	K_t S_t
		P_m + P_b + secondary stresses (Q)	3.0 S_m	2% strain
		Fatigue loading	S_a	S_a
Emergency	(b)	P_m + P_b	2.25 S_m	1.8 S_m or K_t S_t
Faulted	(b)	Primary stresses due to rapid depressurization	2.25 S_m or no damage or deformation which would interfere with safe shutdown of the reactor	1.85 S_m or no damage or deformation which would interfere with safe shutdown of the reactor

[a]Limits are to be established during the gas-turbine HTGR licensing reviews or topical report reviews.
[b]Not yet defined; depends on material selection.

Abbreviations (as defined for the ASME Boiler and Pressure Vessel Code):

S_m = Time-independent design stress intensity.
S_t = Time-dependent design stress intensity.
S_a = Design allowable stress in fatigue.

Table B-3. The temperature (T) limits and structural limits controlling the design of the class B2 thermal barrier for the gas-turbine HTGR

Plant condition	Limiting temperature[a]	Loading condition	Design stress allowables[b] $T \leq 1{,}000^{\circ}F$	$>1{,}000^{\circ}F$
Normal and upset	$1{,}669^{\circ}F$ for 300,000 hours	Primary membrane (P_m) plus bending (P_b)	$1.5\ S_m$	$K_t\ S_t$
		$P_m + P_b$ + secondary stresses (Q)	$3.0\ S_m$	2% strain
		Fatigue loading	S_a	S_a
Emergency	$1{,}800^{\circ}F$ for 10 hours	$P_m + P_b$	$2.25\ S_m$	$K_t\ S_t$
Faulted	$2{,}000^{\circ}F$ for 1 hour	Primary stresses due to rapid depressurization	$2.25\ S_m$ or no damage or deformation which would interfere with safe shutdown of the reactor	$1.85\ S_m$

[a]Limits are to be established during the gas-turbine HTGR licensing reviews or topical report reviews.

[b]Limits are to be confirmed by research and testing programs.

Abbreviations (as defined for the ASME Boiler and Pressure Vessel Code):

S_m = Time-independent design stress intensity.
S_t = Time-dependent design stress intensity.
S_a = Design allowable stress in fatigue.

Table B-4. The temperature (T) limits and structural limits controlling the design of the class C (ceramic) thermal barrier for the gas-turbine HTGR

Plant condition	Limiting temperature[a]	Loading condition	Design stress allowables[b]
Normal and upset	1,730°F for 300,000 hours	Primary membrane (P_m) plus bending (P_b)	0.5 x $(S)_{.99}$ x BF x TF
		$P_m + P_b$ + secondary stresses (Q)	
		Fatigue loading	$(S_a)_{.99}$
Emergency	2,500°F for 10 hours	$P_m + P_b$	.75 x $(S)_{.99}$ x BF
Faulted	3,000°F for 1 hour	Primary stresses due to rapid depressurization	.90 x $(S)_{.99}$ x BF or no damage or deformation which would interfere with safe shutdown of the reactor

[a]Limits are to be established during the gas-turbine HTGR licensing reviews or topical report reviews.
[b]Limits are to be confirmed by research and testing program.

Abbreviations:

$(S)_{.99}$ = The strength of the material corresponding to a 99% probability of survival.
BF = Biaxiliaty or triaxiality stress factor.
$(S_a)_{.99}$ = Design allowable stress in fatigue associated with a 99% probability of survival.
TF = Time factor to account for static fatigue.

b. Reactor Internals Components

Tables B-5 through B-8 summarize the thermal and mechanical limits established for the four major reactor internals components, i.e., the core support floor, permanent side reflector, core lateral restraint, and core peripheral seal.

The core support floor and permanent side reflector are graphite structures and the core lateral restraint and core peripheral seal supports are metallic structures. New mechanical limits for graphite components will be established by code committee activity and confirmed by research and testing programs as required.

c. Liners, Penetrations, and Closures

The thermal and mechanical limits for liners, penetrations, and closures have been established in previous licensing reviews and are contained in the ASME Boiler and Pressure Vessel Code, Section III, Divisions 1 and 2. Tables B-9 and B-10 contain lists of the applicable code references that specify these limits.

d. PCRV

The thermal and mechanical limits for the PCRV have been established in previous licensing reviews and are contained in the ASME Boiler and Pressure Vessel Code, Section III, Division 2. Table B-11 lists the applicable code references that provide these limits.

e. Core and Control Rods

1. Mechanical Limits

The primary mechanical design basis for the reactor core is that the array of fuel and reflector elements is capable of efficiently transferring the generated fission heat to the helium coolant while maintaining structural integrity and containing the fission products under all normal operating conditions and anticipated transient conditions.

The following limits apply to the graphite fuel elements:

(a) The irradiation-induced dimensional change of the individual graphite elements shall be maintained within the following limits:

Element length	0.5% expansion 5.0% contraction
Element width	0.5% expansion 2.0% contraction
Element bow	0.15 in.

Table B-5. Core support floor (graphite components)

Limits plant condition	Limiting condition	Comment
Thermal[a]		
Normal	1,562°F--Resulting in a maximum spring pack deflection of 0.1"	There are no specific temperature limits on graphite itself, but thermal expansion of the graphite compresses the core lateral restraint spring packs and these are limiting as defined.
Upset	Same as normal	Same as normal.
Emergency	2,000°F--Results in complete compression of soft spring	0.6" soft spring deflection.
Faulted	3,400°F--Results in complete compression of soft and hard spring at end of life less maximum PCRV movements inward	0.6 soft spring deflection + 0.625 hard spring deflection-- PCRV creep PCRV shrinkage.
Mechanical		
Normal	$\sigma \leq 0.2\ \sigma_{ult}$	Primary stress.
	$\sigma \leq 0.4\ \sigma_{ult}$	Primary plus secondary stresses.
Upset	Same as normal plant condition.	Same as normal plant condition.
Emergency	$\sigma \leq 0.33\ \sigma_{ult}$	End of life, operating basic earthquake, oxidized.
	$\sigma \leq 0.67\ \sigma_{ult}$	Primary plus secondary stresses.
Faulted	No loss of function	The required safety factor of the component must be demonstrated by testing.

[a]To be established during the gas-turbine licensing reviews or topical report reviews.

[b]Established by past HTGR licensing actions. (New limits will be established by code committee activity and confirmed by research and testing programs.)

Table B-6. Permanent side reflector (graphite components)

Limits plant condition	Limiting condition	Comment
Thermal[a]		
Normal	Same as core support floor (CSF)	Same as CSF
Upset	Same as CSF	Same as CSF
Emergency	Same as CSF	Same as CSF
Faulted	Same as CSF	Same as CSF
Mechanical[b]		
Normal	$\sigma \leq 0.33\ \sigma_{ult}$	Primary stress
	$\sigma \leq 0.4\ \sigma_{ult}$	Primary plus secondary stresses
Upset	Same as CSF	Same as CSF
Emergency	Same as CSF	Same as CSF
Faulted	Same as CSF	Same as CSF

[a]To be established during the gas-turbine HTGR licensing reviews or topical report reviews.

[b]Established by past HTGR licensing actions. (New limits will be established by code committee activity and confirmed by research and testing programs.)

Table B-7. Core lateral restraint (metallic components)

Limits plant condition	Limiting condition	Comment
Thermal[a]		
Normal	1,000°F	Yield strength of spring material >100 ksi.
Upset	1,000°F	
Emergency	1,400°F	Temperature at which springs exhibit excessive creep relaxation when soft spring fully compressed to 0.6 in. and hard spring 0.625 in. at end of plant life.
Faulted	1,400°F	
Mechanical[b]		
Normal	σpipe $< 2/3\sigma_{yield}$ at EOL	Load to crush support.
Upset	σpipe $< 2/3\sigma_{yield}$ at EOL	
Emergency	σpipe $< \sigma_{yield}$	It is not arbitrarily assumed that the earthquake occurs at this time. Other stresses are maximum.
Faulted	σpipe $< 0.8\sigma_{ult}$	

[a]To be established during the gas-turbine licensing reviews or topical report reviews.

[b]Established by past HTGR licensing actions.

Table B-8. Core peripheral seal (metallic components)

Limits plant condition	Limiting condition	Comment
Thermal[a]		
Normal	1,400°F	Must maintain prescribed cooling; to be confirmed by testing program
Upset	1,400°F	Same as normal
Emergency	1,700°F for 10 hours	LOFC with full helium inventory
Faulted	2,000°F for 1 hour	Design-basis depressurized accident
Mechanical[b]		
Normal	$\sigma \leq 2/3\ \sigma_{yield}$	Pressure drop
Upset	$\sigma < 2/3\ \sigma_{yield}$	Operating-basis earthquake plus
Emergency	$\sigma \leq \sigma_{yield}$	To be established
Faulted	$\leq 0.8\ \sigma_{ult}$	To be established

[a]To be established during the gas-turbine licensing reviews or topical report reviews.

[b]Established by past HTGR licensing actions.

Table B-9. Liner design limits

Plant condition	Service level	ASME III, Div. 2 mechanical limits
Normal	A	Table 3700-1 and -2
Upset	B	Table 3700-1 and -2
Emergency	C	Table 3700-1 and -2
Faulted	D	TAble 3700-1 and -2

Note: Liner temperatures are limited by the temperature limit for the adjacent concrete (Table CB-3430-1 of Division 2).

Table B-10. Steel penetration (not backed by concrete) and closure design limits

Plant condition	Service level	ASME, Division 1 mechanical limit Free flow area >10 in.^{2}	 Free flow area ≤ 10 in.
Normal	A	NB-3222	NC-3217 or NC-3321
Upset	B	NB-3223	NC-3217 or NC-3321
Emergency	C	NB-3224	NC-3217 or NC-3321
Faulted	D	NB-3225	NC-3217 or NC-3321

Note: Steel penetration and closure temperatures are limited to the maximum values listed in Tables I-1.0 and I-7.0 of Appendix I to Section III, Division 1, of the ASME Boiler and Pressure Vessel Code.

Table B-11. PCRV design limits[a]

Plant condition	Comment
Thermal[b]	
Construction	Temperature limits ensure that the range of material properties considered are maintained.
Normal	
Abnormal and severe	
Environment (upset)	
Extreme environment (emergency)	
Failure (faulted)	
Mechanical[b]	
Construction	
Level A (normal)	
Level B (upset)	
Level C (emergency)	Safe shutdown can be achieved and maintained.
Level D (faulted)	Structural integrity can be maintained.

[a]Limiting condition:
Thermal: Table CB-3430-1 of ASME, Section III, Division 2.
Mechanical: Tables CB-3421-1 and 2 of ASME, Section III, Division 2.

[b]Established in ASME code, Section III, Division 2, and by past HTGR licensing action.

(b) The effect of seismic loads on the fuel elements shall not exceed the following:

(1) Operating-basis earthquake: No core element disarray or damage shall occur such that normal full power operation cannot be maintained or resumed.

(2) Safe-shutdown earthquake: The core elements shall retain their structural configuration to allow sufficient control poison to be inserted into the core to ensure safe shutdown and allow sufficient coolant flow to be maintained through the coolant channels to remove the reactor core decay heat.

The mechanical limits for the control rod pairs are set primarily to ensure insertion of the rods in the core under all normal and accident conditions, and to minimize the probability of binding of a single rod pair under the normal core temperature and radiation environment. Thus, the design of the rod itself consisting of 15 boronated segments attached by ball joints is chosen to allow free motion under gravity with any credible misalignment of the control channel. This design also prevents warpage of the rods due to thermal gradients. Binding of the rods is prevented by providing a nominal diameter clearance between the rod and channel of 1.27 cm (0.5 in.), and a worst-case limit under thermal and irradiation-induced dimensional changes of 0.97 cm (0.38 in.). As a result of thermal growth and tolerance buildup, the control rod pairs have the following maximum envelope dimensions:

Length	273.9 in.
Diameter	3.58 in. (including warpage)
Canister separation	0.293 in.

2. Thermal Limits

The characteristics of the reactor core thermal design are established to protect the integrity of the reactor primary coolant system boundary, the core coolant flow geometry and the channels for insertion of neutron poisons, and the fission-product barriers within the core.

Thermal limits for the fuel elements and hexagonal reflector elements are based on the mechanical strength of graphite, which increases with temperature and reaches a maximum at about 2,500°C (4,350°F). The temperature limits for graphite components are based upon the properties of graphite at elevated temperatures. The limit for normal and upset conditions is set at 2,400°C (4,350°F), which is about 100°C below the temperature at which the strength of graphite stops increasing with temperature and begins to decrease rapidly as the temperature is raised. The temperature limit for emergency conditions is set at 2,500°C (4,530°F) at which the strength reaches a maximum. The limit under faulted conditions is set by making a conservative estimate of 3,000°C (5,430°F) based upon the phase diagram to limit sublimation to a negligible rate. This is about 600°C (1,080°F) below the temperature at which the vapor pressure of graphite reaches 1 atm.

The HTGR fuel and fuel-rod materials consist of refractory graphite and ceramic materials. Uranium carbide has the lowest melting point at 2,450°C (4,450°F). Silicon carbide does not melt, but begins to sublime at temperatures above 2,000°C (3,630°F). Carbon coatings and the fuel-rod matrix begin to sublime at temperatures above 3,000°C (5,430°F).

3. Metallic Core Components

The temperature limits for the plenum elements are set by the mechanical properties of the material. For the steam-cycle HTGR, 316 stainless steel was used and a limit of 427°C (800°F) established for normal and upset conditions. Both the choice of material and the temperature limits are under review for the gas-turbine HTGR and have not been established as yet.

Similarly for the control rod clad, a temperature limit of 870°C (1,600°F) had been established for normal and upset conditions, and 1,090°C (2,000°F) for accident transients shorter than 1 hour integrated over rod lifetime. These limits are being reviewed and updated for gas-turbine HTGR applications. In addition, under accident conditions the general integrity of the poison compacts shall be maintained in control rod channels and the temperature of the poison compact shall not exceed 2,400°C (4,300°F) under any reactor condition of design. This temperature is the conservative upper limit for the prevention of boron transport from the compacts to the graphite blocks. It is also conservative upper limit for compact integrity. No appreciable change in compact geometry takes place at that temperature.

Question 6

What additional features of the plant-protection system or engineered safety features may be needed to cope with failure modes of the grey control rods, the turbine-compressor unit, primary system valve, the recuperator, hot duct, and the precooler? Responses to this question will require identification of or reference to failure mode studies, postulation of a spectrum of accidents, predicted responses of the existing plant-protection system and engineered safety features, and information on potential system interactions. We anticipate that it may not be possible for you to supply definitive responses to this question in the near future. Nevertheless, we expect that you should be able to supply preliminary and conceptual responses together with a discussion of the status of related accident studies together with an estimate of when this question can be answered finally.

Response

The current plant-protection requirements have been developed for the gas-turbine HTGR based on the preliminary safety studies (Refs. 1 and 2) and experience in design of the steam-cycle HTGRs. The basic objective of the plant-protection system is to prevent an unacceptable release of radioactivity that would constitute a hazard to the health and safety of the public and to ensure that the plant can be shut down and maintained in a safe-shutdown mode for a spectrum of hypothetical low-probability events that might lead to failure of the fission-product-retention barriers. The plant-protection system functions to initiate actions that will protect the fission-product barriers. If failures do occur in the barriers, backup actions are initiated to limit the release of radioactivity. To accomplish these functions the plant-protection system includes major systems that perform the following:

a. Initiate rapid reduction in reactor power level following reactivity excursions, loss of adequate core cooling, and other events, in order to minimize the damage to fuel coatings and preserve the integrity of the primary-coolant boundary (reactor trip system).
b. Initiate rapid reduction of helium flow to the turbine to prevent damage to the upper plenum thermal barrier following a total or partial loss of normal precooler flow or to limit peak turbomachine overspeed or primary-coolant overpressure and to allow proper functioning of the core auxiliary cooling system (CACS) (main loop shutdown system).
c. Initiate auxiliary core cooling to prevent damage to core and PCRV internals following the loss of effective main loop cooling (CACS).
d. Limit fission-product release following a precooler-tube rupture (precooler isolation and dump system).
e. Prevent the withdrawal of more than one control rod bank, simultaneously restricting the possibility of reactivity excursions that can be initiated by control rod withdrawals (single rod bank withdrawal interlock system).

Table B-12 is a listing of key protective equipment groups in the gas-turbine HTGR compared with the analogous system in the steam-cycle HTGR. Table B-13 is a listing of the conceptual plant-protection system functions planned for the gas-turbine HTGR and their relation to previously defined steam-cycle functions. (See the response to Question 1 for additional details.) Additional protective system actions, above those presented in Table B-13, are not anticipated for failure of components within the primary system. Extensive analyses of failure mechanisms, probabilities of failure, and failure consequences are planned for all active and passive components of the system. Assurance will be attained that the current design is adequate or that design changes are implemented so that the consequence of any component failure does not present an unacceptable risk. The overall analyses will include plant-system-level analyses, including all subsystems that may be involved in a postulated accident sequence. Guidance in the acceptability of the design can be obtained by employing the methods of probabilistic risk assessment to determine the significance of an accident sequence with respect to other sequences.

Preliminary gas-turbine HTGR accident failure mode and analysis based on the PSID design (Ref. 1) have been performed, but additional effort (which is currently planned) is needed. The comprehensive risk assessment (Refs. 3 and 4) of the steam cycle HTGR demonstrated the effectiveness of the graphite core in retaining fission products under a spectrum of accident conditions. Because of the large heat capacity of the core, a significant amount of time is allowed for operator corrective actions even when hypothesized multiple failures caused all cooling systems to be inoperative. Since the core in the gas-turbine HTGR is very similar to the steam-cycle HTGR, conceptually the same level of safety can be achieved. However, a number of design differences exist that give rise to unique accident sequences in the gas-turbine HTGR. A detailed risk assessment is planned over the next 3 years. The preliminary work to date indicates adequate protection has been considered for the control rods (Ref. 4). The turbomachine failure modes and their transient effects are being studied in great detail. Failure modes of the recuperator appear to be benign with respect to safety. Some failure modes of the hot duct and precooler need additional study and analysis to ensure that the effects of such failures are acceptable (Refs. 1, 2, and 5).

Table B-12. Gas-turbine HTGR protective equipment comparison with steam-cycle HTGR

System	Remarks
Control rod system	Same as steam-cycle HTGR
Core auxiliary cooling system	Same as steam-cycle HTGR
Containment isolation system	Same as steam-cycle HTGR
Containment overpressure protection system	Same as steam-cycle HTGR
Containment cleanup system	Same as steam-cycle HTGR
Precooler isolation/dump system	Analogous to HTGR steam generator isolation/dump system
Safety bypass valve system	Unique to steam-cycle HTGR; redundant valves in each loop, redundant and diverse signals for trip
Turbomachine burst shield	Passive shield capable of preventing internally generated missiles being radially ejected; larger than the circulator burst shield in the steam-cycle HTGR

Table B-13. Preliminary summary of protective functions of the gas-turbine HTGR plant-protection system

Protection function	Initiating condition	Relation to steam-cycle HTGR function
Reactor trip	High-reactor-power-to-helium flow ratio at high flow (1.4)	Analogous
Reactor trip	High reactor power at low flow	Same
Reactor trip	High reactor flux during refueling or low-power testing	Same
Reactor trip	High helium inlet temperature to turbine (1,600°F)	Analogous to high steam generator inlet temperature
Reactor trip	High primary coolant pressure[a]	Analogous
Reactor trip	High containment radiation level	Same
Reactor trip	High containment pressure (20 psia)	Same
Reactor trip	Loss of preferred bus voltage	Same
Reactor trip	Two or main loop shutdown signals)	Analogous to "two-loop trouble"
CACS initiation	Low plant helium flow	Same
Main loop shudown	High power conversion loop exit temperature (975°F)	Analogous to circulator outlet temperature
Main loop shutdown	High turbomachine speed (3,960 rpm)	Analogous to high circulator speed
Main loop shutdown	High primary coolant pressure	Analogous to high primary coolant in steam-cycle HTGR but revised action for relief
Main loop shutdown	CACS initiation	Same
Main loop shutdown	PCS main loop trip	Similar to loop isolation
Main loop shutdown	Manual loop shutdown	Same
Precooler isolation and dump	High activity in precooler water outlet line	Analogous to steam generator isolation and dump
Single control rod bank withdrawal	Detection of outward command to two or more control rod banks	Same

[a]A setpoint for overpressure protection has not been determined since no mechanistic source of significant overpressure has been identified. It will be set at a value that will provide required margins to prevent trip in the event of "all loops to overspeed trip."

Question 7

The discussion of certain low-probability accidents in the PSEID should be amplified beyond the use of the results of the Accident Initiation and Progression Analysis (AIPA) study. In particular, describe the hypothetical consequence of a control-rod-ejection accident, consequences from a spectrum of failures in the core support structure, and the consequences of water injection from a failed precooler with the simultaneous rapid depressurization of the reactor.

Response

Control-rod-ejection accidents and core support failures were not evaluated in detail during the HTGR AIPA study because preliminary screening of these accidents showed them to be of very low probability and, hence, low risk contributors.

The design of the refueling penetration in the PCRV reduces the probability of a control rod pair ejection accident to a low value. The penetration design, materials, and fabrication are in accordance with ASME Section III, Class 1, code requirements for vessels, and are very similar to those of a light-water reactor vessel. All pressure boundary welds are full-penetration 100% radiographed during fabrication, and subject to volumetric examination during in-service inspection in accordance with ASME Section XI, Division 2, code rules. Primary and secondary shear anchors are provided on the refueling penetrations. The secondary anchor is capable of transmitting the axial pressure load on the penetration closures to the PCRV within ASME Section III Service Limit D when the primary anchor is postulated to be ineffective. Although current designs do not include coverplates over the refueling penetrations, as in earlier designs, low probability of closure failure combined with the relatively mild reactivity excursion associated with the accident (especially for the medium-enrichment uranium fuel) and the inherent safety of the HTGR results in relatively low risk. However, this issue will have to be studied in greater detail before a final conclusion can be reached.

A disruption of the core assembly is highly improbable since it requires an occurrence of an immense force to achieve a disarrangement of the core fuel elements. It is unlikely that an earthquake can accomplish such a disruption. To consider the possibility of core disruption further, it is necessary to look for a sufficiently massive structural failure of the PCRV that causes a loss of support or the creation of very large flow forces that can levitate the core. It is difficult to find a mechanism to cause a major loss of core support. Furthermore, such a failure does not result in a direct loss of PCRV integrity and subsequent release of fission products. Major disruption of the core by levitation resulting from a penetration failure or pressure equilibration accident has not specifically been analyzed for the gas-turbine HTGR. However, such an event, leading to core heatup, was considered during the AIPA study (Ref. 4).

Disruption of the core assembly resulting from failure of one or more of the core support posts due to burnoff is also considered improbable even though the posts in the gas-turbine HTGR will be subject to higher temperatures than the steam-cycle HTGR. Although detailed analyses and some experiments will be required to verify this conclusion, it should be noted that the secondary sides (water) of the core auxiliary heat exchangers (CAHEs) and precoolers of the gas turbine operate at lower pressures than the primary coolant system, thereby preventing significant water ingress to the primary coolant system. In addition, studies for the steam-cycle HTGR indicate that core support post burnoff is concentrated on the surface, which results in lower strength loss than from a uniform burnoff distribution.

The consequences of water injection from an accident sequence involving both a failed precooler and rapid depressurization of the reactor has not been analyzed for the gas-turbine HTGR. However, the consequences and uncertainties are believed to be similar to the accident scenarios involving both PCRV depressurizations and steam generator failures which are discussed in Section 5.1 of Reference 3 for the steam-cycle HTGR.

Question 8

The low-probability accident customarily used for siting studies is an adiabatic core heatup caused by the sustained loss of forced convection cooling. Discuss the potentials for mitigation of this accident by designing for emergency heat removal by natural convection. What are the helium pressure requirements for emergency cooling by natural convection and how would these requirements vary with time after the accident? What role might the containment vessel and containment back pressure provide in natural convection cooling?

Response

The siting event for the HTGR already meets the dose requirements given in 10 CFR 100, and therefore mitigation of this accident is not needed or required. However, the technical aspects of natural convective cooling are discussed below.

As presently designed, the gas-turbine HTGR does not have the potential for emergency heat removal by natural convection because, unlike the case with the steam-cycle HTGR, the main loops cannot be used as a cold leg to form a loop with the core as a hot leg. The other alternative for a cold leg, the CAHE, could potentially be used if the check valves (operated by pressure differential) were redesigned to be manually controlled. Preliminary analysis has indicated that the core could possibly be cooled by upflow natural convection through the core and downflow through the CACS cavities; if the system remains pressurized, flow blockage does not occur, and an ultimate heat sink is provided for the CACS.

With regard to the pressure requirements for emergency natural convection cooling, an extensive analysis over the entire pressure range has not been performed. Calculations have shown that the potential exists at normal operation pressures, but that when the primary system is depressurized to containment back pressure, free convection is insufficient to cool the core. The required flow as a function of time after trip follows the decay heat transient such that approximately 1% of normal operation flow is required at 3 hours and 0.5% at 4 days to prevent continued core heatup. In the past, the siting event has been arbitrarily assumed to be a core heatup combined with the design-basis depressurization accident. Results of the AIPA study indicate, on the other hand, that loss of forced circulation is much more likely to occur with the PCRV pressurized.

Since both the steam-cycle and gas-turbine HTGRs use downflow cores and the upper components are designed for temperatures of the core inlet, the top plenum and the upper CACS cavity components are likely to be damaged if the natural convection cooldown is initiated after a loss of forced circulation. The preliminary analysis indicates that perhaps the damage would be reparable and would not cause flow blockage. If there is any period of forced circulation after reactor trip for as little as 5 to 10 minutes, either by main loop rundown or limited CACS cooling, the prospect of natural convection cooldown without any damage is greatly improved. The large mass of the generator and turbomachine combined with the stored energy in the primary system, if it is not depressurized, should ensure such a rundown.

The objective in the design of the HTGR cooling system is a high degree of reliability which, as shown in the AIPA study, has been achieved through the diverse and redundant main loop cooling system and CACS and the capability of the primary coolant system to tolerate long interruptions in forced circulation without sustaining damage. Whether natural convection could significantly increase this level of reliability is not certain.

Question 9

Substantially more information should be supplied with respect to internal pressure-equilibration accidents in comparison with rapid depressurization accidents. Describe design criteria and design changes that might be needed to cope with the larger differential pressure forces experienced by thermal barriers, flow diffusers, and other primary system components and boundary surfaces. Are any of the needed design changes sufficiently beyond the state of the art that development programs will be necessary?

Response

Because of the large turbine and compressor pressure ratios, the primary coolant system is divided into "high" and "low" regions of pressure. Accidents such as turbine deblading, compressor deblading, and catastrophic failures of the recuperator can lead to "pressure-equilibration accidents," wherein some regions of high pressure depressurize and regions of low pressure pressurize. If the failures are assumed to occur over short periods of time, local rates of pressure change can be significantly larger than those which would occur during a DBDA. An extensive search for failure mechanisms has been carried out and analysis indicates that complete deblading of the turbine may result in the most rapid pressure transient.

The following provides more information about the design criteria and design changes that might be required to cope with the larger differential pressure forces experienced by the thermal barrier due to the pressure-equilibration accident experienced by the gas-turbine HTGR. To ensure the structural integrity of the thermal barrier, design verification and support requirements are identified.

a. Design Criteria

The thermal barrier design criteria for the pressure-equilibration and rapid-depressurization accidents are the same, since both are faulted plant conditions. The structural criteria for the faulted condition require no damage or deformation that would interfere with the safe shutdown of the reactor. Structural limits for the thermal barrier are given in Tables B-1 through B-4 in the response to Question 5.

It is assumed in the definition of the failure criteria for metallic components that these types of accidents occur so rapidly that the time-dependent stress intensity, S_t, is not an important parameter.

b. Design Changes

Since it must be assumed that the gas-turbine HTGR can experience pressure-equilibration accidents with depressurization rates significantly higher than the maximum rate of 50 to 60 psi/sec which was expected for steam-cycle HTGR plants, the effects on the thermal-barrier design are important. To accommodate the larger pressure differentials experienced by the gas-turbine thermal barrier, the coverplates are designed to be thicker than those for the steam-cycle HTGR and a venting capability must be incorporated in the thermal-barrier design. One design concept uses a mesh-supported vent cavity with vent holes in the coverplate. Analytical results indicate that in the absence of flow resistance from the fibers, the vent area that is required to ensure structural integrity of the cast coverplates is acceptable for the coverplate design. If coupled with a preferential vent cavity flow path, there is no adverse effect on the amount of permeation heat flow experienced by the thermal barrier during normal operation.

Current information indicates that the ability of the thermal barrier to survive a pressure-equilibration accident is controlled by the ventability of the fibrous insulation and not by the coverplate.

Based on available analyses, the following design verification and support tests may be required:

- Permeation flow tests on the candidate fibrous insulation materials; tests to be conducted in helium.
- Depressurization tests to determine the effects of depressurization on the fibrous and ceramic materials. Evaluation of rapid depressurization/pressurization to be considered as well as the cyclic effects of relatively low rates of depressurization/pressurization; tests to be conducted in helium.
- Permeation flow tests on the various full-scale thermal barrier sections; tests to be conducted in helium.
- Depressurization tests of the full-scale thermal barrier sections; tests to be conducted in helium.

Question 10

The direct cycle concept offers the potential advantage that water and other oxidant materials could be totally eliminated from the primary system by using a nonoxidant fluid in the precooler. Discuss the practicalities of this suggestion.

Response

The use of nonoxidants such as helium in the secondary cooling system has been briefly considered but rejected for reasons of economics. The precoolers themselves would be excessively expensive and the resulting increase in PCRV size would be very costly.

We intend to use a controlled water chemistry during normal operation. If the system is drained, it will be inerted with N_2 as in the case of a steam cycle. During normal operation, the water pressure in the precooler is well below that of the helium, and because the precooler operating temperature is rather modest, the potential for water inleakage is alleviated.

Question 11

The information provided in the PSEID on in-service inspection and testing was too generalized for our needs. Further, while you maintain that state-of-the-art equipment and practices are adaptable to current ASME Code requirements, we point out that Division 2 of Section XI has not yet been adopted by either the ASME or the NRC. Please revise your response with emphasis on the needs and means for in-service inspection, with special consideration of the following: (1) base and lateral core support structures, (2) the thermal barrier, (3) the PCRV liner, and (4) the restraint mechanisms that preclude control rod ejection. As equipment design relative to the gas-turbine plant develops in more detail, we will expect more information than presented on February 27th pertaining to the needs and means for inspection of these developing designs.

Response

Proposed Section XI, Division 2, ASME Boiler and Pressure Vessel Code, "Rules for Inspection and Testing of Components of Gas-Cooled Plants," was originally published for a 1-year review and comment period, terminating September 15, 1978, and subsequently extended to September 15, 1979. Following the disposition by ASME of comments and discussions resulting from the trial period, it is the intent of the Society to publish the Code as a mandatory division of Section XI.

Covered by the proposed rules are the following major areas:

Subsection IGA - General Requirements
Subsection IGB - Requirements for Class 1 Components
Subsection IGC - Requirements for Class 2 Components
Subsection IGD - Requirements for Class 3 Components
Subsection IGG - In-service Inspection of Reactor Internals
Subsection IGK - In-service Inspection of Concrete Reactor Vessels
Subsection IGQ - In-service Testing of Pumps
Subsection IGP - In-service Testing of Compressors
Subsection IGV - In-service Testing of Valves

Subjects recently approved by ASME Section III Main Committee and to be included in the planned issue of the Code are the following:

Subsection IGH - In-service Inspection of Elevated Temperature Material
Subsection IGI - In-service Inspection of Non-Metallic Material in Reactor Internal Components

Section XI, Division 2, rules address a single concept of gas-cooled reactors, namely the HTGR. Concepts other than the HTGR were considered to be of lesser priority at the time of initial code development. The code in its present form is not editorially applicable to the gas-turbine HTGR, but certain design features characteristic of the HTGR are also used in gas-turbine HTGRs. The charter of the ASME group responsible for the development of rules for gas-cooled reactors is currently being modified to include gas-turbine HTGRs in the spectrum of code applicability.

In response to the areas of concern identified and consistent with the scope of proposed Section XI, Division 2, and subsequent planned additions, the proposed inspection requirements in Table B-14, below, are applicable.

Table B-14. In-service inspection requirements

System, Component, Part	Inspection Method	Area, Material To Be Inspected
Base and lateral core support structures		
Graphite structures	Visual	Exposed and accessible areas
	Material surveillance	Test specimens of floor blocks, posts, and seats, including side reflector components
Lateral restraint	Material surveillance	Test specimens of elevated-temperature and other structural metals
Thermal barriers	Visual	Exposed and accessible areas
	Material surveillance	Test specimens of elevated-temperature structural metals
		Test specimens of nonmetallic materials--fibrous blanket and ceramic block insulation, ceramic support pad
PCRV liner	Material surveillance	Test specimens of structural metal
Refueling penetration, shell and closure[a]		
Welds	Volumetric	
Bolting	Surface and volumetric	

[a]The refueling penetration forms the restraint preventing control rod ejection from the PCRV. Primary coolant boundary (Class 1) in-service inspection requirements apply to those penetration structures outboard of the penetration shear anchor assembly.

Question 12

Based on past licensing reviews for HTGRs, it is likely that seismic design requirements will restrict siting choices to locations of relatively low ground accelerations in comparison with those acceptable for LWRs. Discuss this siting flexibility limitation from the standpoint of environmental and cost-benefit considerations in comparison with the other NASAP reactors.

Response

The ground acceleration level chosen for the base HTGR design is presently 0.15/0.30 g (OBE/SSE) for a general range of soil sites. This is a moderate level and encompasses most of the present and proposed U.S. sites. In fact, this includes all sites east of the Mississippi River. With certain soils and construction design, the base HTGR design can be used for even higher g levels (see Question 4) and can also encompass the majority of western sites, except coastal.

It is not considered economically sound to force a more expensive HTGR design in order to take in the few remaining possible high seismic sites that may not be used in the future for any reactor plant. Regardless of whether the reactor is water cooled or gas cooled, the high seismic plant and components would then have to be specially designed for the more stringent requirements. There is nothing inherent in the HTGR which precludes such a high seismic design. It does not, therefore, seem that there would be any particular problem of siting flexibility.

Question 13

What additional information with respect to occupational exposure can be made available beyond that provided in the PSEID relative to LWRs and the other NASAP reactor designs? Consider normal operation, refueling, inspection, and decommissioning requirements.

Response

Preliminary occupational dose assessments associated with a 3,000-MWt gas-turbine HTGR unit are summarized in Table B-15. The expected total occupational

Table B-15. Expected annual man-rem exposures for a 3,000-MWt gas-turbine HTGR unit

Type of operation	Annual exposure (man-rem)
Reactor operation	7
Refueling	8
Reactor turbine system maintenance and in-service inspection	10
Balance-of-plant maintenance (assumed)	25
Reactor turbine system special maintenance	2
Total	52

exposure for the gas-turbine HTGR amounts to 52 man-rem per unit per year, as compared with 400-1,000 man-rem (Ref. 6) actually experienced at LWR plants. Initial operating experience at the Fort St. Vrain HTGR plant has resulted in less than 1 man-rem exposure per year and confirms that man-rem exposures for the HTGR are indeed lower than those for LWRs with similar rated powers. No assessment has been performed of occupational doses in connection with decommissioning.

Question 14

Past experience and Sections 2.4 and 2.5 of the PSEID illustrate the point that the more we know about the conceptual design of a reactor, the more issues we are able to define for resolution. The gas-turbine HTGR concept will likely undergo significant evolution before design details become firm. By that time, more detailed safety programs may also be defined. In spite of these difficulties, costs in time and dollars should be estimated for the resolution of design and safety issues. In responding to this question, we recommend that tables of a format similar to Table 2-17, Chapter 2, be included with expansions that compare research and development requirements, costs, and schedules of the reference gas-turbine HTGR, promising alternatives, and a base case for the 900-MWe steam-cycle plant.

Response

Table B-16 provides the technological advance requirements for gas-turbine HTGR plant components. The detail design and development costs for the steam-cycle and gas-turbine HTGR plants are currently being reevaluated, but the present cost ranges are \$250 to \$350 and \$400 to \$550 million, respectively. Table B-17 provides a breakdown of the costs for the steam cycle based on the 3,360-MWt lead plant as of February 1979.

Figure B-4 shows a program schedule for the 900-MWe steam cycle plant as of early 1978. With the shift of funding emphasis in FY-79 to the gas-turbine HTGR, the schedule shown would slip about 1 year if the program were reestablished with appropriate funding in FY-80.

A program milestone schedule for the gas-turbine HTGR demonstration plant is shown in Figure B-5.

Table B-16. Technological advance requirements

Plant Components

Concept: Gas-Turbine HTGR Demonstration Plant design 800 MWe (for MEU cycle)

	No new knowledge required	Contemporary technology with modified configuration/application	Modest improvement in performance or size from present knowledge	Modest improvement in performance or size from present knowledge	Major improvement in performance or size from present knowledge	Major improvement in performance or size and modified configuration/application	New technology (e.g., materials) required to meet system requirements	New technology and modified design required	Entirely new concept requiring new technology and new design
Nuclear fuel		X	(X)						
Plant control systems		X							
Reactor vessel		X							
Core support structure				X		X			
Reactor vessel internals including shielding, control rod guides, etc.			X						
Auxiliary systems	X								
Primary coolant chemistry/ radiochemistry control		X	(X)						
Primary-system heat exchangers		X							
Thermal-barrier system						X			
Emergency core cooling/ safe-shutdown systems			X						
Containment, containment-cleanup systems and effluent-control systems		X							
Other accident-mitigating systems, i.e., plant-protection system			X						
On-site fuel-handling storage/shipping equipment		X							
Main helium gas turbine						X			
Other critical components, i.e., control valves, hot duct						X			
Balance-of-plant components	X								

Table B-17. Steam-cycle HTGR design and development costs (millions of dollars)

	Design	Development
Steam generators and core auxiliary heat exchangers	26.9	9.4
Instrumentation and control	11.8	1.7
PCRV, liners, closures, penetrations	7.6	1.8
Reactor internals and thermal barrier	12.0	11.3
Safety and reliability	9.0	5.2
Systems engineering	24.6	10.2
Component and structural materials		14.5
Fuel development (medium-enriched uranium)		25.4
Graphite development		8.4
Fission product/coolant chemistry		6.1
Fresh fuel process development		12.0
Remainder of component tasks	45.9	45.4
	137.8	151.4
Total	289.2	
Engineering support	18.6	
Total	307.8	

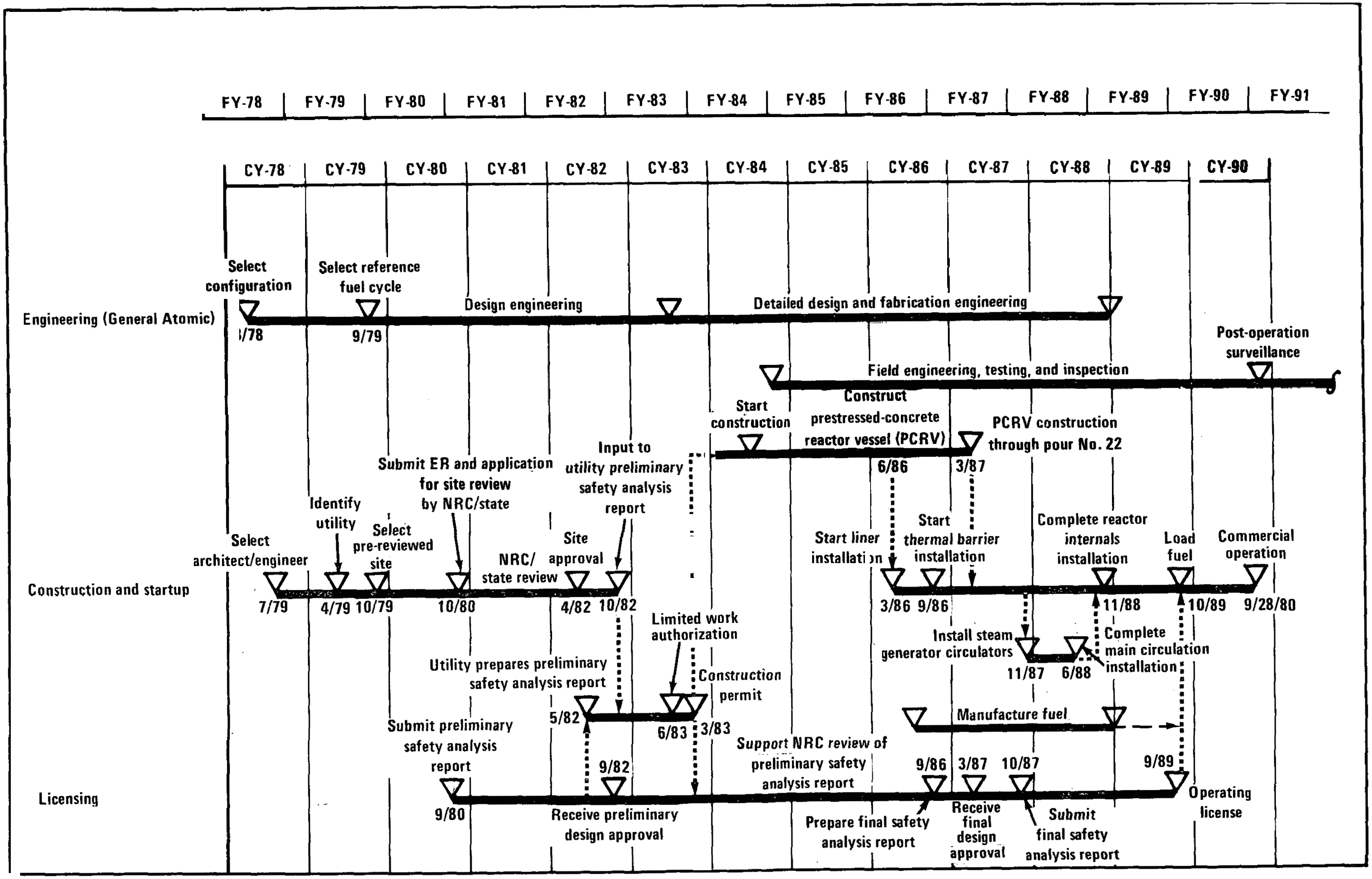


Figure B-4. 900-MWe steam-cycle HTGR summary program schedule.

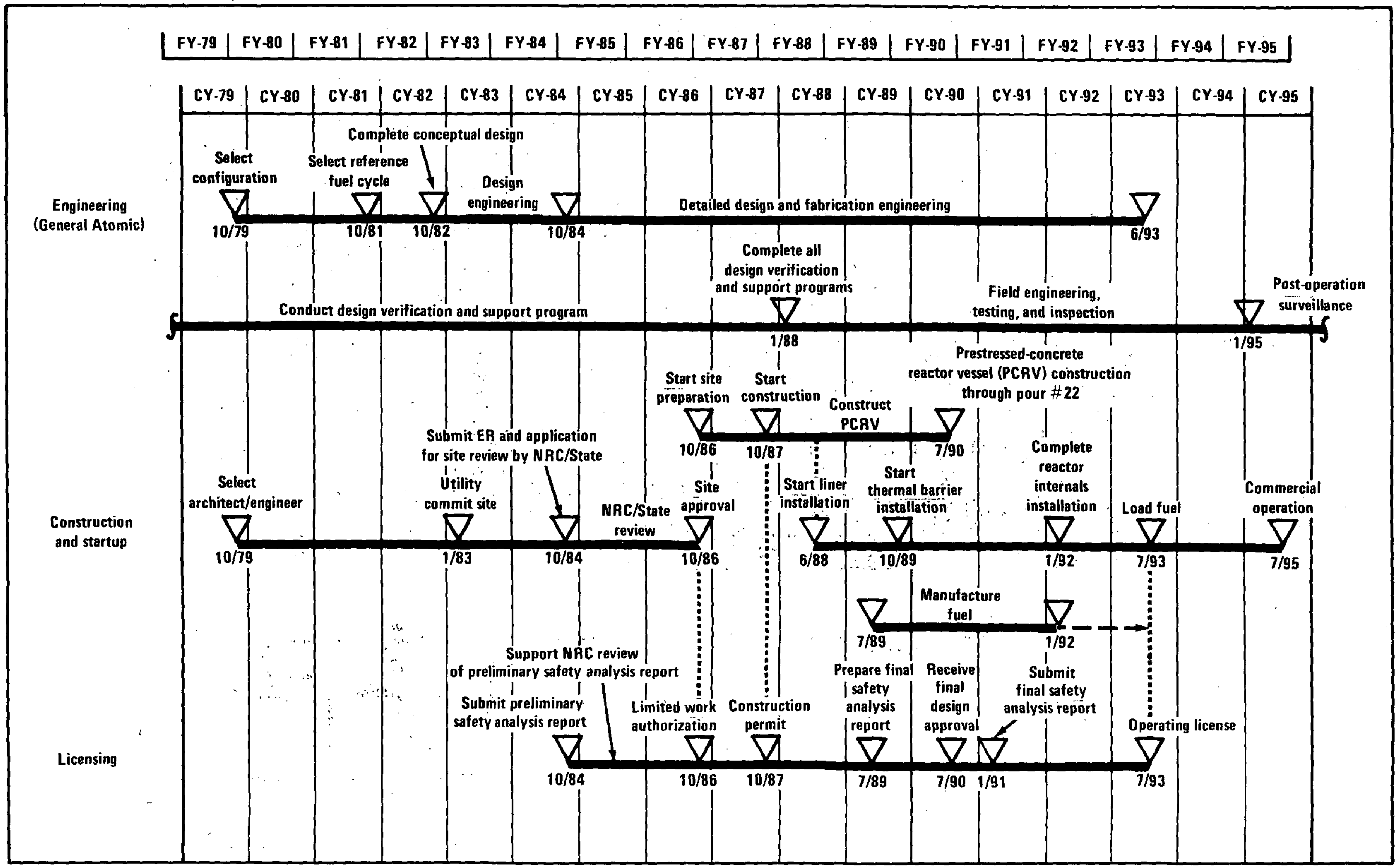


Figure B-5. Gas-turbine HTGR two-loop plant summary program schedule.

References

1. Gas Turbine Preliminary Safety Information Document, GA-13407, General Atomic Company, June 1975.

2. Gas Turbine HTGR Program, GA-A13950, Appendix B, Semiannual Progress Report for the Period January 1, 1976 through June 30, 1976, General Atomic Company, July 30, 1976.

3. HTGR Accident Initiation and Progression Analysis Status Report--Phase II Risk Assessment, GA-A15000, General Atomic Company, April 1978.

4. HTGR Accident Initiation and Progression Analysis Status Report, GA-A13617, General Atomic Company, Volumes I through VII, 1975-76, Vol. VIII, January 1977.

5. J. C. Scarborough et al., Gas Turbine HTGR: A Technology Assessment, NUS-3041, prepared for the Electric Power Research Institute, Palo Alto, California, October 1977.

6. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (ALARA)," U.S. Nuclear Regulatory Commission, March 1979.