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AN EVALUATION OF HIGH-TEMPERATURE GAS-COOLED REACTORS

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Prepared by

OAK RIDGE NATIONAL LABORATORY

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DISCLAIMER

FOREWORD

In implementing the civilian nuclear power research and development program, specific advanced converter concepts have been assessed in detail. This report, "An Evaluation of High Temperature Gas Cooled Reactors," was prepared by an Oak Ridge National Laboratory working group under the direction of the Division of Reactor Development and Technology, U.S. Atomic Energy Commission, and is part of an overall assessment of the Civilian Nuclear Power Program being made in response to a request in 1966 by the Joint Committee on Atomic Energy. As outlined in the 1967 Supplement of the 1962 Report to the President on Civilian Nuclear Power, changes since 1962 in the technical, economic, and resources picture have necessitated further study of the AEC program. This report evaluates one family of reactors in the reactor development program of the USAEC and provides a basis for their consideration in the assessment of the overall program.

The overall assessment will include: the technical status and economic potential of advanced converters and breeders; the role of thorium; reactor fuel cycles; and a system analysis of the future nuclear-electric power complex. Although all phases of the assessment effort are based on one set of ground rules to achieve a common basis for comparison, it is inevitable that when the reports are published, many changes now taking place will not be reflected. The differences will be due to efforts to consolidate and strengthen the reactor development programs and to the rapid expansion of the nuclear power industry. These matters are discussed in more detail in the 1967 Supplement to the 1962 Report to the President on Civilian Nuclear Power.

This evaluation was conducted by the "Advanced Converter Task Force" whose members include representatives of Babcock and Wilcox (B&W), Gulf General Atomic (GGA), Atomics International (AI) — Combustion Engineering (CE) (Joint venture group), Oak Ridge National Laboratory (ORNL), Pacific Northwest Laboratory (PNL), Los Alamos Scientific Laboratory (LASL), Brookhaven National Laboratory, and the Division of Reactor Development and Technology (RDT) of the U.S. Atomic Energy Commission. The evaluation is based largely upon designs and information provided by GGA.

The procedure followed in the study and preparation of this report was

1. presentation or refinement of designs and data by the concept sponsor, Gulf General Atomic,
2. technical and economic assessment of the concept and preparation of draft report by the ORNL working group,
3. review of draft report by design sponsors, task force members, and the Division of Reactor Development and Technology staff,
4. preparation of the final report by ORNL, taking into consideration comments of reviewers,
5. approval of final report by Advanced Converters Task Force, and
6. review of the final report by the Commission.

During this process, wherever possible, data applicable to the systems analysis study were extracted. Finally, before publishing, the report was reviewed by selected representatives of the reactor industry, national laboratories, and the U.S. Atomic Energy Commission. There has been general acceptance of the content, and all comments received have been considered in the final version of the report.

As discussed in the 1967 Supplement in the 1962 Report to the President on Civilian Nuclear Power, a large effort has been required to develop the light-water reactors — starting with naval reactors and followed by the early civilian power experiments and demonstrations, which culminated in major engineering efforts to construct large central station plants. The wide-spread acceptance of the light-water reactor is an established fact. The large industrial commitments and improvements in technology should result in further improvements in performance. These factors will make difficult the introduction in the United States of any new system, even though a potential economic gain is indicated. Further, the continued improvement in the industrial posture of the light-water reactors plus the urgent need to introduce breeder reactors at the earliest date possible will narrow the time span in which advanced converter reactors could be successfully introduced. The possible future role of such reactors in the U.S. nuclear power economy is, therefore, not yet clear. However, a significant role still exists for the economic advanced converter as a more efficient user of our uranium reserves and thorium than existing thermal reactors. This would assume increasing importance in the event of delays in availability of economic fast breeders.

If the assessment of the high-temperature gas-cooled reactors and other advanced converter reactors indicates that a specific reactor concept appears to have an attractive economic potential, such a conclusion could provide a basis for further development. The basis for development, however, would depend upon additional factors, such as the interest of U.S. utilities in high-temperature gas-cooled reactors, and the ability of the U.S. Government, industry, and utilities to support parallel reactor development.

In large measure, the design evaluated in this report is based on information provided by the developer of the systems and, therefore, generally reflects his viewpoint. As fossil fuel was a moving target for the light-water reactors, so now light-water reactors, as well as fossil fuel, will be the moving targets for advanced converters such as the high-temperature gas-cooled reactor. This report presents an evaluation of high-temperature gas-cooled reactor technology that the proponents of High-Temperature Gas-Cooled Reactor believe can be achieved.

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Division of Reactor Development
and Technology

PREFACE

High-temperature gas-cooled reactors for production of commercial power possess the potentials of better fuel utilization and significantly higher net thermal efficiency than either the light-water or heavy-water reactors. The first gas-cooled reactor to operate at a significant power level was the X-10 pile at the Oak Ridge National Laboratory. This reactor was started up in the early 1940's, and it operated for above 20 years. The reactor was cooled with air and fueled with aluminum-clad uranium-metal slugs.

The first gas-cooled power reactor was built at Calder Hall in Great Britain and achieved full-power operation in the fall of 1956. In Europe the gas-cooled natural-uranium power reactor has dominated all other types and the worldwide total installed capacity of gas-cooled reactor plants now exceeds all other types of power reactors combined. The Calder Hall plant served as a prototype for the Magnox reactor plants built by the British in their nuclear power program. The French have also concentrated on the development of gas-cooled power reactors. Development work has continued in both these countries toward advanced versions of gas-cooled reactors. The British have also built and operated the Advanced Gas-Cooled (AGR), and the French are continuing the development of the Magnox reactor and are working on a gas-cooled heavy-water reactor (EL-4).

The European Nuclear Energy Agency, Organization for Economic Cooperation and Development, sponsored work in the Dragon project to develop a 20-Mw(t) high-temperature gas-cooled reactor experiment (HTGCR), which was completed in 1965. This reactor has a semihomogeneous graphite-moderated core based on the $^{235}U/^{232}Th/^{233}U$ fuel cycle. A variation of the high-temperature reactor concept, the Pebble-Bed Reactor, is being developed by the Brown-Boveri-Krupp organization in Germany. The first reactor plant based on this principle is operating at Julich, Germany at 15 Mw(e).

Interest in gas-cooled power reactors in the United States dates from the design study project of the Daniels reactor in 1945. Though this plant was never built, many of its design features are similar to those of contemporary advanced gas-cooled reactors. Beginning in 1956, development work at Oak Ridge National Laboratory and at General Atomic Division of

General Dynamics Corporation led to the construction of the 30-Mw(e) Experimental Gas-Cooled Reactor (EGCR) and the 40-Mw(e) Peach Bottom HTGR. Peach Bottom was completed in 1965 and went into commercial operation early in June 1967. The EGCR was similar to the British AGR, although cooled by helium. Due to technical and programmatic difficulties, however, construction was never completed. The HTGR operates at higher temperatures than EGCR and uses an all-ceramic fuel without metallic cladding.

Development of larger HTGR plants was begun in 1965 as part of the advanced converter program. As part of this program, a 330-Mw(e) HTGR plant (Fort St. Vrain) is under construction for the Public Service Company of Colorado. Commercial operation is expected in 1972-1973.

The HTGR development in the U.S. has been a cooperative effort by Government, utilities, and industry. To date, Gulf General Atomic, Inc., has worked both on Peach Bottom and the large HTGR reactors in cooperation with utilities and the USAEC.

An overall comparison of high-temperature gas-cooled reactors and other advanced converters is covered in a separate topical report (WASH 1087).

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1. INTRODUCTION AND SUMMARY

This report presents an evaluation of high-temperature gas-cooled reactors (HTGR's) based on two 1000-Mw(e) designs prepared and submitted by Gulf General Atomic. These two designs, a reference design and a backup design, were prepared by GGA during the latter part of 1966 under the ground rules specified by the AEC Task Force on Advanced Converters, and therefore the evolutionary design changes subsequently incorporated in the HTGR design are not included here.

1.1 Reactor Plants Studied

The HTGR is basically a graphite-moderated helium-cooled reactor in which the graphite serves as moderator, reflector, and core structure. The core and helium circulation system are housed in a prestressed-concrete pressure vessel. The fuel cycle normally includes thorium as the fertile material. The bred ^{233}U is recycled to provide fissile material, and uranium fully enriched in ^{235}U is added as required for makeup.

The backup design developed by GGA in 1966 was intended to reflect the state of HTGR technology being developed for the Fort St. Vrain plant. It embodies the same features as the Fort St. Vrain Nuclear Power Station, for which active research, development, and detailed design are now under way. The plant is scheduled to operate on the grid of the Public Service Company of Colorado in the 1972-1973 period. The Fort St. Vrain plant incorporates many components and features considered to be developmental.

The HTGR technology incorporated in the Fort St. Vrain plant will be demonstrated first in the preoperational tests planned for some components and in operation of the plant. Since the backup design is a 1000-Mw(e) single-reactor station, whereas the Fort St. Vrain station will be only 330 Mw(e), the research and development program required for the backup design reflects, in large part, the scaleup in size to 1000 Mw(e) based on successful development of the Fort St. Vrain reactor, together with demonstration of reliability and adequate provision of access for inspection and maintenance for the Fort St. Vrain reactor.

The reference design is intended to represent the more favorable performance that should be possible in an HTGR after development of the Fort St. Vrain plant and the further development and scaleup required for the backup plant design. The principal differences between the backup and reference designs are the following:

1. Radial-flow steam generators are used rather than axial-flow units to permit a more compact arrangement and lower pressure drop.

2. A more efficient steam cycle is utilized.

3. Wirewrap instead of circumferential tendons are used in the construction of the prestressed-concrete reactor vessel.

4. On-line instead of off-line refueling is used, and the conversion ratio and specific power are thereby increased. Availability may be increased, but no credit for increased availability was given in this evaluation.

1.2 General Study Objectives and Ground Rules

In performing the evaluation, independent calculations of reactor performance were made insofar as time and manpower limitations permitted. For those areas in which complete design calculations were not made, the methods and data used by GGA in establishing the design were carefully reviewed.

The overall evaluation criterion employed was based on the power-generation cost under Advanced Converter Task Force ground rules. The entire power plant was considered. The power cost was obtained from the combined estimates of capital, operating, and fuel-cycle costs. The technological evaluation was made on the bases that the reactor design is feasible and that demonstration of the developmental components and design features of the Fort St. Vrain plant on the currently projected schedule will be successful. This latter criterion implies that although engineering development may be required to make the design practicable, no technological breakthrough is required. The difference in power cost between the reference and backup designs is a reflection of the incentive for successful completion of the portions of the research and development program beyond those required for the Fort St. Vrain plant and the backup design that relate specifically to the reference design.

The ground rules for the study are given in more detail in Appenxix A. Some of the more important provisions follow:

1. The technology is based on that considered feasible today, and successful demonstration of all developmental features of the Fort St. Vrain plant is assumed.

2. The fuel-fabrication and processing plants are privately owned.

3. The total electrical capacity of the reactor plant is 1000 Mw(e); however, the fuel-fabrication and processing plants are sized to handle fuel from 15,000-Mw(e) capacity of a given reactor type.

4. The fuel-cycle cost is based on present-value accounting in which the reactor behavior is averaged over a 30-year period.

5. The bred uranium from the thorium-uranium fueled core is recycled throughout the reactor lifetime. Any additional fissile material required (including the entire initial loading) is supplied in the form of uranium fully enriched in ^{235}U.

1.3 Summary of Results

The results of this evaluation indicate that both the reference and backup HTGR design concepts represent feasible extrapolations of technology beyond that incorporated in the Peach Bottom and Fort St. Vrain reactors. Preliminary demonstration of the short-term operating behavior of graphite fuel in helium-cooled systems has been given by the operation of the Peach Bottom and Dragon reactors. Investigation of the developmental components and systems and features related to reliability, maintainability and economics will result from the construction and operation of the Fort St. Vrain reactor.

Based on the economic ground rules of the Advanced Converter Task Force, power costs of 3.7 mills/kwhr(e) for the HTGR reference design and 4.0 mills/kwhr(e) for the backup design were found. These cost estimates are subject to uncertainties in a number of respects discussed in later chapters. The backup design consists of a scaleup from the Fort St. Vrain technology currently under development, while for the reference design, successful completion of certain additional development programs is assumed.

The difference in power costs between the two designs represents the economic incentive for pursuit of these development programs.

Data and information on the operational characteristics of many features of coated particles bonded in a carbonaceous matrix are available from operation of the Peach Bottom and Dragon reactors and from irradiation tests. Data and information are lacking on performance of full-scale Fort St. Vrain-type fuel in a high fast-neutron flux. The irradiation performance of the fuel bonding presently specified requires confirmation. After this report was prepared, the fuel for the Fort St. Vrain plant was modified to include an SiC layer on the coated particles. The performance of a fuel consisting of a fuel-particle kernel covered by a porous-carbon buffer layer and a near-term isotropic pyrocarbon layer, and referred to as a "Biso" particle, was evaluated in this review.

Maximum fuel temperatures in both the backup and reference designs were found to be below the 1500°C value specified by GGA, even though extremely conservative values for thermal conductivity and for the extent of coolant flow shunting were used. If the fuel particles are to be operated at 1500°C for the design burnup, calculations indicate a maximum tangential strain of 1.6%, whereas particle coating failures would occur with tangential strains of greater than 5%. It is concluded that there is a design margin in the fuel performance that would probably allow higher power density, larger fuel particles, or thinner particle coatings in future designs. It may also be possible in future designs to increase the maximum fuel temperature above 1500°C, since such temperature increases would merely result in a gradual increase in fission-product release.

The designs evaluated depend for their economic success on the reprocessing and recycle of bred ^{233}U. The flow sheets for the remote recycle and refabrication process are based on reasonable extrapolations, as are the associated cost estimates; however, there has been no demonstration of the complete fuel cycle, either on a production-plant or a pilot-plant scale, and thus there remains some uncertainty in fuel-reprocessing cost estimates. [An uncertainty in fabrication and processing cost estimates of 50% would result in a power cost increase of 0.2 mill/kwhr(e).]

Recent irradiations of graphite at high temperatures (near 1000°C) indicate a pattern of shrinkage up to fast-neutron doses of about 5×10^{21} neutrons/cm^2, followed by very rapid growth. Since the maximum fast-neutron exposure in the HTGR is greater than 5×10^{21}, there could be problems such as cracking of the graphite and/or jamming of the blocks in the core, but no penalty was assigned to the concept because the available irradiation data are fragmentary and it is not clear that a problem exists. Also, graphite temperatures near 1000°C will occur only in fresh fuel elements. If necessary the situation can be improved by one or more of a variety of steps, such as developing a graphite with relatively better irradiation stability, reducing the maximum graphite temperatures slightly, designing the blocks to permit some swelling, or reducing fuel element exposures.

Reasonable agreement was obtained with GGA physics calculations for both designs. In each case slightly higher fissile inventories and slightly lower conversion ratios were found than calculated by GGA. The differences, which are apparently attributable to the cross sections, make fuel-cycle cost estimates higher than those of GGA by not more than 0.05 mill/kwhr(e).

The prestressed-concrete reactor vessel (PCRV) requires a leaktight liner and a system for controlling the temperature of the insulation and liner that will operate reliably over the 30-year plant life. This is a major area that requires development and demonstration for the Fort St. Vrain reactor. In the backup design the PCRV is not very different from others that have been built. In the reference design the use of wire-wound prestressing will require the development of new equipment and techniques for performing the wire-winding, controlling the tension during installation, and subsequent monitoring of the tension. An increasing body of evidence supports the position that a properly designed and constructed PCRV cannot fail catastrophically, because any crack formed will not propagate. Both designs will require careful attention to questions of inspection and maintenance of components internal to the PCRV.

Calculations of steam generator performance for both designs are in substantial agreement with those of GGA. There are, however, large uncertainties in the correlation of heat transfer coefficients for helium flowing across banks of tubes that will need to be resolved by experiment before the radial-flow steam generator for the reference design can be specified. The implications of a scaleup factor of 6 are noted.

The provisions for replacement of a steam generator require remote equipment to remove and install the steam generator units inside a PCRV. Such equipment is to be developed for the Fort St. Vrain plant. Again the sixfold increase in the size of steam-generator units for the backup design could require further development of the remote-handling equipment. The first test at operating conditions of this type of steam generator will be obtained by operation of the Fort St. Vrain plant.

The helium circulators are unique in that the compressor, steam turbine, and auxiliary water turbine are located on a single shaft. The Fort St. Vrain circulator development and test programs are directed toward many of the development problems that apply to the 1000-Mw(e) designs. The long-term endurance testing of this circulator concept will first be accomplished during sustained operation of the Fort St. Vrain plant during the years following startup in 1972-1973. There may be additional problems in the 1000-Mw(e) system, however, particularly with regard to loop isolation, system interactions, and control during emergency operation. Reliability and maintainability of the circulator units are important factors to be considered.

The backup design has two fuel-handling machines of the same design as those for Fort St. Vrain. The Fort St. Vrain fuel-handling machine is scheduled for demonstration later in the program. The refueling time estimates that propose a 90% plant availability have little or no allowance for difficulties or malfunctions. The reference design has an on-line refueling machine that requires the development of a curved chute and grapple head assembly; this implies the need for solutions to possible problems of fuel wedging due to differential pressure. On-power refueling is different from previous experience in gas-cooled reactors, where there were discrete fuel channels to refuel; in this reactor, the entire core is to be refueled. The design and operating characteristics, as well as the cost estimate for the on-line refueling machine, are necessarily uncertain.

Control rods and drives are the same for both designs and are similar to those for Fort St. Vrain. These are still under development and have not yet achieved desired operation reliability. The HTGR's are inherently stable, slow moving, and easy to control. The system is load-following and can respond to large, rapid changes in load on the turbine.

The HTGR has several good safety characteristics. The high heat capacity, high-temperature capability, and good fission-product retention properties give long time margins for shutting the reactor down or restoring cooling in accident situations. The HTGR can tolerate complete loss of cooling (after a shutdown) for as long as 4 hr. It can also tolerate fairly rapid depressurization of the primary cooling system. There is some possibility that the HTGR may be able to tolerate the loss of all forced circulation of coolant indefinitely with the PCRV liner cooling system providing sufficient heat removal. If this can be demonstrated for the 1000-Mw(e) core, it will provide a backup to the emergency cooling provided by the normally operating heat-removal components.

The GGA design is unique in that it does not provide any specialized emergency cooling system. The acceptability of this approach will have to be established, and special attention must be given to the reliability of the main circulators and steam generators. The design has a number of attractive features in terms of primary component reliabilities.

A steam-generator leak and the possible steam-graphite reaction create a potential problem of corrosion of graphite and fuel. (The allowable water inleakage in the Fort St. Vrain plant is only 0.04 lb/hr.) It is necessary to exclude air from the core under all accident conditions, and the PCRV design assists in meeting this requirement.

The extent of the research and development work required for the backup design is based on successful completion of the Fort St. Vrain project. The principal problem is the scaleup of the 330-Mw(e) components to the size required for a 1000-Mw(e) plant. Considerable additional research and development work would be required for the reference design in working out the details of the wire-wound PCRV, the radial-flow steam generators, and the on-line refueling equipment.

Power operation of Fort St. Vrain is expected in 1972-1973. It has been estimated that the first installation of the backup design could be completed in 1974 and the first reference design in 1976. Commercial availability of the two designs [defined as the completion of the next four 1000-Mw(e) plants] was estimated by GGA to be in 1976 and 1978.*

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*This scheduling is considered by AEC to be optimistic based on previous experience with other projects, including gas-cooled reactors, and on the current construction experience with other large power reactor projects.

2. DESCRIPTION OF REACTORS

The designs reviewed in this report were developed by GGA and are described in their reports.[1,2] The design with the more conservative characteristics is called the backup design and is intended to represent a size extrapolation of the Fort St. Vrain plant, which is now under development. The other design is known as the reference design and is conceived as a reasonable extrapolation of performance characteristics achievable after further research and development work beyond that required for the Fort St. Vrain plant and the work required to extrapolate that technology to 1000-Mw(e) size. Some principal characteristics of the two designs are given in Table 2.1 and compared with those of the Fort St. Vrain reactor. More detailed descriptions of individual systems are given in Chapters 3 through 6.

In both designs the reactor coolant is helium at a pressure of 700 psia. The reactor is graphite moderated and reflected and has thorium in the fuel cycle. The two typical types of fuel element for the backup design are shown in Fig. 2.1. These elements are stacked together vertically and horizontally to form the entire core, so no separate moderator or structure is required. In the backup design a fuel column is six blocks high (174 in.). In the horizontal direction the core is made up of "patches," each of which consists of a single control rod element surrounded by six elements not containing control rods. The core consists of 91 of these patches. The fuel element and arrangement for the reference design are the same as for the backup design, except that the element is only 15.6 in. high (instead of 31.2 in.) and the fuel column is 12 blocks high (instead of 6). A graphite reflector surrounds the reactor core on all sides. The inner row of side reflector blocks, all top reflector blocks, and one layer of bottom reflector blocks are removable through the fuel transfer machine.

Within a fuel element the fuel is contained in particles of UC_2 (or UO_2) coated with two layers of pyrolytic carbon and then bonded with low-density carbon into "fuel sticks" about 0.45 in. in diameter. The fuel particles are of two types: a fertile particle containing the thorium and the recycled uranium (mostly ^{233}U) and a fissile particle containing

Table 2.1. Summary of HTGR Characteristics Specified by GA

	Fort St. Vrain Design	Backup Design	Reference Design
Reactor power			
Core nuclear power, Mw(th)	841.7	2457	2318
Net electrical power, Mw	330	1000	1000
Net thermodynamic efficiency, %	39.2	40.7	43.1
Coolant			
Composition	Helium	Helium	Helium
Core inlet pressure, psia	700	700	700
System pressure drop, psi	14	11	7
Core pressure drop, psi	8.3	7.6	2.7
Flow rate, lb/hr	3.41×10^6	10.27×10^6	9.28×10^6
Core inlet temperature, °F	758	758	803
Mean core outlet temperature, °F	1449	1449	1524
Number of coolant loops	2	3	3
Coolant inventory, lb		23,540	15,640
Core thermal performance			
Maximum fuel temperature, °F	2217	2506	2632
Peak-to-average power ratio	2.67	2.67	2.05
Average core power density, kw/liter	6.3	7.88	7.43
Average specific power, kw/kg of fertile material	41.9	60	55.6
Description of core and fuel			
Moderator material	Graphite	Graphite	Graphite
Reflector material	Graphite	Graphite	Graphite
Fuel material	UC_2 and ThC_2	UO_2 and ThO_2 or UC_2 and ThC_2	UO_2 and ThO_2 or UC_2 and ThC_2
Equivalent diameter of core, ft	19.5	31:1	31.1
Active height of core, ft	15.6	14.5	14.5
Number of fuel elements	1482	3841	7591
Fuel element dimension across flats, in.	14.2	14.17	14.17
Fuel element height, in.	31.2	31.2	15.61
Fuel cladding	None	None	None
Core structure	None	None	None
Description of fuel cycle			
Fuel loading scheme	Six-batch scatter	Four-batch scatter	Continuous
Fuel recycle scheme		Bred-uranium	Bred-uranium
In-core residence time at 0.8 load factor, years	6	4	4
Average fuel exposure, Mwd/MT	100,000	70,000	65,000
Control system			
Type	Rod	Rod	Rod
Number	74	182	182
Material	30 wt % boron	30 wt % boron	30 wt % boron
Reserve system	Poison granules	Poison granules	Poison granules
Reactor vessel			
Type	Hexagonal	Hexagonal	Cylindrical
Material	Prestressed concrete	Prestressed concrete	Wire-wound prestressed concrete
Inside diameter, ft	31.0	47.8	43.5
Inside height, ft	75.0	88.5	79
Turbine plant			
Throttle steam temperature, °F	1000	1000	1050
Throttle steam pressure, psig	2400	2400	3500
Feedwater return temperature, °F	403	410	510
Arrangement of turbine-generators	Tandem-compound, double-flow	Tandem-compound, six-flow	Tandem-compound, six-flow

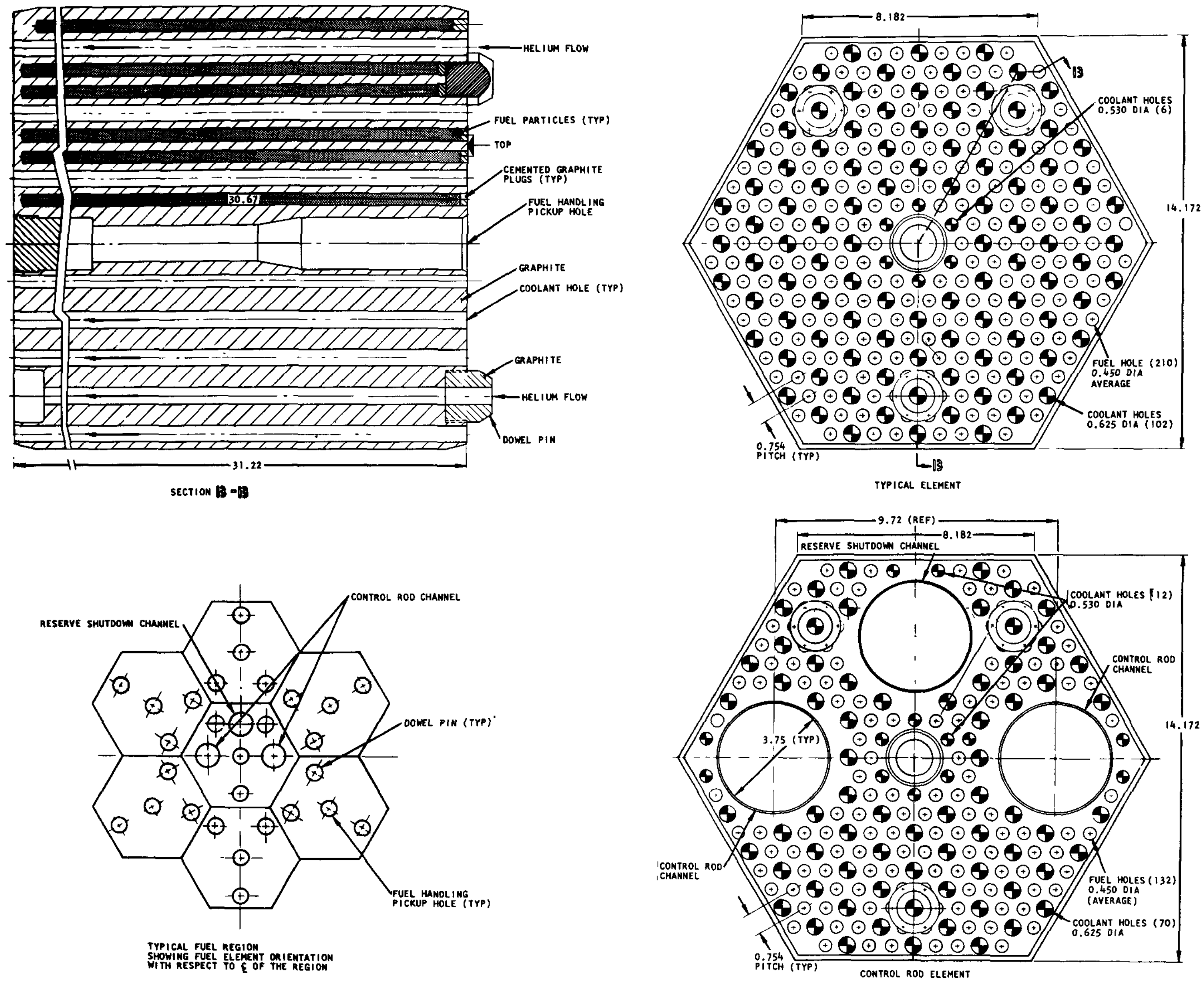

Fig. 2.1. HTGR Fuel Element. (GGA illustration)

the makeup uranium, fully enriched in ^{235}U. The fertile particle has a kernel diameter of 350 μ and a coating thickness of 130 μ, while the fissile particle has a kernel diameter of 150 μ and a coating thickness of 150 μ. The use of two different particles permits separation of the particle types during reprocessing so that the fissile particle, with its high ^{236}U content, can be separated and thus not included in the material to be recycled.

The entire primary cooling system, including the steam generators, the main helium circulators, and the reactor core, is housed within a prestressed-concrete pressure vessel (PCRV). The enclosure of the entire primary system within the PCRV is based on the maintenance concept of shutting down the entire plant and removing from the PCRV any component requiring maintenance. In both designs the cool helium from the circulator discharge flows upward in the annular space between the core barrel and the inner surface of the PCRV liner into the plenum above the reactor core. The helium is then heated during downward flow through the core by passing through coolant passages in the core blocks. After passing through the core the helium is directed to the steam generators, where heat is transferred to the secondary (steam) coolant. The cooled helium then flows to the circulator inlets.

The prestressed concrete serves in the PCRV as the structure to contain the primary coolant pressure, and a steel liner serves as a gastight membrane. Nearly absolute leaktightness of the steel membrane is required and must be maintained for the plant's 30-year life without provisions for maintenance. Thermal protection for the vessel is provided by multiple-sheet steel insulation inside the liner and by water cooling on the outside of the liner. (The Fort St. Vrain plant has recently adopted the use of fibrous insulation.) This thermal protection and temperature control system is vital to the plant, and most components must have sufficient reliability to operate 30 years without provisions for maintenance. For the backup design the prestressing is accomplished by a system of longitudinal, crosshead, and circumferential posttensioned steel tendons. For the reference design the prestressing is accomplished by a system of longitudinally posttensioned steel tendons and by circumferentially wrapped wire on the outside of the vessel wall.

Refueling is done through 91 refueling nozzles, which also house the control rod drives. The fuel-handling machine is positioned over a refueling nozzle and thus serves as part of the containment during refueling. The backup design requires an annual shutdown for refueling and operates on a four-year refueling cycle. The reference design employs on-line refueling, which is carried out continuously with the reactor at full power.

There are six steam-generator modules (two per loop), and each module consists of three tube bundles, a reheater section, a superheater section, and an economizer-evaporator section. In the backup design the tubes are arranged in a series of concentric helical coils with a vertical axis. The helium flows downward across a matrix of in-line tubes. The tube bundle design is a scaled-up version of that of Fort St. Vrain. In the reference design the helium flow enters the module from the top and flows radially outward through the annular tube bundles.

Helium is circulated through each of the three primary loops by two single-stage axial-flow compressors. Each compressor is normally driven by a single-stage steam turbine mounted on the same shaft.

References

1. General Atomic, Backup Design of Twin 1000-Mw(e) HTGR Station, Jan. 26, 1967. (Not documented)

2. General Atomic, Reference Design of Twin 1000-Mw(e) HTGR Station, Jan. 26, 1967. (Not documented)

3. PHYSICS EVALUATION

In both the backup and reference designs the initial operation of the reactor is normally with a fuel using thorium as fertile material and uranium fully enriched in ^{235}U as fissile material. Bred uranium is recovered from the fuel elements removed from the core. This bred uranium is recycled to the reactor after a suitable delay for reprocessing (assumed to be 1.0 year in the calculations), along with sufficient additional uranium fully enriched in ^{235}U to maintain the desired reactivity. In the typical equilibrium cycle the fuel elements are supplied with a mixture of two types of coated particles: makeup particles containing only uranium fully enriched in ^{235}U and fertile particles containing the thorium and all the recycled uranium. Thus bred material is separated from the exposed makeup material for recycle, and the makeup material is discharged (sold) to avoid the penalty from high ^{236}U content.

Calculations of this core concept by ORNL were made independently from those of GGA to provide maximum assurance of correctness of results. Expectedly, the use of different cross sections and calculational methods led to somewhat different results. Although GGA codes were used in some calculations, they were extensively modified. The fine-group microscopic cross-section libraries used evolved in reactor analysis work at ORNL.[1]

Spectrum-cell calculations were made with the code TONG.[2] This code is a coupling of the GGA above-thermal-spectrum code GAM-I,[3] which contains the NIT[4] resonance treatment, with the Brookhaven National Laboratory's one-dimensional integral transport-theory thermal code THERMOS.[5] Dancoff factors were obtained with the first-collision Monte Carlo code RAFFLE.[6] Fifteen broad-energy-group cross sections were calculated with TONG for a composition expected to be near average. Then one-space-point multizone depletion calculations were done with code TONGER[7] for the reactor history, with the approximation that all material is exposed to a single neutron flux at any time, as determined from the average core contents.

For the backup design with batch refueling of one-fourth of the core each time, the multiplication factor was held at unity by adjustment of the density of a smeared absorber nuclide to represent the B_4C control rod. Reactivity lifetime was established as that time when the density of the control rod absorber nuclide became zero. Discharge and makeup requirements were calculated, with account being taken of recycle of the uranium from the fertile particles. Depletion of the material in the makeup fuel particles was determined separately from that in the particles containing recycle fissile material and thorium.

The reference design requires depletion of an initial core and then continuous fueling. The initial core was exposed until the control absorber was completely removed. Then continuous refueling was approximated by a multizone batch-refueling calculational model. Removals from the core were made periodically, and recycle material was added. Reactivity was maintained over an exposure period by adjusting the amount of makeup material (fully enriched uranium) in the zone last fueled. Thus the system was kept critical without control rods. After a specified exposure period, the cumulative amount of makeup feed was determined for the period. The calculation was made for 24 zones corresponding to a refueling schedule of two months real operating time at 0.8 load factor. This should be an excellent approximation to continuous refueling. The calculations were for once-through exposure of makeup material and full recycle of bred material, with a one-year delay in recycle.

Thermal self-shielding effects within a particle were neglected. These effects have been estimated by GGA to be much less than 0.01 $\Delta k/k$.[8]

3.1 Dancoff Factor and Effective Thorium-Resonance Integral

Calculations were made for full fuel elements to determine first-collision probabilities. Defining C as the Dancoff factor, consistent with the NIT treatment, the Monte Carlo results gave:

	C, Dancoff Factor
Noncontrol-rod fuel cell	0.357 ± 0.009
Rod out, rod fuel cell	0.330 ± 0.005
Rod in, rod fuel cell	0.323 ± 0.006
System average, rod out	0.353

The bounds on C represent the 95% confidence level, as indicated by the histories. GGA used a value of C of about 0.4.

The resonance parameters for ^{232}Th used in this study were selected by GGA. They produce an effective shielded resonance integral somewhat lower than that produced by the parameters used in the past by ORNL. This more recently evaluated integral is believed to be as reliable as any available. Although there is uncertainty in the calculation of the ^{232}Th reaction rate (perhaps ±5%), it could be readily adjusted. The carbon-to-thorium ratio could be changed, and both the fuel hole size and distribution could be altered to obtain the required reaction rate.

An additional uncertainty comes from the approximations made in the NIT code in calculating broad-group thorium cross sections. The effects of these approximations are not accurately known but may introduce an uncertainty in thorium loading similar to that from the cross sections.

3.2 Graphite Scattering Kernel

The thermal-scattering kernel used for graphite was calculated with the Parks crystalline lattice model by employing a technique developed at GGA.

3.3 Cross Sections of ^{233}U

There remains some uncertainty in the ^{233}U cross sections. New measurements are becoming available (for instance the recent Weston measurements), but further evaluation of these will be necessary to insure reliability. ORNL believes that the cross sections used in the calculations represent a reasonable interpretation of the available experimental data. The recent analyses made by Drake at GGA, as well as recent measurements, were taken into account. The conversion ratio is quite sensitive to the eta of ^{233}U. Since the amount of bred ^{233}U available for recycle depends on the conversion ratio, and the eta of makeup ^{235}U is relatively low, there is a significant feedback effect. The uncertainty in conversion ratio of the HTGR concept due to uncertainty in the ^{233}U cross sections is believed to be within ±2%.

3.4 Temperature Coefficient of the Thorium

Increasing the temperature of the thorium from 1366 to 2366°K was calculated to change the multiplication factor by −2.7% and give a Doppler coefficient of reactivity of $-2.7 \times 10^{-5}/°C$ late in the core history. If only a fraction of the thorium experienced the temperature rise, the prompt part of the coefficient would be proportionately lower.

A primary concern regarding the temperature coefficient is the prompt response of the reactor to reactivity additions. If the fuel temperature were to increase without immediately increasing the temperature of the fertile material, the Doppler effect would not be available to give a prompt negative reactivity coefficient. Thus complete separation of fuel and fertile material, in general or locally, is undesirable. The initial loading of ^{235}U thus needs to be mixed with at least part of the ^{232}Th fuel. Mixing these materials introduces a penalty in conversion ratio in the first few cycles that was neglected both in ORNL calculations and those of GGA. The penalty results from the need to use some of the highly exposed ^{235}U, contaminated with ^{236}U, as fuel during the first few cycles.

3.5 Power-Density Distribution

ORNL did not make dimensional calculations to examine the distribution of power density. It has been ORNL's experience that

1. such calculations are quite sensitive to the distribution of materials, and hence a single calculation may not be very useful;
2. the concept has considerable flexibility regarding distribution of materials; if poor power-density distribution is associated with one arrangement, small changes in fuel distribution can be made to improve the situation; and
3. the peak power density at one location depends on the local temperature distributions, and a nuclear-thermodynamic analysis would be required that would be beyond the scope of this effort.

Calculations were made by GGA, and their results appear reasonable. However, since the power distribution is sensitive to small changes in absorption cross section, close attention will have to be given to fuel distribution (i.e., the radial zone-loading of the fuel) and control rod programming. Further studies of the behavior of this core with depletion will be necessary prior to construction of the reactor.

The neutron leakage assumed in the ORNL calculations was based on the GGA dimensional calculations and thus reflects the effect of power flattening.

3.6 ^{135}Xe Oscillation

Local reactivity control will be required to avoid excessive ^{135}Xe oscillation in the azimuthal direction. The period of these changes is so long that control presents no problem if the spatial power-density distribution can be adequately determined by sensing devices and if there is adequate local poison in the rods to exercise necessary spatial reactivity control. These conditions appear to have been met in both designs.

3.7 Burnable Poison

Use of burnable poison is proposed to reduce control rod requirements in the backup design. An estimate is made here of the fraction of the poison remaining after periods of exposure. The fraction of ^{10}B left after exposure ϕt is

$$\frac{N_{10}(t)}{N_{10}(0)} = \exp\,(-\sigma_{10}\phi t) \ .$$

The fraction of fed fuel, ^{235}U, remaining is

$$\frac{N_{235}(t)}{N_{235}(0)} = \exp\,(-\sigma_{235}\phi t) \ .$$

According to ORNL point-depletion calculations, the ratio of cross sections (integral average over the neutron spectrum) is $\sigma_{10}/\sigma_{235} = 3.84$, while the value of $\sigma_{235}\phi t$ for one year at a load factor of 0.8 is 0.5. With these data the fractions remaining of ^{235}U and ^{10}B are estimated to be those given in Table 3.1.

Table 3.1. Fractions of ^{235}U and ^{10}B Remaining at Various Exposure Times

Exposure Time for 0.8 Load Factor (Years)	Fraction ^{235}U Present	Remaining ^{10}B Fraction
0	1.0	1.0
1.0	0.607	0.147
2.0	0.368	0.021
3.0	0.223	0.003
4.0	0.136	0.0005
Four-zone end-of-cycle total		0.172

Thus it is estimated that 17% of the initial worth of the burnable poison will remain at the time of refueling. If the initial reactivity worth of the burnable poison in the newly fueled one-fourth of the core is 0.03 Δk, there would be an end-of-cycle penalty of 0.005 Δk reactivity tied up in poison that could not be removed. It is estimated that the effect of this poison would be to increase the fissile makeup required by 5%. This penalty was not included in ORNL calculations. More detailed calculations by GGA indicated that the residual burnable poison would be less than 10% rather than 17%.

Burnable poison is not used in the reference design.

3.8 Exposure Data

A summary of data on exposure of particles is given in Table 3.2. Maximum exposure of graphite to neutrons above 0.18 Mev is estimated at 8×10^{21} neutrons/cm^2.

3.9 First Cycle

Reactivity lifetime, as dependent on initial loading, is given in Table 3.3 for the backup design at a thermal efficiency of 40.7%. The data of Table 3.3 indicate a requirement for a 2298-kg ^{235}U initial

Table 3.2. Exposure and Flux Data for the HTGR

	Backup Core	Reference Core
Design core power level, Mw(th)	2,457	2,320
Core average power density, $w(th)/cm^3$	7.47	7.05
Batch average exposure,[a] Mw(th)d/MT		
Mean batch	66,000	62,000
Longest-exposure batch	90,000	90,000
Highest element exposure in mean batch, Mw(th)d/MT	105,000	99,000
Individual particle exposure, Mw(th)d/MT		
Fissile, equilibrium mean	630,000	650,000
Fissile, equilibrium maximum	700,000	700,000
Mixed, equilibrium mean	52,000	60,000
Mixed, equilibrium maximum	104,000	96,000
Mixed, first-loading mean	140,000	130,000
Mixed, first-loading maximum	170,000	160,000
Neutron flux level at equilibrium, $neutrons/cm^2 \cdot sec$		
Total, average	(b)	22×10^{13}
Total, maximum		$\sim 50 \times 10^{13}$[c]
Fast (E > 0.18 Mev), average		4.2×10^{13}
Fast (E > 0.18 Mev), maximum		$\sim 7 \times 10^{13}$[c]
High-energy flux (E > 0.18 Mev) exposure, $neutrons/cm^2$		
Equilibrium, average		4.2×10^{21}
Equilibrium, maximum		7×10^{21}
Maximum		8×10^{21}[c]

[a]Exposure is based on feed material; that is, tons of heavy metal.

[b]Flux and exposure are about the same as for the reference design.

[c]Maximum flux and exposure based on GA flux-peaking calculations.

Table 3.3. Reactivity Lifetime of Initial Loading at a Thermal Efficiency of 40.7%

Case	^{235}U Loading (kg)	Initial k_{eff}	Reactivity Lifetime (years) Full Power	Reactivity Lifetime (years) 0.8 Load Factor
M-1	2038	1.143	1.2	1.5
M-2	2298	1.169	1.6	2.0
M-3	2585	1.193	2.0	2.5

loading to achieve a 2.0-year reactivity life at a 0.8 load factor, which compares with an estimate by GGA of 1947 kg at 0.85 load factor.

For the reference design at a thermal efficiency of 43.1%, the initial loading required to achieve a 1.5-year life at a 0.8 load factor is estimated to be 1870 kg of ^{235}U compared with the GGA estimate of 1690 kg at a carbon-to-thorium ratio of 200 and 1764 kg at a ratio of 210.

3.10 Neutron Balances

ORNL calculations of neutron balances for the backup design are given in Table 3.4. Neutron balances for the reference design are given in Table 3.5, which presents results obtained for carbon-to-thorium ratios of 200 and 210. Results obtained for a lower exposure cycle are also shown. It is to be noted that when continuous fueling is initiated for the reference design, the power density might be excessive in the fresh elements. However, it would be simple enough to alter the refueling schedule to rectify this difficulty; for instance, by starting continuous refueling earlier. Such a change would not significantly affect the 30-year fuel-cycle history.

3.11 Mass Balance Histories

Mass balance histories for the 30-year core histories are given in Appendix B for the backup design and in Appendix C for the reference design.

3.12. Plutonium Makeup

Calculations were made to estimate the behavior of the concept with plutonium makeup. It was found that (1) the behavior is very sensitive to the amount of feed and causes great difficulty in "settling down" a calculation and (2) the resonance shielding changes considerably during exposure and makes simple calculations unrealistic. No results are reported. ORNL agrees with findings by GA[1] that only gradual changeover from ^{235}U makeup to plutonium would be likely to be possible.

Table 3.4. Calculated Behavior for the HTGR Backup Design Before Refueling[a]

	GA Calculation	ORNL[b] Calculations					
Refueling frequency, years at 0.8 load factor	1.0625	~0.75		1.0		~1.25	
Period of exposure, number of refuelings counted from start	9	13	38	10	29	8	23
Exposure, Mw(th)d/MT	72,000	50,800	50,100	67,000	66,500	77,400	82,500
Operation time, years at 0.8 load factor	10.63	10.49	29.97	11.0	30.0	10.63	30.19
Fissile loading, kg	1,540	1,475	1,606	1,580	1,729	1,578	1,898
Fissile makeup, kg/year	246	261	256	287	277	288	302
Fissile recycle, kg/year	242	381	430	304	341	251	280
Conversion ratio	0.798	0.803	0.798	0.777	0.773	0.771	0.738
Neutron balance, fractional absorption							
Nuclide							
^{232}Th	0.3411	0.3463	0.3349	0.3360	0.3247	0.3331	0.3109
^{233}Pa	0.0092	0.0102	0.0096	0.0097	0.0090	0.0096	0.0083
^{233}U	0.3350	0.3356	0.3284	0.3274	0.3200	0.3328	0.3096
^{234}U	0.0319	0.0343	0.0444	0.0327	0.0434	0.0325	0.0420
^{235}U	0.1196	0.1244	0.1333	0.1337	0.1432	0.1277	0.1512
^{236}U	0.0062	0.0047	0.0083	0.0063	0.0096	0.0058	0.0114
^{237}Np	0.0028	0.0019	0.0036	0.0028	0.0046	0.0026	0.0060
^{238}U	0.0021	0.0016	0.0014	0.0021	0.0019	0.0020	0.0026
^{239}Pu	0.0021	0.0016	0.0014	0.0021	0.0019	0.0020	0.0026
^{240}Pu	0.0008	0.0005	0.0005	0.0007	0.0007	0.0007	0.0009
^{241}Pu	0.0001	0.0004	0.0004	0.0006	0.0006	0.0006	0.0008
Fission products	0.0993	0.0799	0.0768	0.0895	0.0856	0.0947	0.1009
Carbon	0.0159	0.0192	0.0176	0.0178	0.0162	0.0177	0.0146
Leakage and control	0.0339	0.0394	0.0394	0.0386	0.0386	0.0382	0.0382
Total	1.0	1.0	1.0	1.0	1.0	1.0	1.0

[a]Conditions: carbon-to-thorium ratio of 200; one-fourth core refueling; recycle delayed one refueling.

[b]Exposure time through life varied slightly from the average; end-of-life results are most representative here.

Table 3.5. Calculated Behavior of the GA 1000-Mw(e) HTGR Reference Design with Continuous Fueling After First Exposure and ^{235}U Makeup

	GA Calculation	ORNL Calculations				
Exposure time,[a] years	4	4		4		2.667
Carbon-to-thorium ratio	200	200		210		210
Operation time, years	10	10	15	10	15	10
Fissile loading, kg	1,290	1,400	1,462	1,320	1,370	1,196
Fissile makeup, kg/year	180	199	191	198	196	182
Fissile recycle, kg/year	251	283	298	262	276	384
Specific power, kw(th)/kg fissile	1,800	1,657	1,587	1,760	1,690	1,940
Exposure, Mw(th)d/MT	62,600	63,050	62,820	66,100	65,900	44,600
Eta, ^{233}U	2.230	2.227	2.223	2.229	2.228	2.232
Eta, ^{235}U	1.979	1.978	1.975	1.983	1.980	1.989
Eta, system ^{233}U + ^{235}U	2.173	2.159	2.155	2.159	2.156	2.172
Conversion ratio	0.826	0.816	0.816	0.804	0.805	0.857
Neutron balance, fractional absorption						
Nuclide						
^{232}Th	0.3556	0.3533	0.3471	0.3476	0.3419	0.3655
^{233}Pa	0.0087	0.0100	0.0097	0.0101	0.0098	0.0108
^{233}U	0.3423	0.3342	0.3310	0.3297	0.3264	0.3456
^{234}U	0.0315	0.0328	0.0394	0.0335	0.0395	0.0340
^{235}U	0.1150	0.1256	0.1296	0.1301	0.1342	0.1102
^{236}U	0.0039	0.0042	0.0051	0.0044	0.0054	0.0028
^{237}Np	0.0014	0.0017	0.0021	0.0019	0.0023	0.0010
^{238}U	0.0014	0.0015	0.0014	0.0016	0.0015	0.0010
^{239}Pu	0.0013	0.0013	0.0013	0.0014	0.0013	0.0009
^{240}Pu	0.0005	0.0004	0.0004	0.0005	0.0004	0.0003
^{241}Pu	0.0004	0.0004	0.0003	0.0004	0.0004	0.0002
Fission products	0.0819	0.080	0.078	0.083	0.0819	0.0682
Carbon	0.0170	0.0157	0.0151	0.0170	0.0163	0.0187
Leakage and control	0.0391	0.0386	0.0386	0.0386	0.0387	0.0408
Total	1.0	1.0	1.0	1.0	1.0	1.0

[a]At 0.8 load factor.

3.13 Summary

A somewhat higher fissile loading was estimated for a given exposure than calculated by GGA for the reference design with a carbon-to-thorium ratio of 200. To achieve a 1.5-year life for the initial loading of the reference design, an initial ^{235}U loading of 1870 kg was estimated by ORNL, which is to be compared with 1690 kg calculated by GGA. Also a somewhat lower conversion ratio was obtained that increased the feed requirements. A mass balance yielded the following relationship,

Annual Net Fissile Makeup = Annual Feed — Annual

Discharge = [1 — Conversion Ratio] × Annual Consumption,

$$F - D = (1 - CR) \times Q \;,$$

with which a direct comparison of results may be made. A crude estimate of gross consumption and roundoff of data gave the following results for the reference design with continuous fueling after first core:

Calculated By	F — D = (1 — CR) × Q	Carbon-to-Thorium Ratio
GGA	180 — 16 = (1 — 0.826) × 943	200
ORNL	193 — 20 = (1 — 0.816) × 940	200
ORNL	203 — 20 = (1 — 0.805) × 940	210

References

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2. D. R. Vondy and T. B. Fowler, Computer Code TONG for Zero-Dimensional Reactor Depletion Calculations, USAEC Report ORNL-TM-1633, Oak Ridge National Laboratory, June 1967.

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8. Public Service of Colorado, Fort St. Vrain Nuclear Generating Station, Preliminary Safety Analysis Report, Amendment 2, Docket 50-267.

4. ENGINEERING EVALUATION OF REACTOR CORES

4.1 HTGR Core Thermal-Hydraulic Analysis

A summary of the characteristics of the core for both the backup and reference designs, as given by GGA, is presented in Table 4.1. This summary was collected from reports, drawings, correspondence, and personal communications.

From the calculations described in detail in this section, it was estimated that the pressure loss through the core would be 2.90 psi for the reference design. This value agrees quite well with the GGA calculation of 2.7 psi; the difference could well be attributed to a difference in the treatment of the entrance and exit losses. For the backup design, the pressure loss was considerably larger because the flow was orificed to a basic radial factor of 1.44 to give a greater mass flow rate in the hot channel. The ORNL calculation gave 5.31 psi through the core compared with a GGA result of 7.6 psi; the difference in these two values may be caused by the allowance for junction losses due to the misalignment of fuel blocks, which was included in the GGA calculation but not in ORNL's. Lack of detail on block and core tolerance made it difficult to calculate this contribution to the pressure loss. Block junction losses in the reference design are negligible because of the lower maximum mass velocity.

In the determination of maximum temperatures developed in the fuel, the highest center-line temperature calculated without using engineering factors was 2189°F for the reference design and 2090°F for the backup design. Both these maximum temperatures occurred at the exit from the core, and they are about 80°F higher than the highest temperatures at the midplane of the core.

The principal problem in the evaluation of the engineering factors was the determination of the effect of flow shunting around the coolant holes due to shrinkage of the graphite during irradiation. In evaluating this effect, it was assumed that 20% of the coolant flow was bypassed over the entire length of the coolant channels. With the inclusion of all engineering factors, it was estimated that the maximum fuel center-

Table 4.1. HTGR Core Thermal-Hydraulic Characteristics

	Fort St. Vrain Design	Backup Design	Reference Design
Core power, Mw	837	2457	2318
Mean core diameter, ft	19.5	31.1	31.1
Active core height, ft	15.6	15.5	15.5
Top axial reflector height, ft	3.25	3.225	3.225
Bottom axial reflector height, ft	3.90	5.167	5.167
Fuel blocks			
Number	1482	3841	7591
Distance across flats of hex, in.	14.2	14.2	14.2
Height, in.	31.2	31.2	15.6
Coolant hole diameter, in.			
Small hole	0.530	0.530	0.530
Large hole	0.625	0.625	0.625
Number of coolant holes			
Small holes	6	6	6
Large holes	102	102	102
Fuel hole diameter, in.	0.45	0.45	0.45
Number of fuel holes	210	210	210
Coolant conditions			
Coolant	Helium	Helium	Helium
Inlet coolant pressure at core, psia	700	700	700
Pressure drop across core, psi	8.3	7.6	2.7
Mass flow, core and reflector, lb_m/hr	3.33×10^6	9.8×10^6	8.85×10^6
Mass flow, total, lb_m/hr	3.41×10^6	10.27×10^6	9.28×10^6
Core inlet temperature, °F	758	758	803
Core outlet temperature, °F	1449	1449	1524
Change in temperature across core, °F	691	691	721
Orificing to basic radial flux pattern	No[a]	No[a]	Yes
Peaking factors			
Axial			
Basic peak to average flux		1.31	1.31
Position of peak flux		Midplane	Midplane
Local peaking due to axial fuel gap[b]		0.20	0.20
Total peak-to-average flux at midplane	1.44	1.51	1.51
Total peak-to-average flux at core exit		1.115	1.115
Radial			
Maximum bundle-to-average flux		1.44[c]	1.10
Maximum-to-average for bundle		1.12	1.4[c]
Total maximum-to-core average	2.3	1.61	1.54
Flux tilt and overpower		1.10	1.10
Summary			
Total peaking factor at midplane	3.3	2.67	2.56
Total peaking factor at core exit	1.96	1.98	1.89

[a]Adjustable orifices are regulated to give the same exit coolant temperature from each coolant channel.

[b]This factor, as given by GA, should be added to the basic axial peaking factor to give 1.51 rather than multiplying by 1.20 as is usually done with such factors.

[c]These factors contain the fuel aging factors.

line temperature would be 2632°F for the reference design and 2477°F for the backup design. Both these values are below the maximum allowable value of 2732°F, which was proposed by GGA. These maximum temperatures occur at the exit from the core, and they are about 150°F more than the center-line temperatures at the midplane.

4.1.1 Pressure Distribution

The pressure loss through the core of the HTGR was calculated by using the graphical correlation for the friction factor for smooth tubes as presented by Coulson and Richardson.[1] This friction factor, f, is defined as the shear stress at the wall divided by the product of the density of the fluid and the velocity squared. The friction factor is one-half the conventional Fanning friction factor and one-eighth the Moody friction factor.

The assumption of a smooth surface may result in a nonconservative estimation of the pressure loss through the core; however, the difference between this assumption and the normal roughness encountered in a hole fabricated in a graphite block should be small. The conservative (or low) estimation of the friction factor will result in a conservative (low) estimation of the heat transfer coefficient to be used with the Reynolds analogy in subsequent calculations. Almost any degree of smoothness should be easy to obtain in the fuel assembly fabrication. Any degree of roughness will add to the pressure loss through the core, but at the same time it will enhance the rate of heat transfer.

The pressure loss through the core was estimated by adding entrance and exit losses to the friction losses through the core and upper and lower axial blankets. The entrance and exit losses were difficult to evaluate because of the lack of detail concerning the inlet plenum and outlet header. It was assumed that the channel with the maximum mass velocity was not orificed and that one-half a velocity head was lost at the entrance. At the exit, it was assumed that one velocity head was lost. The pressure loss calculated by using these assumptions may be large, since a properly designed exit header could remove part of the velocity head existing in the coolant holes. However, since transition losses between the blocks of the core and reflector are neglected, any overestimation of the exit loss will, in part, be compensated.

In the reference design, the inlet and exit losses were found to be 0.091 and 0.283 psi, respectively. The friction loss based on the average of the squares of the inlet and exit velocities was found to be 2.530 psi. For this calculation 15.5 ft was assumed as the axial length of the core and 38.7 and 62.0 in. were assumed for the upper and lower axial reflectors, respectively. The total pressure loss for the core is thus 2.90 psi. This compares well with the value of 2.7 psi given by GGA as the pressure loss across the core for the reference design.

When the same method of calculation was used for the backup design, the inlet loss was estimated to be 0.182 psi and the outlet loss, 0.569 psi. The loss due to friction in the core for the same axial lengths as the reference design was estimated to be 4.55 psi. This gives a total pressure loss through the core for the backup design of 5.31 psi. This value is significantly less than the value of 7.6 psi given by GGA. This difference would not be accounted for even if significant roughness were assumed in the coolant holes. However, the omission of junction losses in the ORNL calculation could explain it.

Table 4.2 summarizes the principal parameters and results of these pressure loss calculations for the two designs. Physical properties of helium at the average coolant temperatures used in these calculations were obtained from Ref. 2.

4.1.2 Coolant Temperatures and Power Distributions

The distribution of temperatures was evaluated at two locations along the axial length of each core; these were (1) the midplane where the maximum flux occurs, and (2) the exit of the core where the maximum coolant temperature occurs. The temperatures for the two designs at the core exit are given in Table 4.1. The temperature at the midplane of the reactor was calculated by using the GGA finding[3] that 56.2% of the heat is generated above the midplane. The temperature of the coolant at the midplane for the reference design was 1208°F, and the temperature of the coolant at the midplane for the backup design was 1146°F.

Table 4.2. Pressure Loss Summary

	Backup Design	Reference Design
Coolant density at inlet, lb_m/ft^3	0.21	0.20
Coolant density at outlet, lb_m/ft^3	0.14	0.13
Maximum coolant velocity at inlet, ft/sec	126.0	92.0
Maximum coolant velocity at exit, ft/sec	197.0	142.0
Maximum mass coolant flow, $lb_m/ft^2 \cdot sec$	26.8	18.4
Maximum Reynolds number at average temperature	51,500	35,500
Friction factor (from Ref. 1)	0.0026	0.0027
Calculated pressure loss over core, psi	5.31	2.90
Inlet loss, %	3.44	3.14
Exit loss, %	10.73	9.73
Friction loss, %	85.82	87.13
Deviation from stated core pressure loss, psi	−2.29	+0.20
Deviation from stated values, %	−43.12	+7.41

For the reference design, the average power density in the fuel was 38.6 w/cc. The power density at the midplane was found by multiplying this quantity by the factor 2.56, the product of an axial peak-to-average flux ratio of 1.51, a radial peak-to-average ratio of 1.54, and 1.10 to allow for overpower and flux tilt. The factor of 1.10 can either be considered an allowance of 10% for overpower or combined allowances of 5% overpower and 5% flux tilt. The allowance for power uncertainty is compatible with the allowances used in other reactor evaluations.

In the thermal analysis of the Fort St. Vrain plant,[4] GGA proposed a 10% uncertainty on the peak-to-average flux ratios both in the axial and in the radial directions. It has not been ORNL's procedure to impose uncertainties on the nuclear physics calculations of power distribution, although certainly these uncertainties exist. The physics calculations present the most probable calculated values. It is these most probable values that were chosen to represent a later-generation plant.

For a first-generation plant, it is desirable to use more conservative values in the design, as it appears GGA has done for Fort St. Vrain.

The combination of peaking factors gives a maximum power density at the midplane of 98.8 w/cc for the reference design. The same procedure applied to the core exit gives a total peaking factor of 1.89 and a maximum power density of 72.9 w/cc.

For the backup design, the average and maximum at the midplane and maximum at the exit power densities are 41.0, 109.5, and 80.9 w/cc, respectively. The multiplying factor to convert the average power density to the maximum power at the midplane is 2.67; it is the product of three factors: (1) the total axial peaking factor of 1.51, (2) the total radial peaking factor of 1.61, and (3) the overpower-flux tilt factor of 1.10. At the exit from the core the factor is 1.975.

4.1.3 Temperature Difference Between the Bulk Coolant and the Coolant Hole Wall

The film heat transfer coefficient was estimated by using the Reynolds analogy; that is, that the Stanton number is equal to the friction factor when the friction factor is defined in the manner described previously in this section. The Reynolds analogy holds exactly only for fluids with a Prandtl number of one. For fluids such as helium with a Prandtl number less than one, the analogy predicts a conservative estimate (i.e., underestimate) of the heat transfer coefficient. It is estimated that the degree of this conservatism is approximately 10%.

Theoretically the molal heat capacity of helium should be 2.5 times the gas constant, R, which on a weight basis gives 1.24 $Btu/lb_m \cdot °F$. The experimental values reported range from 1.235 to 1.245.

From the relationship

$$\frac{h}{GC_p} = f$$

the value of h was found to be 222 $Btu/hr \cdot ft^2 \cdot °F$ for the reference design and 288 $Btu/hr \cdot ft^2 \cdot °F$ for the backup design. The value of h will vary only as the value of f varies. The friction factor in turn is dependent on the Reynolds number, and variations in the Reynolds number over the length of the core depend only upon the variation of the viscosity. A conservative estimate of the heat transfer coefficient at the hot end is obtained by evaluating the coefficient at the average temperature of the core coolant. Also, as mentioned previously, an increase in roughness will produce an increase in the friction factor and a proportional increase in the heat transfer coefficient.

In order to determine the temperature difference between the bulk fluid and the coolant hole wall, it is necessary to determine the thermal flux at the surface. The average for the core was found to be 14.2 w/cm^2 of surface for the reference design and 15.1 w/cm^2 for the backup design. The maximum surface flux at the midplane for the reference design was 36.3 w/cm^2, and at the exit from the core it was 26.8 w/cm^2. For the backup design the corresponding values were 40.2 and 29.7 w/cm^2.

The temperature differences between the bulk coolant and the coolant hole wall for these fluxes and the estimated heat transfer coefficients were calculated to be 520 and 383°F at the midplane and exit for the reference design and 442 and 327°F, respectively, for the backup design.

4.1.4 Temperature Difference Across the Graphite

The path length for thermal conduction is assumed to be the minimum distance between the fuel and the adjacent coolant hole, or 0.20 in. For a thermal conductivity of the graphite equal to 14 Btu/hr·ft·°F, the temperature difference across the graphite is 104°F at the midplane and 76°F at the exit of the reference design. For the backup design the temperature difference across the graphite is 115°F at the midplane and 85°F at the core exit.

The thermal conductivity of 14 Btu/hr·ft·°F used for the graphite is for irradiated material. For the reference design, the age peaking factor will be highest for the new fuel, which will contain graphite with a lower exposure to the radiation. The use of this thermal conductivity introduces another degree of conservatism in the calculation. Since graphite is anisotropic and its thermal conductivity is dependent on the method of fabrication, a more detailed treatment of this variation would be too uncertain for this evaluation.

4.1.5 Temperature Gradient in the Fuel

The temperature difference between the edge and the center of the fuel was calculated with the expression

$$\Delta t_f = \frac{q_f R^2}{4k_f} ,$$

where q_f is the power density in the fuel, R is the radius of the fuel hole, and k_f is the thermal conductivity of the fuel (assumed to be independent of temperature). A value for k_f of 3 Btu/hr·ft·°F (0.052 w/cm·°C) was taken for the fuel; this was assumed to be an effective conductivity for the compact, and it therefore includes the effect of the contact resistance or backfill gas gap at the graphite surface. This value is based on experimental measurements made by GGA. The radius is 0.225 in. or 0.572 cm. With fuel power densities of 98.8 and 72.9 w/cc for the reference design at the midplane and exit, the corresponding temperature differences are 280 and 206°F. For the backup design with fuel power densities of 109.5 and 80.9 w/cc, the temperature differences at the respective locations are 310 and 229°F.

The total temperatures and temperature differences are summarized in Table 4.3 for the two designs at the two locations. These temperatures include the total axial and radial peaking factors and the 10% allowance for overpower and/or flux tilt. They do not include any engineering factors or allowance for flow shunting due to the shrinkage of the graphite.

From Table 4.3 it can be deduced that the maximum graphite temperatures are 1703°F (928°C) at the midplane and 1861°F (1016°C) at the exit for the backup design, and at the same locations, 1832°F (1000°C) and 1983°F (1083°C) for the reference design. It should be noted, however, that the calculation of these temperatures includes fuel aging factors (see Table 4.1) of 1.31 for the backup design and 1.25 for the reference design. This peaking occurs immediately after the insertion of a new fuel patch (or column) in the core structure to replace spent fuel. As burnup of this new fuel proceeds, these temperatures decrease. The corresponding temperatures at end of fuel life are 1571°F (855°C) and

Table 4.3. Temperatures in the Core Without Engineering Factors

	Backup Design		Reference Design	
	Midplane	Exit	Midplane	Exit
Temperature of coolant, °F	1146	1449	1208	1524
Power density in fuel, w/cc	109.5	80.9	99.0	72.9
Heat flux at hole wall, w/cm^2	40.2	29.7	36.3	26.8
Heat transfer coefficient, Btu/hr·ft^2·°F	288	288	222	222
Temperature drop from gas to wall, °F	442	327	520	383
Temperature drop across graphite, °F	115	85	104	76
Temperature drop to fuel center line, °F	310	229	280	206
Maximum fuel temperature, °F	2013	2090	2112	2189

1764°F (962°C) for the backup design and 1707°F (931°C) and 1891°F (1032°C) for the reference design.

4.1.6 Engineering Factors

In the evaluation of the engineering factors, it is desirable to reduce them to two groups that will render them comparable with engineering factors applied in other evaluations. The two groups are those factors that affect the temperature change in the coolant as it flows through the core and those factors that affect the difference in temperature between the center of the fuel pin and the bulk coolant. For convenience these factors are called the engineering factor on the coolant temperature rise and the engineering factor on the temperature difference.

Contributing to the variations in the coolant temperature rise are flow maldistribution and disturbances in the amount of heat added to the coolant. Specifically, an increase in the temperature of the coolant that might develop at any position will result from a decrease in the flow of coolant or an increase in the amount of heat input. Two factors account for this effect; these are a dimensional tolerance allowance and an error in the measurement of the exit coolant temperature. As discussed in the design report for Fort St. Vrain,[4] the dimensional tolerance is a consideration that the coolant hole diameter could be 0.620 in. rather than 0.625 in. as specified. In the Fort St. Vrain design, orifices are provided in each flow path to continuously control the exit coolant temperature. Flow maldistribution is therefore directly dependent on the ability to measure accurately the outlet coolant temperature and the ability to adjust the flow to control this variable. Accurate temperature measurement depends upon the proper design to avoid thermal radiation effects and to compensate for gamma heating. Subsequent interpretation of the sensor signal depends on the sensitivity of the indicating device; for the Fort St. Vrain design, an instrument error of 20°F was postulated,[4] so a temperature error at the core exit greater than this value would be expected for these designs. The backup design also presents the feature of controllable orifices, but in the reference design the flow is regulated to a fixed pattern by adjustment of the coolant holes in the axial reflector blocks.

Since both the backup and reference designs have considerably more flow channels to control than the Fort St. Vrain design, it is anticipated that the effects of flow maldistribution will be more severe. The effect will probably be most pronounced in the reference design, which does not have the controllable orifices. For uniformity, however, the same factor was applied to each design. It appears reasonable to assume a 5% flow maldistribution factor on the coolant temperature rise. This in effect would increase the coolant temperature rise in direct proportion and would reduce the heat transfer coefficient between the gas and the wall in proportion to the change in velocity to the 0.8 power. The net effects of flow maldistributions shown in Table 4.4 are to increase the fuel center-line temperatures in the range 37 to 51°F.

Another factor to be considered is the uncertainty in the heat generation. This uncertainty can be attributed to variances in fuel loading, fuel enrichment, fuel hole size, and other nonuniformities that can contribute to a local perturbance of the flux pattern other than control rod manipulation. The uncertainty resulting from fuel loading in Fort St. Vrain is stated by GGA[4] to be 6%. This was a 6% increase in local power, and it is assumed that the 6% would apply only to the difference between the

Table 4.4. Temperatures in the Core with Effects of Engineering Factors Included

	Temperatures (°F)			
	Backup Design		Reference Design	
	Midplane	Exit	Midplane	Exit
Temperature of the bulk coolant	1146	1449	1208	1524
Coolant temperature rise from inlet	388	391	405	721
Temperature difference between center line and bulk coolant	867	641	904	665
Temperature difference between coolant wall and bulk coolant	442	327	520	383
Center-line temperature without effects of engineering factors included	2013	2090	2112	2189
Engineering factor effects				
Flow maldistribution				
Coolant temperature rise (5%)	19	35	20	36
Film temperature difference (4%)	18	13	21	15
Total	37	48	41	51
Fuel loading				
Coolant temperature rise (10%)	39	69	41	72
Total temperature difference (10%)	87	64	90	67
Total	126	133	131	139
Flow shunting				
Coolant temperature rise (25%)	97	173	101	180
Film temperature difference (19%)	84	62	99	73
Total	181	235	200	253
Total effect of engineering factors	344	416	372	443
Center-line fuel temperature with effects of engineering factors included	2357	2506	2484	2632

center-line temperature of the fuel and the bulk coolant temperature. It can be argued, however, that a variation from the sources indicated above might well be effective over a significant axial length of the coolant channel and might, therefore, also affect the coolant temperature rise.

Without further detailed investigation of the effect of the above perturbances, it is felt that an appropriate allowance for uncertainty on the heat generation would be 10% and that this factor should also be applied both to the temperature difference at the axial position and the temperature rise to that position. The reasonableness of this assumption is supported by the fact that several other uncertainties evaluated in the Fort St. Vrain design were neglected, as indicated below.

Accumulative increases to the fuel center-line temperatures due to the 10% uncertainty on both the temperature rise and the temperature difference are shown in Table 4.4 for the two designs. The maximum increase of 190°F in the reference design may be compared with the maximum value of 189°F determined for the Fort St. Vrain design.

The uncertainties for the thermal conductivity of the fuel and graphite, as indicated for the Fort St. Vrain design, were omitted from this analysis. This omission is consistent with the previously stated policy of using the most probable values at all points in the evaluation rather than the most conservative value. In this manner the designs are evaluated as later-generation concepts rather than first-of-a-kind concepts.

The remaining factor to be considered is the effect of flow shunting. Flow shunting refers to the bypassing of coolant around the coolant holes because of the development of horizontal and vertical apertures between the graphite blocks from anisotropic dimensional changes in the graphite due to irradiation and thermal cycling. According to preliminary detailed mathematical analysis of the Fort St. Vrain design by GGA, 20% of the coolant, at most, might bypass a coolant channel. In the following calculations the treatment is such that all the flow bypassing is considered to occur at the inlet to the channel; if the bypassing occurs farther downstream the percentage bypassed can be greater and still produce the same increase in temperatures. Bypassing 20% of the coolant flow will produce a 25% increase in the coolant temperature rise, but because of the dependence of the heat transfer coefficient on flow, the corresponding increase in the temperature difference between the bulk gas and the wall will only be about 19%. The maximum additive temperature due to flow shunting is 253°F at the exit of the reference design core. The fuel temperature increases due to this effect, shown in Table 4.4, can be compared with the value of 190°F that was considered by GGA as an extreme value for the Fort St. Vrain design.[4]

Numerous suggestions are made in the recent literature[5,6] for the statistical treatment of engineering, or hot-channel, factors. Such a treatment is based on the limited probability that all effects will occur at the same time and place. In addition, it can be argued that the events producing the various overtemperature effects are not statistically independent but are actually caused by the same source; they are therefore most likely to occur in conjunction with each other. As a result, in the evaluation of effects that are to be expressed additively (as is being done in this analysis), it is important to choose the most probable or most realistic value. In the combination of the factors statistically, a much more liberal choice for the variant can be made; it must be related, however, to the choice of the associated probability. It has been observed that the two methods usually result in similar overall values for the uncertainties.

Table 4.4 summarizes the calculated core temperatures, including the effect of engineering factors and uncertainties. As with Table 4.3, the graphite temperatures that can be deduced from the tabulation are those which include the age peaking factors; therefore they are the temperatures that might be developed under the maximum deviation from normal operation in a fresh fuel patch (or column). Since flow shunting due to deformation of the fuel blocks under exposure to irradiation occurs toward the end of fuel life, the combination of these effects represents unusually severe conditions. Removal of the age peaking factors, which occur only at the beginning of fuel life, would result in a maximum graphite temperature of 1970°F (1076°C) at the midplane and 2220°F (1216°C) at the exit for the backup design and 2132°F (1166°C) and 2372°F (1300°C) at the same locations in the reference design.

4.1.7 Summary

From the foregoing calculations, it is concluded that the pressure loss for the core of the reference design agrees quite well with the specified value. For the backup design, ORNL calculated a slightly lower pressure drop than that given by GGA. The difference can be attributed in part to pressure losses in the inlet and outlet plenums, but the difference is larger than can be assigned completely to this effect. Junction losses at the interface between fuel blocks were not included in this analysis, but if they had been they would account for the difference. Insufficient information is available on the method of orientation of the blocks to allow us to make a careful analysis of the junction losses.

The center-line temperatures ORNL calculated are below the maximum allowable value of 2732°F (1500°C) proposed by GGA. This fact is even more significant when the degree of conservatism in the ORNL calculations is considered. Evaluation of this temperature limit is discussed in Section 4.2.

From the calculated maximum temperatures, it can be concluded that the backup design can be operated at about 15% higher power than specified. The difference in the maximum temperature calculated for the reference design and the maximum allowable temperature is not significant in view of the uncertainties in the calculation.

The major uncertainty at present is the allowance to be made for flow shunting. This particular aspect is the major item for which additional research and development is needed. GGA is developing both analytical and experimental methods for the evaluation of this effect.

4.2 Fuel Element Performance Evaluation

The fuel element for the proposed 1000-Mw(e) HTGR (Fig. 2.1) is very similar in design to the element now proposed for the 330-Mw(e) Fort St. Vrain reactor.[7] Since the Fort St. Vrain fuel elements will be irradiated under conditions similar to those anticipated in the 1000-Mw(e) concept, a proof test of the general design should be available before operation of a 1000-Mw(e) plant. However, many details of the fuel element design are uncertain at the present time. For example, the 1000-Mw(e) concept may use $(U,Th)O_2$ fuel rather than the $(U,Th)C_2$ fuel used in the Peach Bottom and initial Fort St. Vrain concepts, and the grade of graphite to be used has not been specified for the Fort St. Vrain or 1000-Mw(e) concepts. Peach Bottom has purged elements, while Fort St. Vrain and the later plants are to use nonpurged elements. Further, the coating specifications may be subject to modification; that is, there may or may not be SiC layers present. Thus the fuel element requires considerable additional development.

Since the HTGR can use ^{233}U as a major fissile component, a successful fuel cycle would depend on the development of remote reprocessing and refabrication schemes for mixed uranium-thorium coated-particle fuel. Development of remote reprocessing and refabrication for such fuels is currently being done at ORNL and GGA. The head-end processes, such as those for mechanically breaking the hex blocks, burning the carbon, and screening or leaching to remove ^{233}U, are presently in the concept stage. If SiC were added to the particles, this would have some effect on reprocessing. The solvent-extraction process for separating ^{233}U from ^{232}Th is well understood and is not considered a developmental problem. Sphere forming and coating are being done on a moderate scale in contact facilities. Some equipment development will be required before these operations can be done remotely. Manufacture of the fuel sticks is still in the laboratory stage. Some further development is required before full-scale equipment design can begin. The loading of the fuel sticks and inspection of the fuel blocks are still in the concept stage.

The only difference between the backup and reference designs from the standpoint of fuel element design is that the hex block element is longer by a factor of 2 in the backup design. The reference design has a block length of 15.6 in. This may make some difference in fabrication costs.

4.2.1 Prediction of Coated-Particle Performance

ORNL carried out a series of calculations based on the Prados-Scott mathematical model* of coated-particle irradiation behavior[8,9] to assess

*If SiC-coated particles were used, it would be necessary to use a different mathematical model (designated the Kaae model) to design the particles and to analyze their behavior. The Kaae model is in use by GGA to design particles and to interpret irradiation test results.

the expected performance of the coated particles for the HTGR reference-design fuel element. These calculations provided estimates of the stresses and strains in the coatings of both fissile and fertile particles resulting from fuel-kernel swelling, fission-gas pressure, and fast-neutron damage in the pyrolytic carbon as functions of irradiation exposure. The performance of coated-particle fuel specimens predicted by the model shows good agreement with results of a number of irradiation experiments tested to HTGR design burnup, as reported by Coobs and his associates.[10]

In making these calculations, coated-particle dimensions and properties were taken from HTGR reference-design specifications; these are listed in Table 4.5. The mechanical properties and creep behavior expected for the outer primary-containment layer of the pyrolytic-carbon coatings were estimated from data published by Price and Bokros.[11] The center-line temperature was assumed to be 1500°C, which is the value specified by GGA as the design maximum and is slightly higher than the predicted maximum. It

Table 4.5. Dimensions and Properties of Coated Fuel Particles for Reference-Design HTGR

	Fissile Particle	Fertile Particle
Fuel kernel		
Diameter, μm	150	350
Composition, at. %		
^{232}Th	0	97.3
^{233}U	0	2.4
^{235}U	93.0	0.3
Maximum center-line temperature, °C	1500	1500
Buffer coating layer		
Thickness, μm	90	30
Density, g/cm^3	1.0	1.0
Fractional free volume[a]	0.275	0.275
Outer coating layer		
Thickness, μm	60	100
Density, g/cm^3	>2.0	>2.0
Bacon anisotropy factor	<1.1	<1.1

[a]Assumed to be 50% of total porosity.

was also assumed that the Bacon anisotropy factor (BAF) could be reproducibly controlled at a value of 1.1 or less. Inspection and control procedures to assure maintaining such parameters in the production fuel are required.

The results of these calculations indicate that the maximum tangential stresses developed in the particle coatings during the fuel operating lifetime will be below 12,200 and 17,800 psi for the fissile and fertile particles, respectively. Maximum expected tangential strains will be below 1.6% for either type of particle. From the data of Price and Bokros,[11] coatings would be expected to withstand tangential stresses in excess of 30,000 psi and strains in excess of 5%. Although the calculations and much of the irradiation data are based on mixed carbide kernels, recent irradiation data indicate that mixed oxide fuels will exhibit similar behavior.

The results indicate that the coatings are quite conservatively designed, and that after considerable operating experience, subsequent reactors might expect enhanced fuel performance by using thinner coatings or larger particles. However, no firm predictions can be advanced because irradiation tests to date have achieved only about one-half the design level of fast-neutron dose (8×10^{21} neutrons/cm^2, E > 0.18 Mev). Results of these tests show that particles with two-layer carbon coatings, such as those specified for the reference HTGR, will survive burnups of 20 to 24% FIMA and fast-neutron doses of approximately 5×10^{21} neutrons/cm^2 (E > 0.18 Mev) over the range of temperatures specified. However, significant increases were observed in the anisotropy of coatings that were originally almost isotropic (BAF $\leq$ 1.1).[12,13] Higher fast-neutron doses would be expected to induce higher stresses in the coatings, which would produce further creep and increases in preferred orientation and might lead to premature failure.

During similar testing of dense, relatively isotropic pyrolytic-carbon strip specimens at fast-neutron doses as high as 5.6×10^{21} neutrons/cm^2 (E > 0.18 Mev), net volume increases and anisotropic dimensional changes were measured.[14] The dimensional changes were greatest for specimens with densities greater than 2.0 g/cm^3 and BAF's of 1.05 or greater, and the changes were observed to accelerate with increasing dose; this again indicated that significant increases in anisotropy occurred. Increases in the degree of preferred orientation were previously observed after irradiation at lower doses and were confirmed by x-ray measurements.[15]

These results emphasize that caution must be used in predicting performance, and they stress the importance of further testing to fast-neutron doses at or near the design values. Experiments now in progress at GGA and ORNL should help to meet this objective.

4.2.2 Selection of Optimum Design for Graphite Structural Elements

Many factors enter into the design of the graphite hex blocks. The graphite to be used has not been specified for the present design except to state that it is "nuclear grade." For the purpose of the ORNL evaluation, EGCR-type AGOT graphite was assumed. However, this grade may not withstand the stresses generated in the structure. Further analysis and testing is obviously necessary before specifying the grade of graphite.

Since the coated fuel particles are bonded into a "stick," particle containment in case of gross fuel element failure is not a major problem in this design. Little or no contamination should result from breakage of an element if the particle bond integrity remains after block failure.

The strains generated in the hex blocks are difficult to calculate. In addition to strains due to column weight and coolant drag, two other sources of strain should be considered. These are dimensional changes due to radiation damage in the graphite and to gradients in the flux and temperature.

Experimental evidence of the dimensional changes in nuclear-grade graphites irradiated to high fast-neutron doses at high temperatures has become available only recently. Most of the data are from specimens irradiated in the GETR to doses of greater than 10^{22} neutrons/cm^2 ($E > 0.18$ Mev) at temperatures up to 1000°C.[12] In the radial direction, $\Delta L/L$ versus the fast-neutron dose shows a minimum at some dose less than 10^{22} neutrons/cm^2. The position of this minimum decreases rapidly with increasing temperature and reaches less than 5×10^{21} neutrons/cm^2 at 1000°C. The position of the minimum is probably also a function of graphite grade. The minimum corresponds to about 1% shrinkage in the radial direction, which again depends on grade. There is also a shrinkage of about 3% in the axial direction that is not of concern here. After passing through the minimum the graphite expands at a rapid rate. Typical expansion rates are on the order of 3% per 10^{21} neutrons/cm^2. This expansion of graphite leads to cracking and eventual failure at strains of about 20%.[12]

If radiation damage to graphite is to be comparable between reactors with different moderators and different fast flux spectra, it is necessary to generate a factor that makes the flux scales comparable. The proper factor for comparison of GETR and HTGR appears[13] to be about 1.16. That is, the fast dose expected in HTGR must be increased by a factor of 1.16 before comparison with the experimental results quoted above.

The maximum integrated fast-neutron flux ($E > 0.18$ Mev) for the present design has been calculated to be about 0.7×10^{22} neutrons/cm^2. Some fuel in the first core loading will have longer residence time and will accumulate a dose of about 0.8×10^{22} neutrons/cm^2. Since the particular grade of graphite to be used has not been specified in the present design, it is difficult to predict the extent of dimensional changes from the above data. In addition the data on the position of the minimum as a function of temperature is at present very uncertain, and the temperatures will drop during irradiation.

In the present design it is assumed that there will be 1% radial shrinkage. If the graphite is near the minimum in the $\Delta L/L$ versus fast fluence curve, this assumption appears reasonable. However, the particular grade of graphite chosen must be tested to high fast-neutron dose at the temperature that will occur during operation. Radial expansion of the hex blocks greater than the manufacturing and design tolerances would lead to jamming of the blocks, which could interfere with refueling operations.

Gradients in the flux and temperature, especially in the outer columns, might lead to stress levels higher than the design stress. Calculations based on cantilever-beam behavior and an unrestrained column of hex blocks subject to maximum flux gradients indicate that the column might suffer enough shrinkage on its inner edge to bow about 6 in. Since in the reactor the column would be supported by bearing on the surrounding columns, higher stress levels would occur than would be calculated due to the column weight and coolant drag. These problems are made less severe by the occurrence of creep.

From the above ORNL thinks that a theoretical calculation of the stress a hex block must support without gross failure would be difficult to perform. However, it should be attempted to give a better indication of the severity of the problem and insight into possible design measures to reduce the operating stresses. This calculation, which is being made by GA, was beyond the scope of this study.

Although the stresses to which the graphite structure will be subjected are not known precisely, graphite has a capacity to absorb stresses without massive failure. In a small area where the applied stress exceeds the fracture stress in the material, a crack will form that will relieve the excess stress. These cracks will not, however, tend to propagate into gross cracking and ultimate failure.

The shoulder on which the hex block is lifted seems adequate for the job. ORNL calculations indicate that the present design will survive a reasonable drop (~12 in.) onto the charging machine. This calculation was based on a tensile strength of 1500 psi for the graphite.

Adequate space is provided for loading the required amounts of fuel into the reactor if the present coating thicknesses are maintained. The entire reactor loading for an equilibrium core of the backup design (39,030 kg of thorium and 2600 kg of uranium) fits at a packing density of 52%, while packing densities of 60 to 64% are attainable. The zone of maximum loading will have a packing density of 54%, which is again well below the maximum attainable. This figure is calculated for a stick diameter of 0.445 in. to give a clearance of at least 0.004 in. on the diameter so that the sticks may be readily loaded into the fuel holes. The design provides for 1% shrinkage of the graphite structure during radiation. If no shrinkage occurs in the fuel sticks, a diametral clearance of 0.0025 in. is necessary so that the sticks will not be loaded in compression by shrinkage of the hex block. However, it appears that a clearance of 0.0025 in. will not be sufficient for physically loading the fuel sticks into the holes. Therefore the clearance necessary because of radiation shrinkage will always be less than that necessary for loading.

It may be possible, therefore, to increase the loading of the reactor without any great changes in the hex block design. If physics or heat transfer considerations limit the design to the present loading, there are several ways in which the packing density can be altered without grossly affecting the performance of the fuel element. For example, the coating thickness can be varied to give uniform packing while varying the heavy-metal contents for the variously loaded zones.

One of the major technical problems foreseen in hex block manufacture will be to hold the relatively tight tolerances on gang drilling of the fuel and coolant holes. Drilling the fuel and coolant holes from both ends seems feasible. The major disadvantage of single-end drilling appears to be the possibility of excess runout. A major expense may be incurred in control and inspection of runout. These approaches require process development. Another difficulty with the hex block design is the requirement for flat parallel ends. Design tests are now being conducted by GGA to determine the effect of out-of-flat ends on vibration in the core. Possibly these tests will lead to relaxation of the ±0.005-in. tolerance on the ends.

4.2.3 Development of Remote Refabrication Technology

The operation of the HTGR with an optimal fuel cycle is dependent on the development of a feasible economic process for recycle of ^{233}U. Recycle is complicated by several factors. First the entire operation has been assumed to be remote. Second, in most fuel-management schemes, the ^{235}U content must be minimized in the recycle fuel to minimize buildup of ^{236}U. Development of a plant-scale remote process for fuel suitable for HTGR has been under way for some time at ORNL. A remote recycle facility will require a large amount of development. The major areas of necessary development are summarized below.

The first area is the necessity of a head-end process to separate and remove ^{235}U particles to minimize contamination of the recycle ^{233}U with ^{236}U. Several processes are being developed. They involve coating the several types of particles with an extra layer to make them either physically or chemically different from each other. One suggestion is to use an inert carbide (e.g., SiC) coating on the fissile particles. The entire hex block could then be burned and the ^{233}U recovered by leaching.

The production of adequate coated particles does not seem to be a critical problem at present, but the diameter and coating density requirements necessary to insure proper loading in the various reactor zones may be difficult to meet under production conditions. Research to define the effect of process variables on the properties of the coatings is in progress. More research in this area is obviously necessary so that effective predictions of the effect of processing variables on coating properties can be made.

After the particles are coated, they may be loaded into molds to be bonded into fuel sticks or loaded directly into the hex blocks before bonding. Several types of particles must be blended in each stick in accordance with strict composition specifications. Developments at ORNL have indicated that controlled feed-rate blending will produce particles without segregation.[14]

After loading, a binder consisting of a mixture of an epoxy resin and powdered graphite or charcoal is infiltrated into the mold. This bonds the particles together when cured. Techniques of providing a fast curing cycle to optimize production operations are being developed by ORNL.[14] Stick loading into the fuel holes has not yet been attempted under remote conditions, but no serious problems should be encountered if adequate diametral clearance is provided and the curing conditions are adjusted so that out-of-roundness and bowing are minimized.

The partially completed TURF facility at ORNL is designed to demonstrate all the remote refabrication steps listed above and, perhaps, also the complete reprocessing. This facility is now scheduled for operation in 1974 or later.

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2

5. ENGINEERING REVIEW OF PLANT DESIGN

The plant systems of the 1000-Mw(e) backup and reference designs are generally described in two GA documents[1,2] on the design of twin 1000-Mw(e) HTGR stations. More detailed discussions of design criteria, the designs of components and hardware, and evaluations of system designs particularly applicable to the backup design, except for size, are given in the Preliminary Safety Analysis Report[3] for the Fort St. Vrain 330-Mw(e) HTGR plant. Supplementary information on both the backup and reference designs was also supplied by GGA to ORNL through correspondence, telephone communications, and meetings during the course of this evaluation study. Therefore descriptions of each plant system derived from the above information are presented prior to ORNL's comments on the design.

The backup plant design is based on a scaleup of Fort St. Vrain components, and it is assumed that no further component development beyond that required for Fort St. Vrain and the scaleup of components for the 1000-Mw(e) plant size will be carried out. The proposed backup plant depends on the success of the Fort St. Vrain research and development, design, proof testing, and operating programs, and this review discusses areas where further information on the plant design is required from current development work, licensing decisions, or experience gained in operating the Fort St. Vrain plant. The feasibility of scaling the design of components to the sizes required for the backup plant is also considered.

The reference plant design departs from making sole use of Fort St. Vrain technology and differs from the backup plant design in the following ways:

1. The coolant temperature is increased from 758°F inlet and 1449°F outlet to 803°F inlet and 1524°F outlet, and a supercritical steam cycle is used instead of a 2400-psig steam cycle to obtain a higher thermal efficiency.

2. A wire wrap instead of tendons is utilized for the construction of the prestressed-concrete reactor vessel, and crosshead tendons are eliminated.

3. On-line refueling instead of off-line refueling is employed. This permits the virtual elimination of reactivity shimming and the elimination of controllable orifices.

4. The steam generators are more compact.

Thus the reference design requires initiation of additional development programs for the more advanced designs of some of the plant components. In the following review of the reference plant design features, ORNL outlined special problems to be solved in demonstrating the feasibility of the proposed advanced designs.

5.1 Prestressed-Concrete Reactor Vessel (PCRV)

The reactor vessel for the HTGR is a prestressed-concrete pressure vessel that houses the entire primary cooling system. The major components of the primary cooling system are the reactor core, steam generators, and main helium circulators. Although there are basic differences in the PCRV design for the backup and reference designs, the reactor and other primary cooling system component arrangements are basically the same.

The basic coolant passage arrangement within the pressure vessel is such as to direct the flow of the cooler helium to the passages adjacent to the vessel wall. The reactor core and reflector assembly, consisting of hexagonal fuel and reflector blocks surrounded by a steel core barrel, are supported by graphite blocks and flow-distribution structure. A series of graphite blocks, each supporting seven fuel columns, serves as the support and alignment structure. These support blocks are keyed together radially and are also keyed to the core barrel. Three graphite columns transfer the load from the support blocks to a reinforced-concrete core support floor which, in turn, is supported by a series of columns resting on the bottom head of the PCRV.

Coolant flow, which is downward through the core, converges to the center of the seven-column region within the graphite support block. The region around the graphite support columns serves as a plenum from which coolant is directed through the core support floor in six coolant ducts to the six steam-generator modules located directly beneath the core support floor. After transferring heat to the secondary coolant the helium enters three plenums at the base of the reactor vessel. Each plenum receives coolant from two steam-generator modules and supplies coolant to two helium circulators, which are also located beneath the core. The helium circulators then increase the helium pressure and force the coolant upward around the periphery of the core barrel to the plenum above the core where the flow is directed downward through the core and thus completes the circuit.

The PCRV has penetrations in the top head that house the control rod drives and the helium purification system and penetrations in the bottom head that house the supply and return piping of the six steam generators and the six helium circulators. A central penetration in the bottom head serves as an access penetration for removal of a steam-generator module. Other penetrations are provided for instruments and monitoring equipment.

The prestressed concrete, with a 3/4-in.-thick steel liner as a gastight membrane, serves as the structural component to contain the primary coolant pressure. A very high level of leaktightness is required over the 30-year plant lifetime. Thermal protection of the vessel is provided by a thermal barrier attached to the inner surface of the vessel liner and by cooling tubes attached to the outer surface of the liner. The thermal barrier is not expected to be replaceable and therefore must have adequate reliability to assure a 30-year life. GGA has a continuing program to develop reliable, relatively low cost insulation for the vessel walls. The thermal barrier presently specified for the 1000-Mw(e) designs consists of a number of layers of sheet steel, each approximately 0.020 in. thick, separated by a mesh of 1/8-in.-diam wires spaced on 6-in. centers.* The barrier for the vessel's cylindrical side wall is made in 3.4-in.-thick panels attached to the inside of the vessel liner. A sealing wire on three sides of a panel prevents the development of convection currents adjacent to the vessel wall. The vessel cooling system is made up of redundant water coolant circuits consisting of a series of square tubes attached to the liner prior to placement of the concrete and prestressing members. The

*In the Fort St. Vrain plant, the insulation was recently changed to fibrous material rather than layers of sheet steel.

cooling system is designed to maintain the maximum temperature of the liner and concrete at a safe level of 150°F. Reliability of the thermal barrier and cooling water system is vital to the maintenance of safe concrete temperatures, and appropriate instrumentation is required for monitoring the effectiveness of these systems.

Vessel penetrations are lined with steel that is integral with the vessel liner. Cooling tubes are also provided for removal of heat from penetration liners. Penetrations are provided with double closures, with the primary closure designed in accordance with the ASME Boiler Code, Section III, Class A, for normal working temperature and pressure. In addition, the primary closures are designed so that the primary stress will not be greater than 90% of the yield stress with a pressure differential across the closure equal to the ultimate pressure of the PCRV minus atmospheric pressure. Secondary closures are designed in accordance with the ASME Boiler Code, Section III, Class B, for a pressure equal to the PCRV test pressure.

5.1.1 Backup Design

The PCRV for the backup design has an internal shape of a right circular cylinder 47.8 ft ID by 88.5 ft high. The basic outside shape is a hexagonal prism 76.3 ft across the flats by 136.5 ft high, with pilasters added on the hexagonal corners to provide sufficient bearing area for the circumferential tendons. The prestressing is accomplished by a system of longitudinal, crosshead, and circumferential posttensioned steel tendons.

5.1.2 Reference Design

The PCRV for the reference design has an internal shape of a right circular cylinder 43.5 ft ID by 79 ft high. The basic outside shape is also that of a right circular cylinder and is 70.5 ft ID and 114 ft high. Prestressing is accomplished by a system of longitudinal posttensioned steel tendons and by circumferentially wrapped wire on the outside of the vessel wall. Crosshead tendons are eliminated in the reference design. Removal of the crosshead tendons depends on proving the adequacy of the head design.

Figure 5.1 shows principal dimensions and the equipment arrangement for the backup PCRV design. Design data for both the backup and reference designs are given in Table 5.1.

5.1.3 Evaluation

The prestressed-concrete reactor vessel for the backup design is comparable to the design of similar vessels in Europe, as well as the Fort St. Vrain design.[3] The reference design, however, represents a major departure from the current technology applicable to prestressed-concrete reactor vessels. Table 5.2 (Refs. 4 and 5) shows a comparison of design data for these vessels relative to four European designs and the Fort St. Vrain design.

It is to be noted that the size of the vessel for the backup design is less than that of any of the European vessels based on inside diameter, while the operating pressure is approximately 75% greater. The prestressing tendons are assumed to be of 1000-ton capacity, which is the same as the rating of those employed in the Fort St. Vrain design. By comparison the largest tendons currently employed in Europe are those for the G2 and G3 reactors, which are 1200-ton tendons; however, inherent differences in design may require additional tendon and anchor development for the Fort St. Vrain and 1000-Mw(e) designs. Although the operating pressure is 700 psi compared with 385 psi for the Oldbury reactor, the maximum hoop stress to be eliminated by prestressing is 1600 psi compared with 2500 psi for Oldbury. Construction of the backup design vessel is considered to be well within the current technology employed in Europe.

Similarly the reference design vessel can be compared with European designs as far as size is concerned. The major departure from current technology is in the use of wire wrapping to replace the circumferential tendons. The reference design has a smaller vessel than that of the backup design as a result of the use of more compact steam generators.

The use of wire wrapping for prestressing the PCRV should result in considerable cost savings. However, its use depends on development of suitable equipment to apply the wire to the vessel with a continuously measured tension and instrumentation to monitor prestressing forces during service. Development of the wire-wrapping technique may include the need

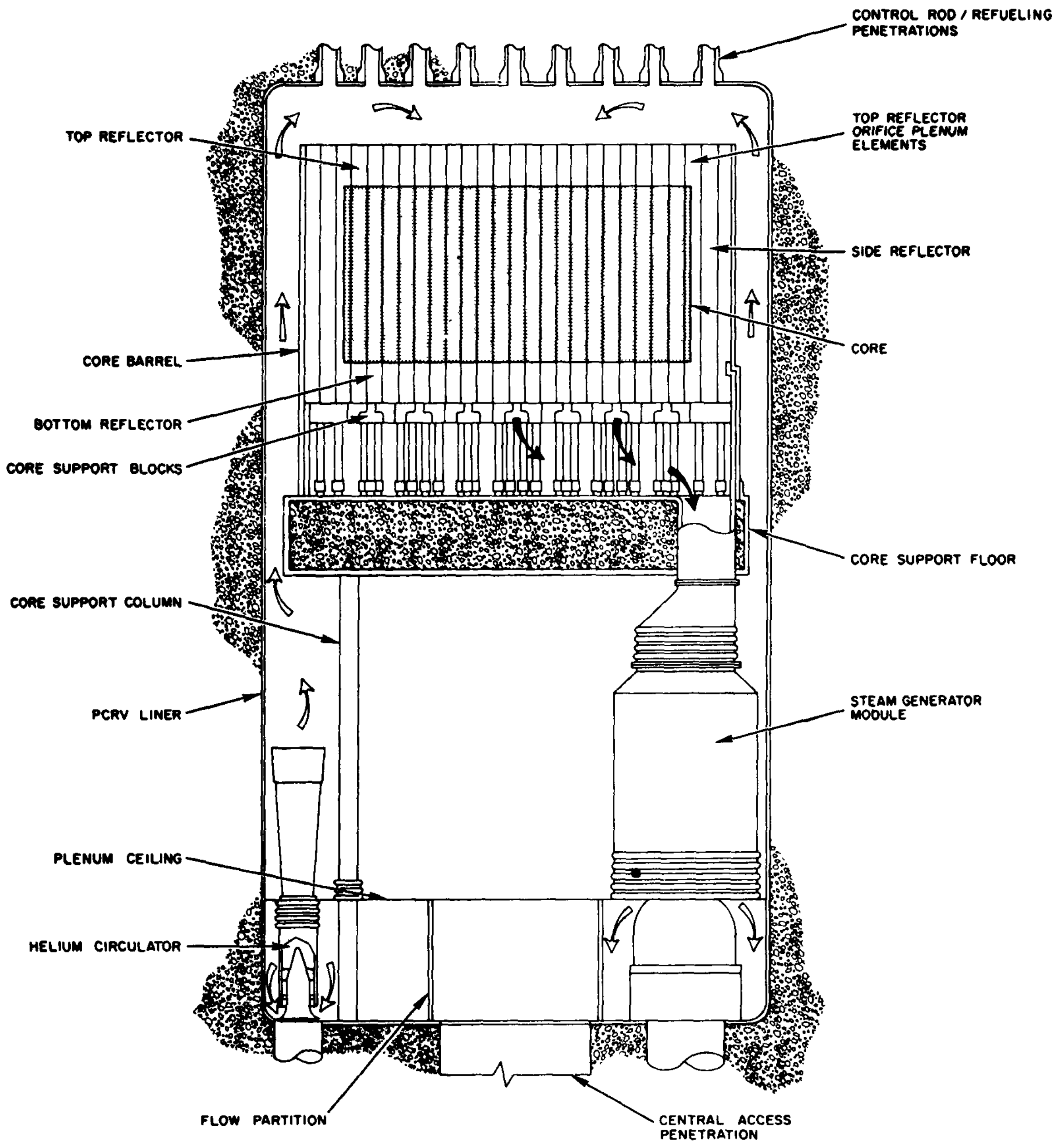


Fig. 5.1. Reactor Arrangement Within PCRV. (GGA illustration)

Table 5.1. Design Data for Prestressed-Concrete Reactor Vessel (PCRV)

Reactor	Backup Design	Reference Design
Internal dimensions (inside liner)	47.8 ft ID, 88.5 ft high	43.5 ft ID, 79.0 ft high
Approximate external dimensions	76.3 ft across flats, 136.5 ft high	70.5 ft diam, 114.0 ft high
Normal working pressure (at circulator discharge	700 psia	700 psia
Peak working pressure	715 psia	715 psia
Average maximum concrete temperature	150°F	150°F
Nominal design temperature difference across PCRV wall and heads	50°F	50°F
Maximum effective fast-neutron (>1 Mev) exposure of liner	2×10^{18} neutrons/cm^2	2×10^{18} neutrons/cm^2
Ratio of reference pressure to peak working pressure	1.2	1.2
Ratio of ultimate pressure to peak working pressure	2.5	2.5
Maximum leakage rate of contaminated helium to reactor building under normal operating conditions	1% of inventory per year	1% of inventory per year
Maximum leakage rate of purified helium to reactor building under normal operating conditions	100% of inventory per year	100% of inventory per year
Type of prestressing system	Posttensioned linear tendons (vertical, circumferential, and crosshead)	Posttensioned vertical linear tendons, circumferential wire wrapping

Table 5.2. Design Data for Prestressed Concrete Reactor Vessels

Reactor	EDF 3	EDF 4	Oldbury	Wylfa	Fort St. Vrain	1000-Mw(e) HTGR backup design	1000-Mw(e) HTGR reference design
Country	France	France	England	England	USA	USA	USA
Reactor thermal power, Mw	1560	1560	834	1875	837	2457	2318
Coolant	CO_2	CO_2	CO_2	CO_2	Helium	Helium	Helium
Coolant operating pressure, psi	440	440	385	385	700	700	700
Coolant temperature, °F							
Core inlet	464	437	473	477	758	758	803
Core outlet	770	752	770	777	1445	1449	1524
Steam generator location	Outside vessel	Below core	Around core	Around core	Below core	Below core	Below core
Vessel shape (outside)	Parallelepiped	Right hexagonal prism	Cylindrical	Five concentric cylinders with 16 vertical ribs	Right hexagonal prism	Right hexagonal prism	Right circular cylinder
Outside dimensions, ft							
Transverse	90 × 90	93.5 top, 102 bottom	90 diam	117 min	49.0 across flats	76.3 across flats	70.5 diam
Height	105	161	104		106	136.5	114.0
Vessel shape (inside)	Vertical cylinder	Vertical cylinder	Vertical cylinder	Spherical	Vertical cylinder	Vertical cylinder	Vertical cylinder
Inside dimensions, ft							
Diameter	62.3	62.3	77.0	95.8	31.0	47.8	43.5
Height	68.9	119.0	60.0		75.0	88.5	79.0
Number and size of penetrations, ft	16 of 7.2 diam 5 of 3.3 diam 211 of 1.5 diam	2 of 1.6 diam 211 of 1.3 diam in top 211 of 1.3 diam in bottom	4 of 8.5 diam	4 of 19.7 diam	39 of 1.6 diam in top 1 of 3.8 diam in top 16 of 3.3 diam in bottom 1 of 6.0 diam in bottom	93 of 1.6 diam in top 6 of 7.3 diam in bottom 6 of 4.4 diam in bottom 1 of 14.25 diam in bottom	93 of 1.6 diam in top 6 of 7.3 diam in bottom 6 of 4.4 diam in bottom 1 of 14.25 diam in bottom
Cable protection	Cement Grout	Cement Grout	Ungrouted	Ungrouted	Ungrouted	Ungrouted	Ungrouted
Liner thickness, in.	1.0	1.0 and 1.375	0.5 and 0.437	0.750	0.750	0.750	0.750
Prestressing members							
Longitudinal	Tendons	Tendons	Tendons	Tendons	Tendons	Tendons	Tendons
Circumferential	Tendons	Tendons	Tendons	Tendons	Tendons	Tendons	Wire wrap
Cross head	Tendons	Tendons	Tendons	Tendons	Tendons	Tendons	None

for one or more PCRV models, as well as development of new fabrication and inspection techniques. The safety analysis will also have to be reviewed. The schedule for the planned introduction of the reference design in 1975 or shortly thereafter is also questionable in light of these requirements.

Thermal protection of the vessel liner and the concrete presents difficult design, fabrication, and installation problems. Since it would be difficult, if not impossible in some locations, to repair or replace the thermal insulation, it is necessary that the thermal barrier maintain its insulating properties and integrity over the 30-year life of the plant. The barrier must withstand pressure and temperature transients experienced in the PCRV with little or no reduction in its thermal resistance. This requires that the barrier and seals remain essentially impermeable to the convective flow of high-temperature helium inside or around the insulating material. It must also resist deterioration in the HTGR environment. There are several candidate materials for the thermal barrier: metals, fibers, and others. The final choice for future HTGR plants will depend on the outcome of a continuing development program.

The hot-face gas temperatures in the HTGR designs (backup, 758°F; reference, 803°F) are comparable to those in Oldbury (770°F). The Oldbury reactor has been delayed because of a problem of coolant gas leakage through the thermal barrier.[6] It is understood that this problem resulted from inferior workmanship during the installation of the barrier. The difficulty was discovered in the vicinity of a penetration where vessel liner temperatures were monitored. Inspection revealed that part of the sealing strips between panels had been omitted during installation. This experience indicates the need for rigid control and inspection of the installation of the thermal barrier. It is felt that the experience gained with the vessels for the Oldbury and Fort St. Vrain reactors should provide for the development of an adequate thermal barrier and suitable fabrication and installation techniques.

The vessel cooling system provides redundant cooling tubes attached to the liner to allow the system to suffer a failure in a single tube and still maintain sufficient coolant capacity. Unfortunately there is no way to repair a leaking cooling tube; however, it appears that the principal consequence of a leak would be just the loss of cooling capacity. Leakage of water into the concrete structure would not damage it, since the prestressing tendons are enclosed within tubes for corrosion protection. Sufficient design margin must be allowed to provide sufficient cooling capacity over the 30-year life of the vessel.

Design of the core support floor presents several rather difficult problems. Thermal protection is even more difficult to provide than the thermal protection of the vessel because coolant gas at its maximum temperature (1449°F for backup design; 1524°F for reference design) must pass through the support floor to reach the steam generators. Thus the thermal insulation and cooling system have to provide a temperature difference of approximately 1350°F in order to adequately protect the concrete structure. Since the concrete is used as a structural material, its protection is vital to the safety of the reactor system.

Providing thermal protection of the support floor around the graphite structures that support the core requires the same technology as that applied to the vessel. Greater thicknesses of thermal barrier will be required. In addition, instrumentation must be provided to monitor the operating temperature of the structural components. For the Fort St. Vrain plant, GGA has shown that for a complete loss of helium circulation, followed by a loss of one-half the core support cooling system, the core support floor will not fail structurally.

Hydrogen formation in the core support structure, which will result from neutron bombardment, will require that the structure be vented to prevent excessive pressure buildup during its 30-year life[7] and to prevent internal pressure buildup due to helium leakage through the structure lining. The size of leak that can be vented and the consequences of a larger leak developing need to be defined. Venting the structure presents other problems, since it is not practical to vent it into the reactor vessel. This requires that the vent be brought outside the vessel, monitored for leakage, and connected to the radioactive gaseous waste system. Design of such a vent system appears feasible but will require close attention to the safety aspects. Details of the vent design were not available for this evaluation study. However, GGA states that the dimensions of this vent are far too small to cause a safety problem.

Another area not investigated by ORNL in detail was the adequacy of the PCRV or its core support structure to withstand earthquake motions. In the present design the core support structure is keyed into the side walls of the PCRV to prevent lateral movement. The Fort St. Vrain plant is in a zone 0 site but is designed for a zone 1 site. According to other design criteria applicable to the backup and reference designs, the plant will be capable of a safe shutdown during and after earthquake motions corresponding to a vertical acceleration of 0.053 g and a horizontal acceleration of 0.08 g.

The reference hypothetical Middletown site is also a zone 1 site as designated by the Uniform Building Code, and thus the Fort St. Vrain criteria would be applicable. It is considered beyond the scope of this evaluation to determine whether these are acceptable earthquake design criteria or whether the design as proposed satisfies the criteria. A number of studies in earthquake design for reactor structures are presently being made, and the information developed in these studies should be applicable to the HTGR design.

In summary, it appears that scaling up the Fort St. Vrain PCRV design for the backup HTGR PCRV can be accomplished. Technology developed for European prestressed-concrete reactor structures will be applicable to the HTGR PCRV, as will technology gained through development work in this country on prestressed-concrete vessels. Close attention to fabrication techniques will be required to assure sound construction and proper strength in areas where access for pouring concrete is difficult, such as around penetration nozzles and at junctions between the vessel heads and side walls. Panel tests to prove the integrity of the thermal insulation for the vessel walls and core support structure will be required, and close inspection and attention to detail during the installation of the thermal barrier will be most important to insure against convection of hot gases against the vessel liner. Redundancy and reliability of the core support cooling system are also very important in avoiding the possibility of overheating the concrete and causing it to deteriorate, since failure of the core support structure cannot be tolerated.

Further development work and analyses will be required to demonstrate the adequacy of certain of the design features of the PCRV for the reference design. In particular, equipment to apply wire to the vessel with a continuously measured tension and instrumentation to monitor prestressing forces during service will require development.[8] The wire-wrapped PCRV should be simpler to construct than present PCRV concepts and should cost substantially less.

5.2 Fuel Handling

The reactor core consists of 91 fuel regions of which 85 contain seven fuel columns and six contain five fuel columns. Each region has a central column containing two control rods. A radial reflector surrounds the reactor core and each fuel column includes reflector blocks on each end that make up the axial reflectors. The portion of the core that requires refueling includes all the 625 fuel columns and the adjacent reflector blocks in both the top and bottom axial reflectors. In addition, the 96 adjacent columns of radial reflector blocks require replacement.

All refueling is accomplished through the 91 refueling nozzles, which also house the control rod drives. The refueling nozzles are 19-in.-ID penetrations through the top head of the PCRV and are spaced on a 37.6-in. triangular pitch.

In both the backup and reference designs, refueling is accomplished by a fuel-handling machine positioned over a refueling nozzle. During refueling the refueling machine serves as a part of the PCRV. An isolation valve provides the vessel closure during removal of the refueling equipment.

5.2.1 Backup Design

The backup design requires an annual shutdown for refueling and operates on a four-year refueling cycle. Refueling is done by core region. A typical core region contains seven fuel columns consisting of 42 fuel elements and 35 reflector elements. The fuel regions around the periphery of the core include the 96 radial reflector columns, which are replaced along with their adjacent fuel columns. Thus each of the 30 peripheral regions includes those reflector columns adjacent to it. This results in 24 fuel regions that contain ten columns (seven fuel and three radial reflector) and six fuel regions that contain nine columns (five fuel and four radial reflector). Based on a four-year refueling cycle, 23 regions would be refueled in an average year (15 regions containing seven columns, six regions containing ten columns, and two regions containing nine columns).

Refueling equipment for the backup design includes two refueling machines, four reactor isolation valves, four fuel storage isolation valves, one auxiliary transfer cask, four reactor refueling sleeves, four fuel storage loading sleeves, two portable jib cranes, and a reactor viewing device. Both new and spent fuel elements are stored in 36 fuel storage wells. The fuel storage capacity will be equivalent to one-third of the fuel and reflector elements.

A refueling machine, shown in Fig. 5.2, consists of a gastight shielded storage magazine, an extendable telescoping boom, a laterally-extendable arm mounted on the boom, and a pickup head assembly attached to the arm. The storage capacity of the fuel-handling machine is about 35 blocks, or approximately one-half the number of fuel and reflector blocks to be replaced in a typical refueling region. Fuel temperature will be maintained at less than 750°F by the cooling system of the fuel-handling machine. Two helium circulation cooling systems are provided on each machine, and each system is capable of removing the maximum design heat load.

The telescoping boom carrying the arm and pickup head is lowered and raised by a counter-weighted chain system, which is enclosed within part of the fuel-handling machine. Positioning of the pickup head assembly is accomplished by the lateral arm and the pickup head's horizontal travel. A probe on the pickup head that engages the central lifting hole in the fuel element locates the head over the fuel element to be removed. Any positioning error causes lateral movement of the probe, which generates an error signal used to correct the position. After correct positioning has been achieved the probe is fully inserted into the fuel block, and a firm grip is established by expanding the probe's stepped collet within the fuel block's stepped hole. The fuel block then acts as part of the pickup head and is lifted into the fuel-handling machine.

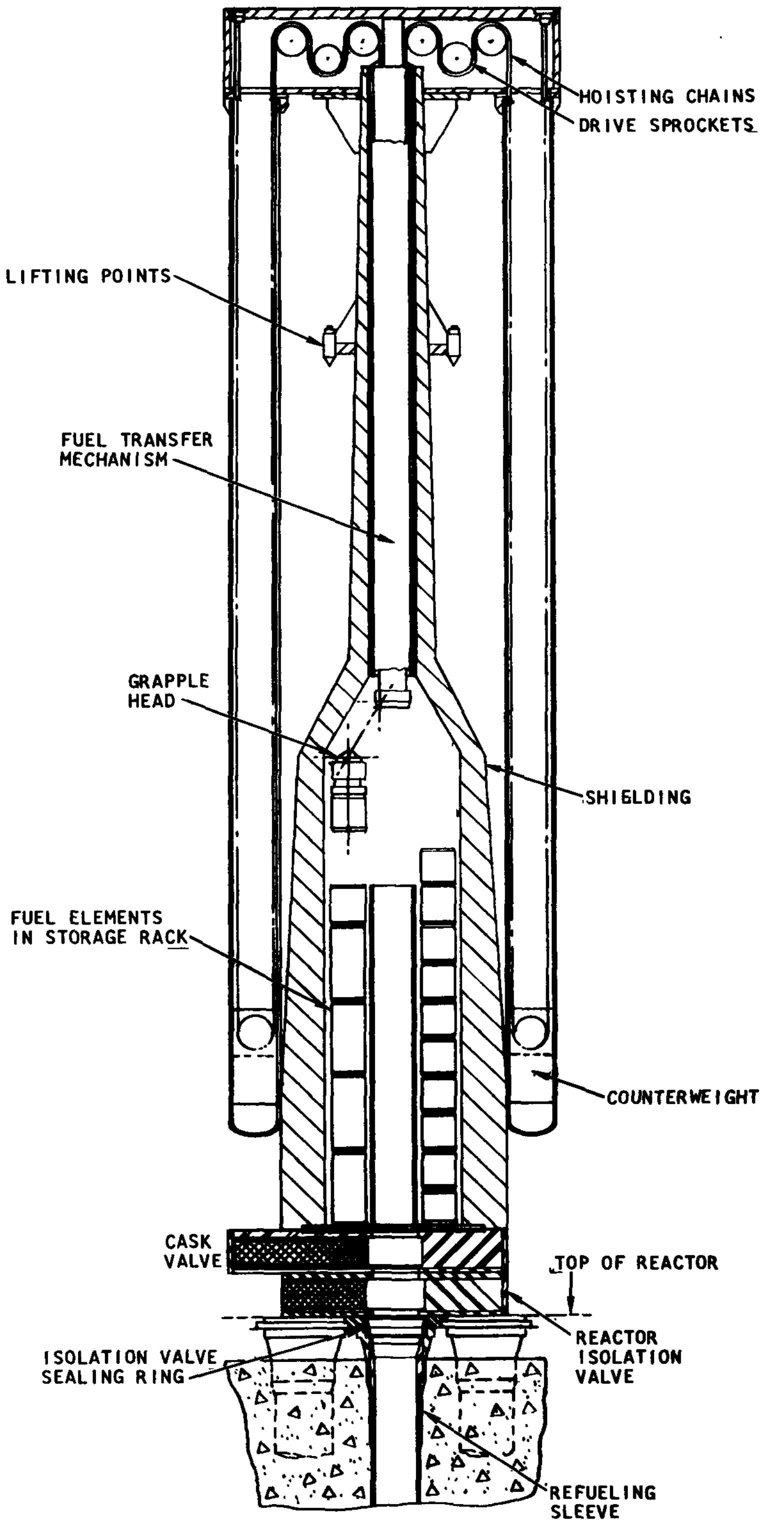


Fig. 5.2. Fuel-Handling Machine for Backup Design. (GGA illustration)

The reactor isolation valves provide the closure for the refueling nozzles during the refueling operation. During use the isolation valve is sealed to the refueling nozzle and provides a sealing surface for either the auxiliary transfer cask or the refueling machine. When either the cask or the refueling machine is in place, the isolation valve can be opened to allow passage of the fuel from the PCRV to the handling machine or of the control rod assembly to the auxiliary transfer cask. The fuel storage isolation valves function in the same manner at the storage wells.

As mentioned above, the auxiliary transfer cask is used for removing the control rod drive assembly and installing the fuel transfer sleeve. When out of the PCRV the control rod drive assembly is stored in a storage well or is transferred to the hot-maintenance facility for repair.

The refueling process is started after a period for reactor cooldown, coolant cleanup, and PCRV depressurization. After completion of the depressurization and removal of the nozzle's secondary closure, an isolation valve is mounted on the nozzle. The auxiliary transfer cask is then positioned on the isolation valve by the overhead crane, and the control rod drive assembly is removed to its storage well. A refueling sleeve is then placed in the penetration by the auxiliary transfer cask to complete the preparation for refueling.

The cask is then removed and the refueling machine is positioned on the isolation valve over the refueling nozzle.

The sequence of operation after the fuel-handling machine is in place is (1) to purge the interspace between the isolation valve and fuel-handling machine closures and to fill it with purified helium, (2) open both the closures, and (3) lower the fuel-handling mechanism into the PCRV. As the mechanism is lowered into the PCRV, the probe on the fuel-handling head enters the top reflector block in one of the six outer columns of the fuel region. After the correct position is obtained the pickup head engages the reflector element and the grasping device is actuated. The element is then removed and placed in the storage rack within the fuel-handling machine. This procedure is repeated, with the pickup head advancing to the next element, until a complete layer of elements is removed from the fuel region. The process is continued to remove complete layers within the fuel region until the fuel-handling machine is filled.

Reversing the sequence of the connection operation allows the interspace to be purged of coolant and filled with dry air, after which the fuel-handling machine transfers the spent fuel to the storage wells. Removal of spent fuel continues until all the fuel is removed from a fuel region, at which time the process is reversed and new fuel is transferred from the storage wells and placed in the reactor. After the new fuel loading is complete, the control rod drive assembly is installed, the isolation valve is removed, and the secondary closure is reinstalled. This completes the refueling for that fuel region.

The ORNL evaluation of the refueling system for the backup design is based on the Fort St. Vrain refueling description[3] and the machine-limited refueling time chart for Fort St. Vrain. The refueling system components for the backup design are the same as those for the Fort St. Vrain design. However, the time chart for Fort St. Vrain is based on using one refueling machine and a six-year refueling cycle, whereas the backup design has two machines and a four-year refueling cycle. The various refueling operations are the same, and the Fort St. Vrain time cycles can be extrapolated for the backup design. Most of the required operation times given for Fort St. Vrain appear reasonable, and there are no significant conflicts of equipment usage. It is noted, however, that the projected operating speed of the fuel-handling machine is higher than most previous experience. Based on this time chart and using one refueling machine the refueling times are 18 hr for a seven-column region, 24 hr for a nine-column region, and 28 hr for a ten-column region. These times do not include installation and removal of the secondary closure, isolation valve, control rod drive assembly, and the refueling sleeve. These operations are performed during the refueling of the preceding and following fuel regions, and thus the installation and removal times must be added for the first and last regions, respectively. This amounts to approximately 8.5 hr. A typical refueling cycle with a single set of equipment would result in a refueling time of approximately 495 hr for the 1000-Mw(e) HTGR backup plant, exclusive of the reactor cooldown time.

The use of duplicate sets of equipment and refueling two fuel regions simultaneously, as would be required for the 1000-Mw(e) backup plant, would be inefficient due to overhead crane availability. The time chart shows that the overhead crane is used, on the average, 63% of the time when a single refueling machine is in operation. This indicates that the second refueling machine could only perform at approximately 65% efficiency, and this would result in a fuel-handling time of 12.5 days. Additional reactor cooldown and pumpdown time results in the reported 14-day refueling time for the backup design.

However, there are times within the Fort St. Vrain refueling cycle that the crane is used almost continuously for as much as 8 hr. Thus the 14-day refueling time could not be achieved by refueling two regions simultaneously. The use of two fuel-handling machines working on the same fuel region would give the best possible fuel-handling time of approximately 13 days. With the addition of cooldown and pumping time the 14-day refueling time allows no time for malfunctioning of equipment or handling of damaged fuel elements. ORNL concluded that only under the most ideal conditions could refueling be accomplished in the 14 days estimated by the designers. These conditions have not usually been present in refueling experience obtained to date. Increasing the storage capacity of the refueling machines by 20 to 30% would improve crane availability and thus the utilization of the second refueling machine.

The importance of maintaining a minimum downtime for refueling depends on the utility's practice for scheduled outages for equipment maintenance and overhaul. Partial turbine-generator inspection and overhaul can be accomplished with an annual shutdown of about four weeks [for each of the 1065-Mw(e) Browns Ferry plants TVA plans an annual outage of 27 days for turbine-generator overhaul of one spindle, at which time refueling and reactor plant maintenance can be carried out].[9] This scheme of operation would not require additional shutdowns for refueling or reactor plant maintenance, and a 0.9 plant availability factor could be achieved if forced outages did not exceed 2.5%. This percentage for forced outages appears reasonable but requires confirmation as operating experience with large reactor systems is accumulated. Under these circumstances a longer refueling time than the estimated 14 days for the 1000-Mw(e) backup design could be tolerated without penalizing reactor availability if the refueling downtime in all cases coincided with the scheduled annual outage.

Another practice would be to schedule major turbine-generator overhauls every three or four years (after the first year's inspection, which would require seven to eight weeks) with an outage of approximately six weeks. Additional outages of about one week per year would be required for reactor equipment maintenance and inspection. The availability loss due to the scheduled outages under this latter plan would then be about 5%, or with 2.5% allowance for forced outages, the potential plant availability factor would be about 0.92. A 14-day refueling schedule would reduce the potential availability about 1.5% and could still be tolerated under this latter plan and be within the ground rules of this study (which require 0.9 plant availability to satisfy an average 0.8 load factor). It is assumed that refueling and scheduled outages would coincide.

As presently designed, the fuel elements have sufficient tolerance accumulation for interference to develop at the interface between two fuel elements. It appears that this problem can easily be eliminated at little or no increase in fabrication cost by changing the dowel design to allow it to be located on a shoulder rather than a threaded surface. The fuel element drawing is shown in Fig. 2.1. Reflector blocks are essentially the same shape as the fuel blocks but consist only of graphite.

The alignment of fuel elements may be a source of concern. Due to the tolerance problems mentioned before and the parallelism of the ends of the elements, some deviation from a vertical stack could occur. The nominal gap between fuel columns within the core is 0.040 in. The characteristics of the stacked fuel columns will be determined in the Fort St. Vrain plant, and the problems for the 1000-Mw(e) designs should be no greater.

Broken fuel elements inside the PCRV present maintenance problems that should be considered in the design of the fuel-handling equipment. The viewing device and some small handling tools used in conjunction with the auxiliary transfer cask are intended to be used in overcoming these problems. No details are available on this type of equipment, but ORNL considers it feasible to design equipment of this type for the removal of broken elements.

5.2.2 Reference Design

The reference design specifies on-line refueling that is carried out continuously with the reactor in operation at full pressure and full power output. Refueling is done by replacing fuel columns, with a single fuel column within a fuel region being replaced each time the nozzle is opened. Refueling is accomplished during one shift with a frequency of one column every two days.

Refueling equipment for the reference design includes one refueling machine, one fuel-loading chute, a reactor-viewing device, and manipulator and small tools for nonroutine handling operations within the PCRV. Both new and spent fuel are stored in 36 storage wells designed to store 20% of the fuel and reflector elements for the reactor. The refueling machine is shown in Fig. 5.3.

The refueling machine consists of a gastight prestressed-concrete cask equipped with a cable-operated grapple head and a refueling chute that can be forced into a curved configuration to guide fuel elements into the outer six columns within a fuel region. The refueling machine has sufficient storage capacity to store two control rod drive assemblies, as well as one load each of new and spent fuel and the refueling chute.

The refueling sequence starts at the fuel storage well, where the refueling machine is loaded with one load of new fuel and a new set of control rods and drives. The machine then proceeds to the appropriate nozzle via rails located between the nozzles. After being sealed onto the nozzle a secondary seal is pressurized, and the closures in both the nozzle and the refueling machine are opened. The control drive assembly is then lifted from the PCRV and stored in the refueling machine, and the refueling chute is inserted into the penetration. The chute is then positioned over the proper fuel column by the grapple head. This is done by attaching the grapple head to the lower end of the chute and lifting upward. The spent fuel column is then removed, one element at a time, and stored in the lower section of the fuel-handling machine. After spent fuel removal is complete, new fuel stored in the upper part of the refueling machine is loaded into the reactor by the grapple head. After loading is completed the refueling chute is removed, the control rod drive

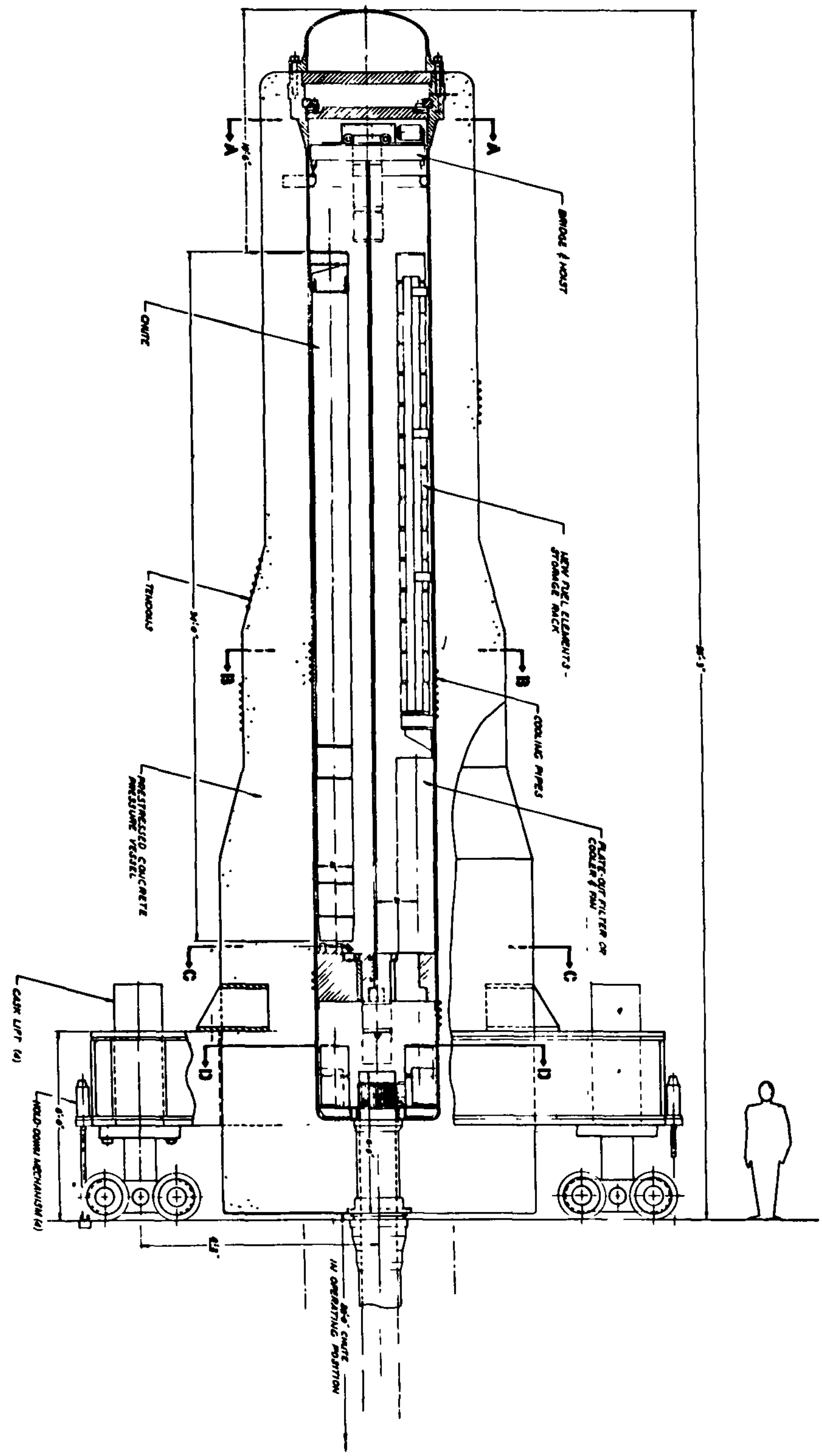


Fig. 5.3. Fuel-Handling Machine for Reference Design. (GGA illustration)

assembly is installed, and the nozzle and refueling machine are closed. The spent fuel is then transferred to the storage wells to complete one refueling cycle.

Evaluation of the on-line refueling concept is rather limited, since the design of the equipment is only conceptual. The ORNL evaluation consisted primarily of outlining the problems associated with the on-line refueling concept. Previous gas-cooled reactors that have used on-line refueling have had discreet fuel element channels in a graphite core into which the fuel elements are inserted. This is the first instance in which the entire core structure is progressively replaced during on-power refueling. One of the problems stems from the location of the fuel columns relative to the refueling nozzles. With the present core design only one fuel column out of seven within a fuel region is located directly beneath the refueling nozzle. Using a single fuel column as the refueling increment requires the removal of a peripheral column within the region. This brings about the use of the curved refueling chute and the cable-mounted grapple head. There is not sufficient detail available to evaluate this design completely. The design would require an extensive development program to determine the operational characteristics and the reliability. A chute of somewhat similar design is used in the British Magnox reactors at similar operating temperatures.

Another problem that may develop in on-line refueling is that the wedging of a fuel column by the surrounding columns due to differential gas pressures may make removal difficult. This problem is not well understood presently and will require development work for quantitative assessment. As a means of counteracting the tendency to wedge, the designers suggest the use of separating forces developed by power-driven rollers attached to the grapple head. The feasibility of this concept for separating the columns can only be determined when more is known about fuel column stability during reactor operation.

Other problems associated with on-line refueling are a result of the opening of the PCRV during reactor operation. This requires that the refueling machine provide high integrity and that the sealing mechanisms be of high quality and reliability. Interlocks must be provided to insure the sequence of operation for opening and closing the nozzles and the refueling machine. Provision for testing leaktightness of connections must be made. In addition, the refueling machine must be provided with the proper restraints to provide protection against seismic loads.

In order to take full advantage of on-line refueling the refueling equipment must possess high reliability and should be capable of being maintained in the available maintenance periods. The use of unitized construction and the availability of spare parts will enhance the maintainability of the refueling equipment. For instance, the grapple head should be designed for easy removal as a unit and a spare head should be provided.

The fuel element for the reference design is of the same shape as for the backup design but is only half as long. This doubles the number of mating interfaces between fuel elements within the core and thus requires that the alignment tolerances be given close attention during the design and fabrication of the elements.

5.3 Primary Heat Transfer System

The function of the primary heat transfer system is to transfer the heat from the reactor core to the secondary coolant system. The system is located entirely within the PCRV and consists of three parallel cooling loops. Each loop includes two steam-generator modules, two helium circulators, and the interconnecting ducting to route the coolant from the reactor outlet plenum through the heat transfer system and back to the annulus around the reactor core barrel.

The coolant is essentially pure, dry helium gas. Coolant flow is described in Section 5.1. The primary coolant system operates at a pressure of 700 psia at the circulator discharge. Helium purity is maintained by withdrawing a side stream for processing by the helium purification system located in the top head of the PCRV. Purified helium is returned to the system principally as a buffer gas for purging the PCRV penetrations.

5.3.1 Helium Circulators

Helium circulation through each of the three primary coolant loops is accomplished by two single-stage axial flow compressors, with each equipped with an annular inlet and an axial diffuser. Each compressor is normally driven by a single-stage steam turbine integrally mounted on the same shaft and operated on cold reheat steam supplied directly from the main high-pressure turbine exhaust. The total cold reheat steam flow passes through the circulator turbines, with each of them receiving one-sixth of the flow. Also mounted on the same shaft is a single-stage water turbine to be used for emergency motive power. The water turbine is supplied with water from the main boiler feed pumps, the condensate pumps, or the firewater system, depending on the nature of the emergency.

The circulators are vertically oriented with the compressor, and the two drive turbines are overhung from a central bearing and seal section. Helium and steam flow upward through the circulator and turbine, respectively. Bearings are of a combined hydrostatic-hydrodynamic type, with water as the lubricant. A system of labyrinth seals operating with purified helium as the buffer gas serves to contain and separate the water, water vapor, and the primary system helium. The major design parameters and the full-load operating conditions for the backup and reference design circulators are compared with those of the Fort St. Vrain circulators in Table 5.3. Circulators for the backup and reference designs are basically a scaleup of the type of circulator employed in the Fort St. Vrain plant, which is shown in Figs. 5.4 and 5.5.

The circulators are located within the PCRV, with the compressor being just above the bottom head and the bearings, seals, steam turbine, and the water turbine located within the PCRV penetrations. A compact pressure casing surrounds the turbine drives, and the steam and water supply and return piping are housed in a series of concentric pipes within the PCRV penetration. Circulator removal is accomplished by lowering the circulator from its operating position directly into a shielded circulator removal cask. Removal would be initiated only during reactor shutdown and after depressurization of the primary coolant system.

Auxiliary equipment associated with the circulator seals and bearings is located outside the PCRV and as close to the circulators as practical. The equipment includes water pumps, water coolers, helium-water separators, helium compressors, and a helium dryer bed.

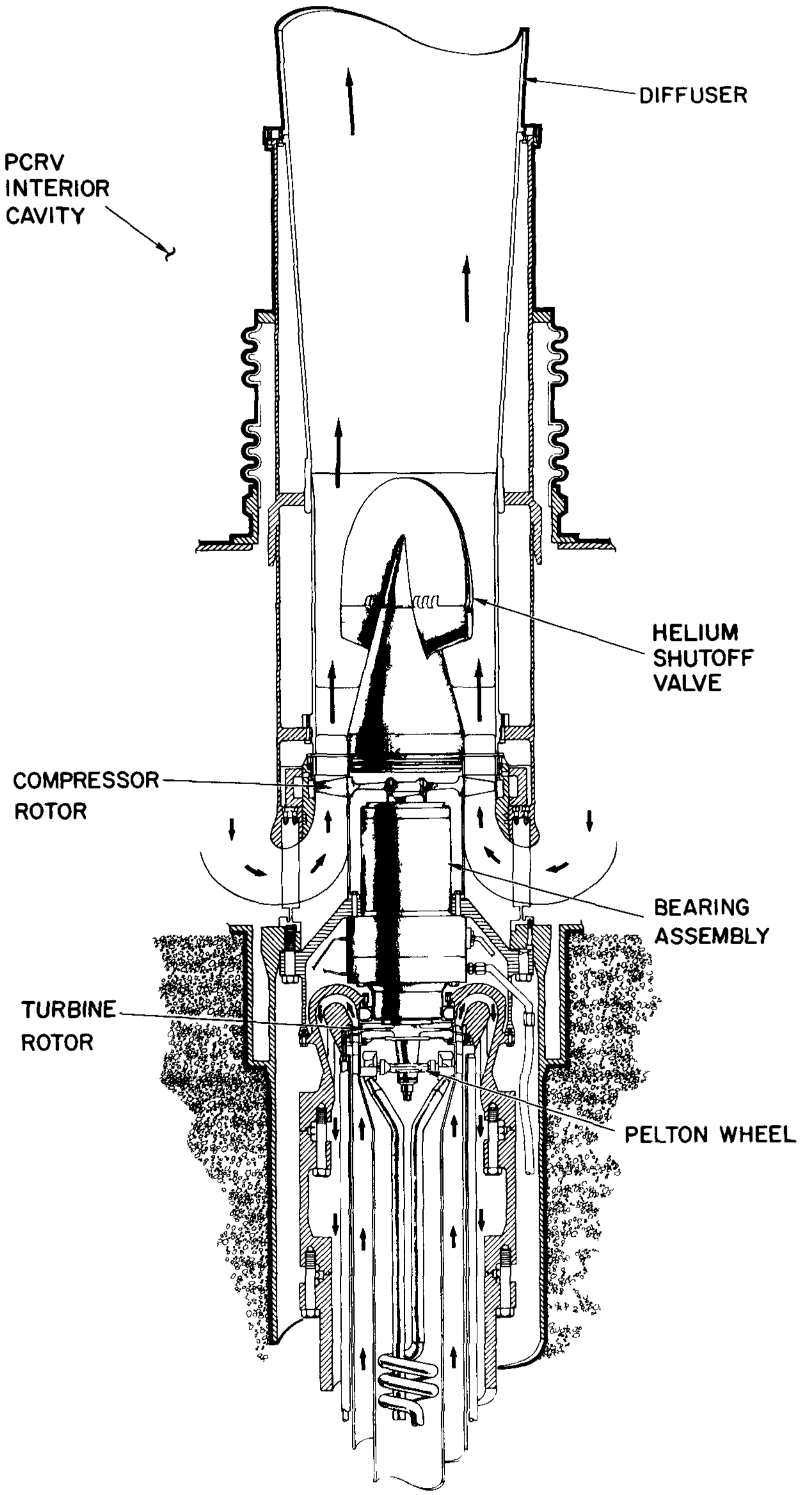


Fig. 5.4. Outline of Helium Circulator. (GGA illustration)

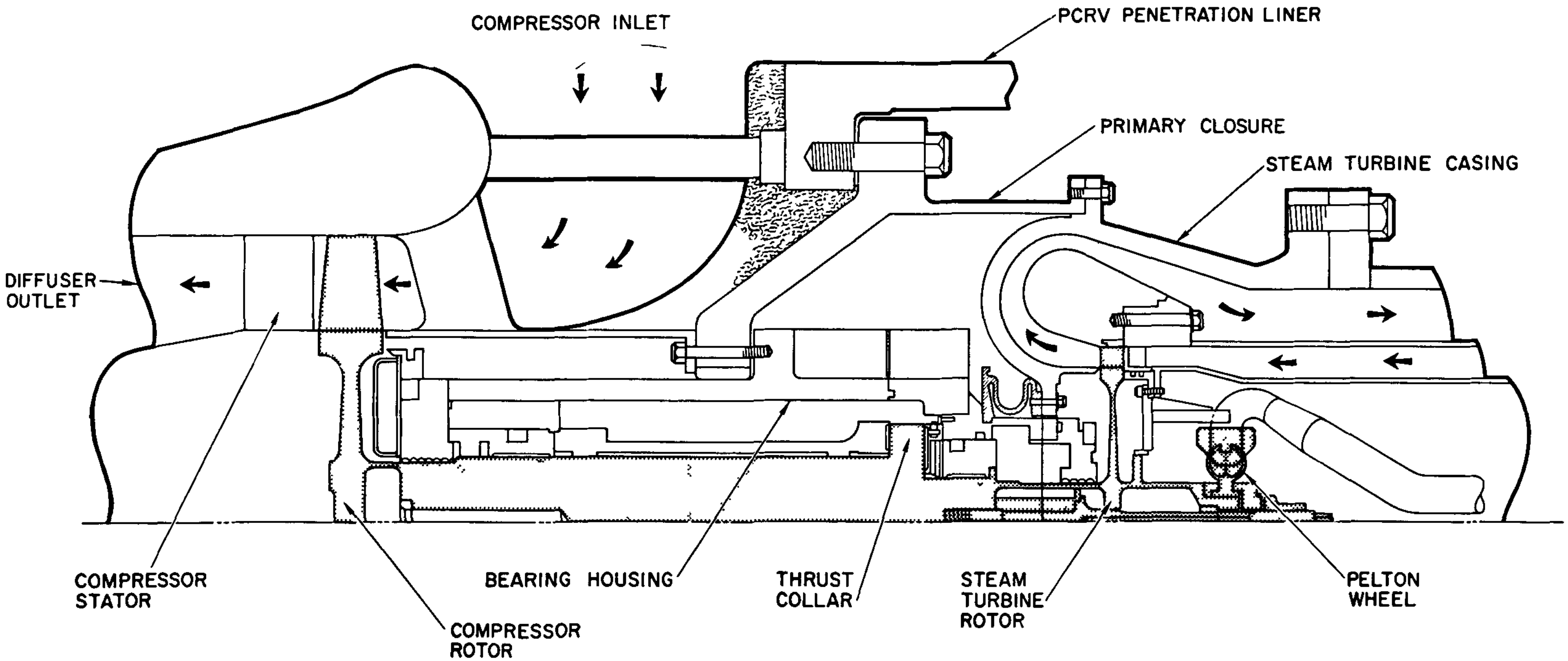


Fig. 5.5. Helium Circulator Cross Section. (GGA illustration)

Table 5.3. Operating Conditions for Helium Circulators

	Fort St. Vrain Design	Backup Design	Reference Design
Total helium flow rate, lb/hr	3.49×10^6	10.27×10^6	9.28×10^6
Number of circulators	4	6	6
Flow rate per circulator, lb/hr	0.873×10^6	1.71×10^6	1.55×10^6
Helium inlet temperature, °F	742	744	793
Helium inlet pressure, psia	686	689	693
Helium outlet pressure, psia	700	700	700
Helium temperature rise, °F	12	10	6
Steam flow (including bypass) per circulator, lb/hr	0.55×10^6	1.11×10^6	0.983×10^6
Steam inlet temperature, °F	738	718	645
Steam outlet pressure, psia	645	645	645
Circulator shaft power, hp	5750	8300	4700

Shutdown and isolation of a helium circulator requires the use of a shutdown seal and an isolation valve. The shutdown seal is actuated by a decrease in lubricant pressure when the circulator is stopped. Isolation of the circulator is accomplished by the isolation valve located in the outlet diffuser. The valve is designed to be held open by the differential pressure developed by the circulator, and it closes automatically when the circulator stops to prevent backflow through the cooling loop.

Evaluation of the helium circulators for the backup and reference designs indicates that the significant development problems associated with these units are essentially identical to those for the Fort St. Vrain circulators. As shown in Table 5.3, the helium flow rates are approximately twice those employed in the Fort St. Vrain circulators. If a similar operating speed is assumed, this corresponds to an increase of approximately 1.4 in the compressor-blade tip diameter, which is considered feasible. Although the Fort St. Vrain circulator development and test programs are directed to provide solutions to many of the development problems associated with the 1000-Mw(e) designs, this does not mean that there will not be problems associated with the 1000-Mw(e) designs. Problems may occur, not only with the circulator design, but also in the cooling loop operating characteristics. Some of the development problems and the Fort St. Vrain programs associated with them are discussed in the following sections.

5.3.1.1 Bearings and Seals. Development of the bearings and seals is currently in progress at GGA's Valmont Test Facility. The planned studies include investigations of the rotor vibrational characteristics and the bearing and seal operational characteristics. Dynamic response of the bearings and the characteristics of the seals under both dynamic and static conditions will be determined. Results of this program should be directly applicable to the design of the 1000-Mw(e) circulators, although some slight increase in shaft size may be required.

Control of the bearing and seal system is critical, and small differential-pressure-level control points are provided. With small pressure gradients for control in the seal and bearing system, difficulty may be experienced in maintaining stability throughout the operating speed range of the circulator. Tests conducted in the Fort St. Vrain program are directed toward establishing the stability characteristics and allowing the necessary conditions to be made.

The use of water as a bearing lubricant is developmental, and the long-term operating characteristics will be investigated during Fort St. Vrain operation. It has attractive features, since it allows use of a single control point at the helium-water interface and a single control system. If oil were used, two control points and two control systems would be necessary: helium-to-oil and oil-to-steam.

5.3.1.2 Loop Isolation. Failure of a steam generator tube or a gas circulator requires the isolation of a coolant loop to prevent backflow through the loop that would divert coolant flow from the core. This is accomplished by an isolation valve which, according to the 1000-Mw(e) designs, functions like a check valve. However, a pneumatically operated valve is reported for the Fort St. Vrain design.[3] With either type of valve the reliability of operation must be established to insure closure upon demand and to avoid inadvertent closures. Load transients developing in the operating loops when one loop is isolated should be thoroughly investigated.

5.3.1.3 System Interactions. A major concern with respect to all gas-cooled reactors that have at least two loops and two axial flow compressors and their associated drives is the dynamic interaction between the circulators. In order to determine the characteristics of parallel-type operation, GGA intends to determine the characteristics of a single compressor operating in a loop equipped with a bypass arrangement at the Valmont Test Facility. The operating characteristics will then be used in computer studies to analyze the dynamic interactions and determine the control requirements.[10] Computer studies are limited, however, by the assumptions made in formulating the programs. Thus the dynamic interaction response and control requirements can only be completely determined and evaluated in the operation of parallel loops that simulate the complete coolant loop flow characteristics. This in essence requires simulation of the complete system, which may be considered to be economically prohibitive. The solution to such development problems can be obtained only in the Fort St. Vrain plant at power.

5.3.1.4 Emergency Operation. Removal of decay heat requires the operation of the helium circulators, which are driven by water turbines supplied with feedwater from various sources. Control of the emergency operation may be critical due to temperature gradients and the resulting thermal stresses. During decay heat removal, restriction of helium flow is required to avoid reducing the helium temperature to an undetermined critical level, since feedwater flow is available from at least three sources and its temperature may vary from a low of ambient temperature to a high of approximately 300°F. The combination of any of these variations and the need for accommodating sudden changes in the mode of operation require that a detailed analysis of the operating characteristics be made. Operating temperatures and possible transients should be determined to allow proper design of an adequate control system.

5.3.1.5 Reliability and Maintainability. Every effort must be made to maximize the reliability and maintainability of the gas circulator units. Throughout the development programs attempts should be made to determine the reliability of the various components and to evaluate alternate materials and/or designs from the standpoint of operating life and ease of maintenance. The effect of abnormal operation on the reliability of the machine must be considered as it may result in a change in operating philosophy.

In summary, it is recognized that the circulators are rather unique in that the compressor, steam turbine, and auxiliary water turbine are located on a single shaft. No known unit of similar design is currently in operation. The design appears to be quite attractive for reactor operation in that it is compact and is well suited for use within the PCRV penetration. Following successful completion of the Fort St. Vrain development programs, the 1000-Mw(e) circulator design, being a reasonable extrapolation, would evolve in logical order with some additional development effort and full-scale testing in a test stand. The importance of the circulators' reliability may necessitate provision of a spare unit to limit reactor downtime and to facilitate periodic inspection and maintenance.

5.3.2 Steam Generators*

Each of the primary coolant systems contains one steam generator, which consists of two identical modules. A module includes three basic tube bundles: a reheater section, a superheater section, and an economizer-evaporator section. The modules are located below the core support floor, as shown in Fig. 5.1. The feedwater supply, main steam outlet, cold reheat supply, and hot reheat outlet are housed within a concentric pipe arrangement such that all the supply and return lines for one module can be accommodated within a single PCRV penetration. Helium enters the top of the module from a duct through the core support floor and flows outside the tubes perpendicular to the tube axis. The design is basically that of a once-through boiler with integral superheat and a reheat section arranged in counterflow fashion so that the hot helium passes through the reheat section, the superheater, and finally the economizer-evaporator

*The term "steam generator" is used to designate the heat exchanger unit that contains economizer-evaporator-superheater I, superheater II, and reheater sections.

section. A cylindrical shroud encloses the tube bundles and is connected by ducting to the reactor outlet plenum above and the circulator inlet plenum below. Connections of the module to the entrance duct are not leaktight; therefore, some of the higher pressure and cooler helium being discharged from the helium circulators infiltrates through the connection and results in an entering helium temperature that is somewhat less than the core exit temperature. The major design parameters and full-load operating conditions for the steam generators are given in Table 5.4.

In each steam generator the two modules have a common header outside the PCRV and operate as a single unit. The feedwater header to each steam generator has two feedwater isolation valves in series that are located upstream of the feedwater control valve. Trim valves are provided in the feedwater branch lines that feed each module. The superheat steam header is equipped with a stop valve and a check valve to allow isolation of one steam generator from another. Similar valves are provided on the hot reheat header. Isolation of the cold reheat header is provided by the turbine overspeed trip valve on the helium circulator.

A steam and water dump system is provided for each steam generator. Excessive moisture indication in a primary loop results in a reactor scram, shutdown of the defective loop, and dumping of the contents within the steam generator in the defective loop to a dump tank. The maximum time required to empty the steam generators after sensing a tube failure is 15 to 20 sec.

5.3.2.1 Backup Design. The steam-generator module for the backup design consists of three tube bundles, with the tubes arranged in a series of concentric helical coils with a vertical axis. Flow in the superheater is cocurrent with helium flow, while the flow in the reheater and economizer-evaporator bundles is countercurrent to the helium. The arrangement is such that the helium which flows downward through the tube bundles also flows across a matrix of in-line tubes. Tube bundles are stacked vertically, with the economizer-evaporator bundle situated in the bottom, the superheater bundle directly above, and the reheater in the top position. Each tube has its entire length contained within a single circumferential layer, with the helix angle being varied in an attempt to minimize the difference in tube length for the range of concentric

Table 5.4. Steam-Generator Design Parameters for One of Six Modules in 1000-Mw(e) HTGR Plant

	Fort St. Vrain Design, Axial Flow[a]	Backup Design, Axial Flow	Reference Design, Radial Flow
Total surface area, ft^2	3,738	21,200	17,100
Active tube bundle diameter, ft	5.3	13.7	9.5
Active tube bundle length, ft	12.9	16.0	14.3
Total unit diameter, ft	5.5	13.7	10.5
Helium flow rate, lb/hr	0.284×10^6	1.664×10^6	1.510×10^6
Feedwater and main steam flow rate, lb/hr	0.192×10^6	1.124×10^6	1.121×10^6
Reheat steam flow rate, lb/hr	0.187×10^6	1.108×10^6	0.993×10^6
Helium inlet pressure, psia	670	691	697
Feedwater inlet pressure, psia	3,017	3,035	4,135
Reheat steam inlet pressure, psia	636	636	636
Total helium pressure drop, psi	4.4	2.4	3.5
Total feedwater and main steam pressure drop, psi	505	520	520
Total reheat steam pressure drop, psi	36	36	36
Reheater			
Helium inlet temperature, °F	1427	1427	1501
Helium outlet temperature, °F	1331	1327	1369
Steam inlet temperature, °F	675	669	612
Steam outlet temperature, °F	1000	1002	1052
Helium pressure drop, active bundle, psi	0.49	0.37	0.83
Steam pressure drop, active bundle, psi	17.3	12	17
Tube design temperature, °F	1400	1400	1350
Tube outside diameter, in.	1.125	1.125	1.125
Tube wall thickness, in.	0.125	0.134	0.13
Tube arrangement	In-line	In-line	Staggered
Tube pitch, in.	1.58×1.58	1.48×1.33	1.41
Surface area, including excess, ft^2	448	2,650	2,670
Number of tubes	84	528	340
Tube length, ft	18.4	17	24
Frontal area, ft^2	17.2	112	179
Tube material	Incoloy	Incoloy	Incoloy
Superheater			
Helium inlet temperature, °F	1331	1327	1369
Helium outlet temperature, °F	1192	1196	1212
Steam inlet temperature, °F	750	745	802

[a]One of 12 modules.

Table 5.4 (continued)

	Fort St. Vrain Design, Axial Flow[a]	Backup Design, Axial Flow	Reference Design, Radial Flow
Superheater (continued)			
Steam outlet temperature, °F	1005	1005	1054
Helium pressure drop, active bundle, psi	0.98	0.82	0.48
Steam pressure drop, active bundle, psi	150	118	160
Tube design temperature, °F	1200	1200	1150
Tube outside diameter, in.	1.00	1.00	0.75
Tube wall thickness, in.	0.200	0.200	0.151
Tube arrangement	In-line	In-line	Staggered
Tube pitch, in.	1.48 × 1.44	1.58 × 1.44	0.875
Surface area, including excess, ft^2	837	4,080	3,860
Number of tubes	54	324	465
Tube material	Incoloy	Incoloy	Incoloy
Number of subheaders	18	108	93
Tube length, ft	60.3	48	42
Frontal area, ft^2	14.6	91	204
Economizer-evaporator			
Helium inlet temperature, °F	1192	1195	1212
Helium outlet temperature, °F	741	744	793
Feedwater inlet temperature, °F	410	410	510
Steam outlet temperature, °F	750	745	802
Helium pressure drop, active bundle, psi	1.65	0.93	1.73
Steam pressure drop, active bundle, psi	117	83	126
Tube design temperature, °F	1006/870/750	1000/880/840	950/900/850
Tube outside diameter, in.	1.00	1.00	0.75
Tube wall thickness, in.	0.225/0.138/0.138	0.220/0.148/0.148	0.159/0.147/0.140
Tube arrangement	In-line	In-line	In-line and staggered
Surface area, including excess, ft^2	2,453	14,500	10,600
Number of tubes	54	324	465
Number of subheaders	18	108	93
Tube length, ft	170	170	117
Frontal area, ft^2	17.6	114	197/151/161
Tube pitch, in.	1.42 × 1.47	1.75 × 1.44	1.073 × 1.073/0.875
Tube material	Carbon and low-alloy steel	Carbon and low-alloy steel	Carbon and low-alloy steel

layers of tubes. The tube bundle design is essentially identical with that employed in the Fort St. Vrain plant, which is shown in Fig. 5.6.

Feedwater is supplied to the economizer-evaporator unit at 410°F and 3035 psia through 108 subheaders, each of which supplies three of the 324 total tubes. Feedwater subheaders are arranged in a concentric circle surrounding the other supply and return lines within the PCRV penetration. Secondary coolant flows upward through the economizer-evaporator section, where it is evaporated and receives some superheating. Leaving this section the coolant flows through a connecting tube to enter the top of the superheater section. The connecting tube includes a bimetal joint where the carbon steel economizer-evaporator tube is joined to the Incoloy superheater tube. Flow is downward through the superheater bundle, and the superheated steam leaves the PCRV at 1005°F and 2515 psia through 108 subheaders arranged in a concentric circle inside the feedwater subheaders.

Cold reheat steam enters the PCRV at 669°F and 636 psia through an annular header that extends through the center of the economizer-evaporator and superheater bundles. Flow through the reheater is upward and the hot reheat steam leaves the PCRV at 1002°F and 600 psia through a pipe located within the reheat supply annulus.

An excess heat transfer area of approximately 10% is provided to allow for the plugging of leaky tubes. Tube plugging is accomplished by plugging the subheader that supplies feedwater to and returns steam from three tubes in the once-through boiler region. This can be done outside the PCRV. Reheater tubes that develop leaks are plugged individually by a remote tube-plugging machine which operates within the reheat headers. All tube-plugging operations are performed with the plant shut down and depressurized.

Removal of the steam-generator module is possible by separating the module from the penetration assembly, lifting upward, and moving the module radially inward to the central access penetration, where the module can be removed from the PCRV. Replacement of the module can be performed by the reverse procedure.

5.3.2.2 Reference Design. The reference design steam-generator module, which consists of three tube bundles, operates at supercritical

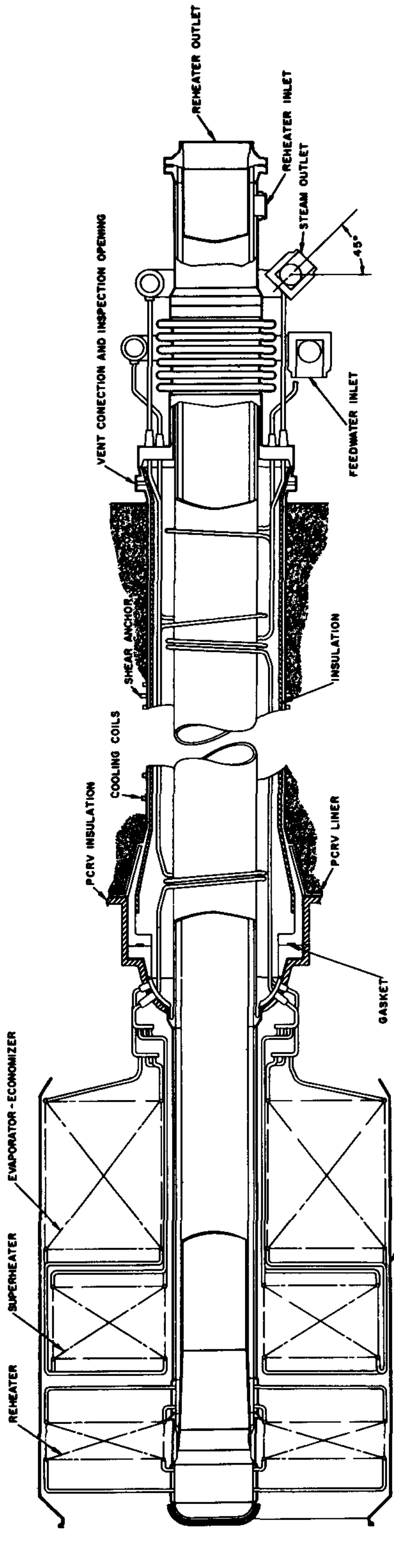


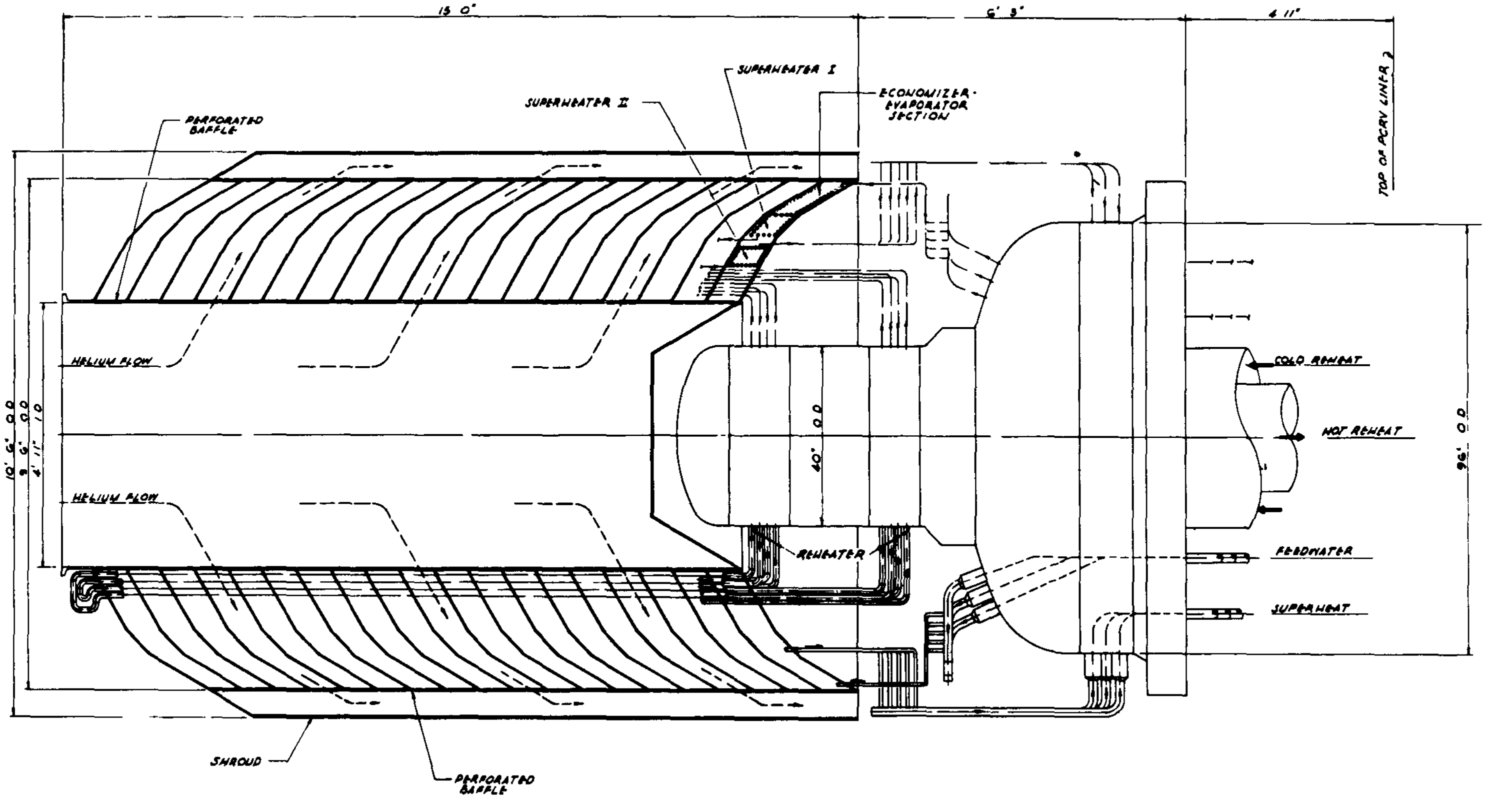
Fig. 5.6. Steam-Generator Module for Backup Design. (GGA illustration)

pressure in the steam-generating section, where supercritical steam is generated at 1052°F and 3615 psia. The so-called "economizer-evaporator" section includes sufficient heat transfer area to increase the temperature of the fluid to above the critical temperature, which is the reference point for measuring superheat. Flow through the economizer-evaporator and reheater sections is generally countercurrent with the helium and cocurrent in the superheater section. Helium flow enters the module from the top and flows radially outward through the annular tube bundles and crosses the reheater, superheater, and finally the economizer-evaporator section in a direction perpendicular to the tube axes. A schematic drawing of a reference steam-generator module is shown in Fig. 5.7.

The reheater tube bundle consists of 340 inverted U-tubes. The steam enters the outer leg and flows upward and then downward in the inner leg to produce generally countercurrent flow. Reheater tubes are nested together in a closely packed triangular array with the spacing maintained by a tube sheet and the conical support plates through which they pass. Cold reheat steam at 612°F and 636 psia enters through an annular header that terminates beneath the tube bundles. Hot reheat steam at 1052°F and 600 psia leaves through a central pipe within the annular cold reheat header.

The economizer-evaporator tube bundle consists of 465 identical, conical, spiral-tube coils stacked vertically. Water and steam flow are inward to produce countercurrent flow. A 60° cone angle provides a closely packed triangular tube array in the first region of the economizer-evaporator section, but the cone angle is changed to 45° to provide a square-pitch array in the superheat section of the economizer-evaporator. Tubes leaving this section are routed to the surface of the superheater section, where the tubes spiral outward to produce cocurrent flow. A 30° cone angle in the superheater section provides a closely packed triangular tube array.

Feedwater at 510°F and 4135 psia enters the PCRV through 93 subheaders, each of which feeds five tubes in the once-through steam-generating section. Supercritical steam at 1054°F and 3615 psia leaves the superheater section in 93 subheaders.


13' 0"
6' 3"
4' 11"
SUPERHEATER I
SUPERHEATER II
ECONOMIZER-EVAPORATOR SECTION
PERFORATED BAFFLE
TOP OF PCRV LINER
HELIUM FLOW
HELIUM FLOW
10' 6" O.D
9' 6" O.D
4' 11" I.D
40" O.D
96" O.D
COLD REHEAT
HOT REHEAT
REHEATER
FEEDWATER
SUPERHEAT
SHROUD
PERFORATED BAFFLE

The maintenance philosophy for the reference design steam generator is assumed to be identical to that described in the preceding section for the backup design. However, the reference design only allows a design margin of 5% in heat transfer surface.

5.3.2.3 Steam-Generator Evaluation. Evaluation of the steam-generator designs consisted of performing the necessary calculations to confirm the adequacy of the design and reviewing the fabrication, operating, and maintenance problems associated with the designs. The following is a discussion of the adequacy and the problems associated with the backup and reference design steam generators.

1. Adequacy of Design. In attempting to confirm the adequacy of the heat transfer areas provided, the first step was to consider the calculational techniques employed by the proponent. It is understood that the GGA computer program is based on data developed by Grimison[11] for the flow of air across tube banks, with the application of the necessary correction factors to allow the data to be used for helium. The computer program consists of a stepwise technique that divides the tube lengths into equal increments and maintains an energy balance along the length of the tube.[12] This appears to be basically a good approach and should lead to a rigorous technique as more experimental data are obtained and factored into the program.

In evaluating the heat transfer surface requirements for the steam generators, ORNL first performed hand calculations by using overall conditions for the various sections of the steam generators. Later, a computer code was developed for the calculations in which overall conditions for each section were still used instead of the stepwise technique used in the GGA computer program. Therefore the ORNL calculations were limited in this respect; however, both the GGA calculations and ORNL's were limited in accuracy by the lack of experimental data on helium flowing across tube banks. Most correlations are based on data for air, with suggested corrections for helium properties. In the reference design the problem is compounded by the fact that the heat transfer data for supercritical steam are also rather limited.

In the calculation of heat transfer surface areas for the backup design, two correlations of the outside helium film coefficient were used.

The first, Method A, applied the correlation as given by Colburn,[13] and the second, Method B, applied a modification of the correlation by Grimison as suggested by Knudsen and Katz.[14] In both cases, inside coefficients were calculated by using the Dittus-Boelter correlation for water and correlations recommended by Heineman[15] for steam. The calculations were done for five sections of the module by utilizing average temperatures for properties and a logarithmic mean temperature difference. The results of these calculations are compared with the reported values in Table 5.5. The ORNL calculations include an allowance of 10% excess area for tube plugging, which is the same as that allowed by GGA.

Table 5.5. Comparison of ORNL and GGA Calculated Values of Heat Transfer Areas for the HTGR Backup Design

Section	ORNL Calculated Area[a] (ft^2)		GGA Reported Area (ft^2)	Apparent Margin[b] (%)
	Method A	Method B		
Economizer-evaporator-superheater I	16,830	13,020	14,500	+11.4
Superheater II	5,080	3,980	4,080	+2.5
Reheater	2,680	2,450	2,650	+8.2

[a]Includes 10% excess allowance for tube plugging.

[b]Apparent design margin is based on Method B.

Comparison of the areas reported by GGA with those calculated indicates that the reported areas would be inadequate relative to those obtained by Method A but would allow sufficient design margin relative to those obtained by Method B. It is further noted that Method A is approximately 20% more conservative than Method B. The deviation of 20% is not uncommon when comparing different correlations for film coefficients of gases flowing across tube banks, as is the case for the steam generators.

It is concluded that the reported heat transfer areas are reasonable, since they are between the two calculated values, which differ by approximately 20% except for the reheater. Deviations of this magnitude point out the need for a comprehensive development program to determine heat transfer correlations more exactly for the geometry and conditions of the Fort St. Vrain and backup design heat exchangers. The first confirmation of the heat transfer correlations will come during operation of the Fort St. Vrain units at power in the plant.

The reference design was evaluated by using the Knudsen-Katz modification of the Grimison correlation, as given in Method B, to calculate the outside film coefficient. The inside film coefficients were determined from the Heineman correlation.[15] GGA used the Colburn equation for steam and water coefficients with properties evaluated at the bulk temperature for both the reference and backup designs. The results of these calculations and a comparison of the reported data are given in Table 5.6. From the comparison, it appears that the areas are adequate; however, because of the uncertainty of the correlations, it is recommended again that experimental correlations with helium be obtained with the tube arrangement proposed for the design. The tube wall stresses were calculated, and in all cases the stress was found to be less than that allowable, as given by Section III of the ASME Pressure Vessel Code.

Table 5.6. Comparison of ORNL and GGA Calculated Values of Heat Transfer Areas for the HTGR Reference Design

Section	Calculated Areas (ft^2)		Apparent Margin (%)
	ORNL[a]	GGA[b]	
Economizer-evaporator	11,060	10,600	−4.2
Superheater	3,880	3,860	~0
Reheater	2,510	2,670	+6.1

[a]Obtained with the modification of the correlation of Grimison, as suggested by Knudsen and Katz; includes 10% excess area for tube plugging.

[b]Obtained with relationships based on data used by Grimison and additional data proposed in Ref. 16; includes 5% excess area for tube plugging.

2. Flow Stability. Flow stability is of major concern in a once-through boiler. The large variation in specific volume with heat addition plus the highly turbulent condition when the two phases are mixed produces a possibility that the flow rate through certain tubes could vary widely from the flow through other tubes.

The backup design has trim valves to adjust the flow for three tube circuits that are assumed to have approximately the same length and bend radii. Ability to adjust the flow for three tubes should help to eliminate flow instabilities. Also, a once-through boiling test loop is planned for the Fort St. Vrain design. Unfortunately, the loop will only test a single coil, which may not give a satisfactory definition of the overall stability problem.

Flow stability or instability can be predicted only after sufficient tests are conducted on a representative section of the steam generator that simulate the variations in pressure drops and heat transfer which will occur in the actual bundle. The first test of flow stability in representative sections of a steam generator will be during power operation of the Fort St. Vrain plant.

3. Tube Vibration. Tube vibration problems are of serious concern in both the backup and reference design steam generators. Tube installation in the backup design requires the tube to be installed by a twisting motion that advances the tube, similar to the advancing of a conventional screw thread. This requires substantial clearance in the tube sheet to allow the helically coiled tube sheet to be installed. A simple method of securing the tubes within the tube sheet is not obvious. GGA is considering the use of wedges to restrict the motion of the tube within the tube sheet. Installation of wedges is a time-consuming operation, and predicting their behavior during thermal cycling is difficult. The Fort St. Vrain steam generators are similar to the backup design, and a testing program is planned to study the vibrational problems.

Tubes for the reference design are separated by spacer strips. Although design details are incomplete as far as mechanical design and fabrication are concerned, it appears that the tubes can be held more securely than in the backup design. Another vibrational problem that may be encountered in the reference design is the so-called "whistle" phenomenon,[17] which consists of the generation of sounds. Boiling generates the sound, and during subcooled boiling and supercritical pressure operation at high heat flux levels, the intensity of the generated sounds can result in mechanical vibration, which has been reported to have caused damage and failure of test sections. It was not possible for ORNL to assess the likelihood of the whistle phenomenon occurring in the reference design. Study of this problem would require testing of a typical section of the reference steam generator.

4. Steam-Generator Isolation and Emergency Operation. Reactor safety requires that a steam generator module be isolated in the event of a tube rupture and also that part of the steam generator modules serve as a decay heat removal system for reactor shutdown. Controls for the various modes of operation must provide the necessary interlocks to prevent damage of the steam generators during emergency conditions. For instance, rapid thermal cycling of the steam generators may result in a loss of integrity. Damage could occur if the steam generator accidentally drained during power operation. This is only one of the possible means of introducing sources of damage if the controls of the emergency systems malfunction.

5. Steam-Generator Maintenance. A single tube leak in the steam-generating section of a module requires the plugging of three or five tubes in the backup and reference designs, respectively. Thus a single tube leak in this section requires the plugging of approximately 0.15% of the area provided by the six modules in the backup and reference designs, respectively.

The situation could be more serious if the failures were not random among the modules. Since the only provision for regulating the helium flow to the steam generators is the circulator speed, this requires two modules in parallel to operate at essentially equal power. Thus substantial failure within a single module might result in an equal reduction of capacity in the parallel module.

The tube-plugging operation in the steam-generating section appears quite feasible, since the subheaders are outside the PCRV and therefore accessible. Reheater tube plugging requires the use of sophisticated equipment and remote techniques because the plugging operation must be performed within the PCRV. Thus the equipment must be designed to operate through the inlet and outlet headers. Similar equipment will be required in the Fort St. Vrain reactor, and programs are planned for developing and testing this equipment. These programs will be applicable to the backup design.

Provision for replacement of a steam-generator module is essential. Replacement is done by severing the connection between the module and the inlet header and moving the module radially inward to the central access penetration, where it is removed. The operations require specialized remote maintenance equipment and portable shielding to perform the replacement operation inside the PCRV. The Fort St. Vrain plant will require performance of the same functions. However, the 1000-Mw(e) plant has steam generator modules about six times as large as the Fort St. Vrain units. Further development of the remote procedures and equipment may be required for the 1000-Mw(e) units.

5.4 Plant Control Mechanisms and Systems

The control mechanisms, control systems, and instrumentation that are unique to the HTGR design and are needed for orderly operation are discussed in this section. The safety aspects of these systems are discussed in Chapter 6, along with the systems that are essential for the protection of the plant or the public.

5.4.1 Control Rods and Drives

The control rod systems are essentially the same for the Fort St. Vrain, backup, and reference designs. The reactor is controlled by selective movement of 182 control rods, which operate in pairs. Reactivity margin is sufficient to allow shutdown from the most reactive condition of the core with any two control rod pairs failed in the withdrawn position. Each control rod consists of 11 cylindrical absorber sections connected by a central spine to form an assembly that is approximately 16 ft long. The absorber sections consist of a pair of concentric steel alloy sleeves (outer 3.34 in. ID; inner 2.05 in. ID) joined by a welded steel cap at the ends. In the interspace between the sleeves there is a boron-carbide-filled graphite compact containing 40 wt % boron.

The upper end of the central spine is attached to the suspension cable. If the suspension cable fails, a crushable tubular structure will serve as a shock absorber on the lower end of the control rod. The control rods are cooled by the flow of coolant between the rod and the surrounding fuel element.

Control rods are operated in pairs by winch-type devices located in the PCRV top head. A pair of control rods is raised or lowered by the winding or unwinding of cables from a duplex drum, which is driven by a three-phase electric motor. The drive is equipped with a brake that is released by deenergizing a dc solenoid. The brake is released when the drive motor is energized or upon reactor scram. The motor is disconnected from the ac power during a scram. The scram velocity is limited by operation of the motor as a generator with capacitive excitation. This requires sufficient residual magnetism in the motor rotor to build up the ac terminal voltage. All control rods are scrammed upon receiving the scram signal.

The performance specifications for the rod drives in the Fort St. Vrain design were modified in Amendment No. 1 of the PSAR.[3] ORNL assumed that this modification represented the latest concept for this type of drive and would be used in the 1000-Mw(e) designs. The revised specifications are listed below:

Number of control rod drives	91
Average shim velocity, in./sec	
Withdrawal	1.05
Insertion	1.10
Average scram velocity, in./sec	1.25
Scram time, sec	152
Maximum time to reach constant velocity, sec	
Shim	<0.5
Scram	<1.0

The rod drives are situated within the PCRV top head, which is relatively cool, and no additional cooling is required. Cooling is provided by the penetration cooling system, with the drive temperature being maintained at 150 to 180°F. Thermal and radiation shields are provided within the penetrations to prevent the transport of excessive heat or radiation to the rod drives. Helium purge flow is provided to prevent the migration of radioactive materials into spaces above the lower control rod drive shields. The space between the top of the reactor core and the inside of the top head of the PCRV is spanned by control rod guide tubes that are spring-loaded downward against the top reflector blocks.

In the backup design, a motor-driven variable orifice device is incorporated in the control rod drive assembly to adjust the coolant flow to the associated fuel region. The adjustable orifice throttles the coolant flow to match the region's power. The temperature of the helium leaving each fuel region provides the information for adjusting the variable orifice. The reference design does not require variable orifices because of the flatter power distribution obtained from on-line refueling. Fixed orifices are provided in the reflector blocks of the outer rows of fuel columns to match flow to the power generation.

The control rods, as designed, provide the ability for each section of the control rod to articulate and thus allow the rod to pass through the clearance hole, which is likely to include some minute offsets at the fuel element interfaces. The maximum variation of location of a control rod hole in a mating fuel element is approximately 0.040 in. This is determined by the tolerances specified for the fuel elements. Although this represents the maximum possible deviation, in all probability this will occur in extremely rare instances. Significant accumulation of tolerances in a control rod column is limited by the 0.040-in. clearance of the surrounding fuel columns. Thus the maximum accumulation of offset in a single column should be limited to approximately 0.120 in. based on the assumption that the column can only deviate by 0.040 in. from a vertical stack. The radial clearance of 0.240 in. between the control rods and the hole appears sufficient to allow control rod insertion. Thus the deviation of a fuel column could be as great as 0.160 in. before control rod insertion would become a problem. The fuel element tolerances are further discussed in Section 5.2 above.

The control rod drives are designed so that in the scram mode the control rods are inserted by gravitational force. Response times were not checked because of the lack of sufficient detailed information. The dynamic retarding method of slowing the rod during the scram appears to be a nonstandard technique in motor control practice; however, it does have the advantage of reducing the complexity of the drive mechanism. This retarding system may have the problem that insufficient residual magnetism in the motor rotor will eliminate this retarding torque and allow the rod to run in at much higher speed and damage the rod or the drive mechanism. Confirmation of the reliability of the control rod drive system will require performance testing. Safety aspects are discussed in Chapter 6.

The cable and drum drive system presents two maintenance problems. It will be somewhat difficult to retrieve the rod after a break in the cable. If the drive system should become inoperable with the control rod fully inserted into the core and if the rod cannot be cranked up by hand, the removal of the longer assembly consisting of the drive and the extended rod may require special handling equipment. Such an operation might be costly and present containment problems. Although the control rods and drives have some problems, the development programs associated with the Fort St. Vrain station should provide the required solutions, since the control rod design is essentially identical to that for the Fort St. Vrain plant.

5.4.2 Reserve Shutdown System

Both designs provide a reserve shutdown system that can be used to insert a poison material into the core if the control rods become inoperable. This system is manually controlled and functions independently of the normal control rod system. It is housed within the control rod drive assembly and contains granular poison material within hoppers connected to individual channels within each fuel region. Application of pressure to a rupture disk in the hopper permits the poison granules to fall freely into a channel provided in the central fuel column in each of the 91 fuel regions. The evaluation of this system is discussed in Chapter 6.

5.4.3 Plant Control System

With the exception of the slow xenon oscillations and the possible instabilities in the once-through boilers, the HTGR-type reactors are inherently stable, slow moving, and relatively easy to control. The Fort St. Vrain plant appears to have a sophisticated plant control system that gives good performance in the simulator studies in the PSAR. The system is the load-following type in which the reactor power is made to follow large rapid changes in load on the turbine and provide steam at a constant temperature and pressure. The control system also has the advantage that it is able to retain control of the plant during disturbances, such as the isolation of the turbine from the external electrical grid or the isolation of a reactor coolant loop. The plant control system employs a number of tight control loops similar to those used in the plant control system suggested by ORNL previously.[18] The setpoints for these tight control loops are continuously computed from the measured flows to give rapid response to changes in load. The setpoints are also slowly adjusted to bring temperatures in the steam system to their desired values.

5.4.4 Spatial Control

The 31-ft diameters of the cores of the 1000-Mw(e) designs are large enough that they will probably be subject to xenon oscillations in the radial direction. The pancake design of the cores greatly simplifies the spatial control problem by eliminating axial oscillations. The radial oscillations can be eliminated by controlling the power in individual zones of the core. The outlet temperatures of the core coolant channels might be used for this zonal control. Some development of the required temperature sensors may be needed. A large number of controllers may be desirable to manipulate the control rods in the individual zones. The larger British reactors have used zonal control[19] where the control rods in a zone are manipulated by conventional controllers to hold the average outlet temperature in that zone constant.

The design has only one rod pair out of 91 for automatic control. The control system could be arranged to allow the movement of only one rod pair a time to meet the one-rod-runaway criterion used in the Fort St. Vrain design. It will probably be possible to increase the number of rods that can be moved at one time, since PSAR Amendment 3 indicates that the present automatic scram system can adequately cope with an all-rod-runaway accident. This larger rate of reactivity of insertion would, however, reduce the time available for manual actuation of the reserve shutdown system if the automatic shutdown system failed.

In order to get a faster response to rod movement, it may be necessary to use in-core flux detectors instead of temperature measurements for spatial control, as suggested in the report[20] on the 10,000-Mw(th) HTGR design. The in-core chambers would have to operate under high temperature and radiation conditions and should be replaceable. Calculations of the spatial dynamic behavior will be needed in determining the requirements for the control and safety systems.

5.4.5 Reactor Startup Control

The Fort St. Vrain design has two startup detectors located right above the neutron sources on the top of the core. The detectors or the sources will have to be relocated so that the detectors see source neutrons that have been multiplied by the core. A larger number of startup detectors may be required in the 1000-Mw(e) designs to monitor local criticality in the large, loosely coupled cores.

5.5 Plant Auxiliary and Service Systems

5.5.1 Systems that Service Both the Reactor and Turbine Plants

Auxiliary and service systems for the plant include service water, domestic water, fire protection, instrument and service air, communications, and the electrical systems. These systems serve both the reactor and turbine plants and are described as follows:

5.5.1.1 Service-Water System. A single service-water system is provided for the station. The service-water system is supplied by water pumped from the cribhouse by three 50% station-capacity motor-driven service-water pumps (one standby). These pumps discharge into a loop equipped with connections for the following services:

1. turbine lube oil coolers,
2. hydrogen coolers,
3. hose connections for service water,

4. reactor plant cooling-water system,
5. miscellaneous small cooling systems.

The service-water design inlet temperature will be 75°F.

5.5.1.2 Domestic-Water System. Water for the station domestic-water system makeup is delivered through a hypochlorinator to a water-storage tank that serves as the source of supply within the turbine plant. Domestic water is also used for condensate makeup after being treated in the makeup demineralizer.

5.5.1.3 Fire-Protection System. A station fire-warning system is provided and consists of spot detectors and an annunciator system in the central control room to initiate an alarm in case of fire in critical areas. Fire protection is provided by fixed fire-protection systems supplemented by portable extinguishers.

Water for fire fighting is provided by the motor-driven pumps, with backup provided by an emergency diesel-driven pump arranged to deliver water from the cribhouse. A fire-water storage tank of 20,000 gal capacity will provide an instantaneous supply of water to the protected areas when required.

5.5.1.4 Instrument and Service Air. Instrument air is supplied at approximately 100 psig by three full-capacity (one standby) unlubricated motor-driven air compressors. Each compressor is served by one receiver and one dryer with a pre-filter and an after-filter.

Service air is supplied at 100 psig by one oil-lubricated motor-driven air compressor. The compressor provides backup for the instrument air compressor.

5.5.1.5 Communication System. The communications provisions include a combination public address and intercommunication system serving both turbine plants and the nuclear steam supply systems and consist of telephone-type handset stations and transistorized amplifier loudspeaker assemblies.

5.5.1.6 Electrical. The selection of equipment and material conforms to modern central-station practice. Sectionalizing of buses, dual power feeds, and automatic transfers are provided consistent with the overall reliability philosophy of the nuclear plant. Bus sections with multiple supplies are provided to serve those loads that require a high order of power continuity in order to insure either uninterrupted plant operation or safe and orderly shutdown and protection of equipment and personnel.

1. Operation. The electrical system is monitored and controlled from the electrical control boards in the central control room. Motors are controlled either from the central control room or from centralized control panels at appropriate locations in the plant.

2. Generator Protective Relays. Generator protective relays include high-speed differential, loss of field, field ground, negative-phase sequence, reverse power, backup, and generator ground.

3. Turbine-Generator Leads. An isolated phase bus is provided from the generator terminals to the low-voltage terminals of the main transformer. Current transformers are furnished on the generator terminals. Potential transformers and generator surge-protection equipment are connected to isolated phase-generator bus taps.

4. Auxiliary Transformer. The auxiliary transformer with the primary winding properly rated for the generated voltage and the secondary winding rated at 4.16 kv is a 3-phase 60-cycle unit and is closely coupled to the generator bus duct. The secondary of the transformer is sized to carry the total auxiliary requirements for full-load operation.

5. Reserve Auxiliary Transformer. A reserve auxiliary transformer of approximately 80% capacity rated at 230 to 4.16 kv is a 3-phase, 60-cycle unit, and the transformer secondary is connected to the 4160-v startup switchgear bus. This transformer is used primarily to start the turbine-generator unit, but it may also function as a backup for the auxiliary transformer and normally supplies plant shutdown auxiliary power requirements.

6. Auxiliary Generators. Two 4688-kw 0.8-pf 4.16-kv 3-phase 60-cycle auxiliary generators are provided. Each generator is driven by a condensing low-pressure steam turbine and is on the line under normal operation to supply power to the boiler feed-pump motor and other essential auxiliaries. Each auxiliary generator has adequate capacity to supply all essential shutdown cooling electrical loads in the event of a loss of outside power.

7. 4160-v Switchgear. The 4160-v switchgear is of dead-front metal-clad construction and is located indoors. There are four separate 4160-v switchgear buses. Two of these buses (designed as unit buses) take their feed during normal operation from the auxiliary tranformer and during startup from the reserve auxiliary transformer. The other two buses (designated as essential buses) are fed via bus-tie breakers from the unit buses. In addition, the auxiliary generator connects to one of the essential buses. Load-center transformers rated at 4160 to 480 v are 3-phase 60-cycle units utilized to supply 480-v loads.

8. 480-v Switchgear. The 480-v switchgear is of dead-front metal-clad construction. Separate 480-v switchgear buses are designated as normal or essential buses, and the auxiliary electrical loads are assigned to these buses as appropriate. Vital 480-v loads are divided between the essential buses.

9. Motor-Control Centers. The motor-control centers are of dead-front metal-clad construction and contain combination line starters and molded-case air circuit breakers for 480-v 3-phase motor control. However, the control rod motor control center is of special design and rated for 120-v 3-phase power. As a general rule, motors rated at 40 hp or below are connected to the motor-control centers. In a few cases, starters are located at or near the motor.

10. Instrument Power. There are three instrument-power buses. Two of these are driven by dc to ac inverters, with continuously charged batteries floating on the line and thus providing critical power for plant instrumentation. The third bus is supplied by a transformer from the 480-v essential switchgear.

11. DC System. Two separate 125-v dc buses are provided, with each connected to its own nominal 125-v dc battery. One of the 125-v dc buses is designated the reactor plant bus and one the turbine plant bus. Separate battery rooms are provided. These rooms are fully enclosed and ventilated and block-wall construction is used.

12. Motors. Motors rated 250 hp and above are supplied from the 4160-v switchgear. Motors rated below 250 hp are supplied from the 480-v system, except that motors smaller than 1/2 hp are supplied 120-v single-phase power.

Motors located outdoors have weather-protected NEMA Type II or totally enclosed fan-cooled enclosures. Weather-protected motors are equipped with space heaters. Drip-proof motor enclosures are standard for normally clean and dry locations. Splash-proof or totally enclosed motors are provided where necessary.

13. Lighting System. The plant lighting system is designed to conform to the latest practice for central-station properties. The plant lighting system distribution voltage may be 120 or 277 v. Incandescent, fluorescent, and mercury-vapor lamps are used in suitable fixtures to achieve the intensities and light distribution required for a general lighting system throughout the plant areas. All lighting levels are maintained-in-service values.

An emergency lighting system is designed to provide adequate illumination in the event of loss of normal lighting power. About 80% of the installed lighting is connected to the essential power supplies and consists of incandescent lamps.

14. Raceways and Wiring. Cables for 4160- and 480-v power service are three-conductor butyl rubber or polyethylene insulated and run on ladder-type cable trays clipped to supports or in conduit. The 120-v ac control circuits, 125-v dc control circuits, and low-voltage (below 125-v) dc instrumentation circuits are multiconductor polyethylene insulated with PVC jackets and run on expanded metal trays or in conduit. Low-voltage (below 120-v) ac transmitter circuits are twisted and shielded cables run on the same expanded metal trays as the low-voltage dc instrumentation circuits or in conduit. Thermocouple circuits are either shielded twisted pair or multiconductor thermocouple cable also run on the same expanded metal trays as the low-voltage instrumentation circuits or in conduit. Nuclear instrumentation and radiation monitoring signal circuits are coaxial, triaxial, or multiconductor cable that may be run in the instrument trays.

Power, control, and instrumentation cable runs penetrating the PCRV are designed to meet the specific requirements of each individual case. These cable runs utilize mineral-insulated silicone rubber or other special insulation systems as appropriate. Shielding and isolation for control and instrumentation wiring are provided similar to that in the rest of the plant.

15. Grounding. A low-resistance copper grounding system is installed throughout the plant, both indoors and outdoors, in accordance with the National Electrical Safety Code.

5.5.2 Reactor Auxiliary and Service Systems

The reactor plant auxiliary and service systems include the helium purification, helium storage, nitrogen supply, reactor plant cooling water, decontamination, radioactive-liquid waste, and the radioactive-gas waste systems. These systems are described below.

5.5.2.1 Helium Purification System. The helium purification system provides for the removal of gaseous activity and chemical impurities from the primary coolant system by purifying a side stream.

Fission-produced isotopes, other than the noble gases (krypton and xenon) and tritium, are removed from the side stream in a high-temperature filter-adsorber unit. A dryer removes water and carbon dioxide and maintains an impurity level of 10 ppm ($CO + CO_2 + H_2O$) with continuous H_2O inleakage of up to 0.04 lb/hr. Krypton and xenon isotopes and chemical impurities, such as carbon monoxide, hydrogen, and nitrogen, are removed by a low-temperature (cryogenic) absorber. Normally, all impurities except hydrogen and tritium are completely removed from the side stream. The hydrogen content is reduced to only a fraction of its design inlet concentration (10 ppm) because of a practical limitation on the size of the low-temperature absorber. The system limits the carbon monoxide concentration in the primary coolant system to less than 10 ppm during normal operation. Except for trace amounts of tritium, the helium purification system normally does not return any activity to the primary coolant system.

The purification system normally processes a side stream helium flow of approximately 935 lb/hr at full load. The purified helium is normally returned to the primary coolant system as purge gas for such components as circulator seals, control rod drive nozzles, penetrations to the PCRV, etc. The major components that cool and purify radioactive helium are located within the top head of the PCRV. The purified helium filters and compressors, instrumentation, and a subsystem for equipment regeneration are external to the PCRV.

There are two parallel purification trains for the reactor, but only one is normally on stream at a time. Major items in each train are a high-temperature filter-absorber (charcoal), a dryer (molecular sieve), a low-temperature adsorber (charcoal), and several heat exchangers. The helium side stream enters the high-temperature filter-adsorber, which removes dust, as well as the isotopes noted above. It is then cooled to about 100°F and flows through the dryer. The dryer effluent is cooled in a regenerative heat exchanger to about −295°F and then passes through the low-temperature adsorber. The purified helium leaving the low-temperature adsorber, which is cooled by liquid nitrogen circulating through tubes wrapped around the adsorber, is rewarmed in the regenerative heat exchanger. Finally, the purified helium is brought out of the PCRV, compressed, and returned to the primary coolant system. Two compressors are provided (one standby). Either compressor can handle the gas from the train on stream.

The low-temperature adsorber has capacity for at least one year's production of ^{85}Kr in the absence of carbon monoxide. However, the adsorbers are taken off-stream for regeneration after six months of operation. Prior to the regeneration, the adsorber is maintained at approximately normal operating temperature for about two months to permit decay of essentially all radioisotopes, with the exception of ^{85}Kr, tritium, and a relatively small amount of ^{133}Xe.

Each dryer is sized for the removal of the water and carbon dioxide formed as a result of a single steam-generator offset tube failure, with no more than a single regeneration required.

The equipment comprising each purification train, which is located within the PCRV, is designed to fit into holes in the top head of the PCRV. Each train requires five holes. The holes in which the high-temperature filter-adsorbers are located are 19 in. in diameter and are open to the primary coolant system at the bottom. The other holes are 27 in. in diameter; these holes are closed at the bottom and do not communicate with the primary coolant system.

Regeneration of the dryers and low-temperature adsorbers is accomplished by circulating a stream of hot helium through the beds at atmospheric pressure to raise the adsorbent temperature to a level where efficient desorption is possible. The regeneration system consists of a blower, a heater, a cooler, a knock-out drum, a dryer, and a filter. Gases produced by the regeneration operation are vented to the radioactive-gas waste system for disposal.

5.5.2.2 Helium Storage System. The helium storage system serves two purposes: first, it provides storage capacity for the reactor helium inventory, and, second, it provides a supply of high-pressure helium for various purging operations. When it is desired to store the reactor helium inventory in the helium storage tanks, the following procedure is used. The primary coolant system is first equalized in pressure with the storage tanks at a controlled rate. Flow of helium is directed through the purification system for removal of radioactive and chemical contaminants. When pressures are equalized, the plant pressure is approximately 390 psia. The helium transfer compressor is then started to take suction from the purified helium line and to discharge to the storage tanks. When plant pressure is reduced to atmospheric, helium pressure in the storage tanks is approximately 1100 psig. The primary coolant system is repressurized by essentially reversing the depressurization operation.

The major nominal design criteria are as follows:

Helium transfer compressor capacity, acfm	300
Plant equalization and pumpdown time to atmospheric pressure, hr	12
Plant equalization and pumpup time to operating pressure, hr	6
Helium storage pressure, psig	1000–1250
Helium inventory, lb	
Backup design	23,500
Reference design	15,600
High-pressure helium supply pressure, psig	100–1250

5.5.2.3 Nitrogen System. The reactor has a nitrogen system designed to furnish liquid nitrogen to the helium purification system and to recondense gaseous nitrogen from this system. This system is located outside the PCRV. An atmospheric-pressure liquid-nitrogen storage tank is provided to supply approximately a one-day supply of liquid nitrogen per reactor for emergency use.

5.5.2.4 Reactor Plant Cooling Water System. Cooling water supplied to some of the reactor plant heat exchangers (primarily the PCRV liner cooling coils) is exposed to radiation fields, which will produce substantial radiolytic dissociation of the water. In order to control this water dissociation and the resulting corrosion problem, a closed circulating water loop is provided to serve these exchangers. This water is also used to cool equipment where water-side scaling could create a maintenance problem. Other equipment is cooled directly with service water obtained from, and returned to, the plant circulating-water system.

The reactor plant cooling water system consists of two cooling water return tanks (one high pressure for PCRV cooling and one low pressure for service water), three 50% capacity high-pressure cooling water pumps, two 50% low-pressure cooling water pumps, two 50% capacity heat exchangers, and a bypass filter and demineralizer, together with associated piping and controls. The cooling water return tank is sized to provide approximately 5 min surge capacity, with 20% freeboard. The tank is maintained at about 100 psia by means of an inert gas blanket.

The demineralizer accomplishes two purposes: (1) removal of impurities in the water, and (2) maintenance of the water pH at approximately 10.5 by release of lithium hydroxide from the demineralizer resin to minimize corrosion problems. Additional chemicals may be injected into the circulating closed water loop to provide further control of water chemistry.

Water in the return tank is circulated through the loop by two of the three high-pressure cooling water pumps. The standby pump starts automatically on reduction of loop flow. Loop water leaving the process coolers is cooled, in turn, in the two heat exchangers, which reject heat to the service water obtained from, and returned to, the circulating-water system. After being cooled, the closed loop water is collected in the cooling water return tank.

The initial charge of water for the closed loop, as well as any required makeup, is obtained from the main condensate pump discharge. This insures that only high-purity demineralized water is supplied to the closed loop.

5.5.2.5 Decontamination System. The decontamination system provides facilities for removal of radioactive contamination from the surfaces of various reactor plant equipment items so that they can be safely maintained. Decontamination operations are carried out in the hot service facility.

The specific procedure to be followed depends on the items to be decontaminated and the nature and amount of radioactivity involved. Therefore, facilities are provided that permit the utmost flexibility in the selection and sequence of the processing steps.

The decontamination system consists of a solution storage tank, a pump to transfer the decontamination solution to the hot service facility, a recirculation pump, and a solution filter, together with air, steam, and hot water supply headers and drain headers. Detergents are supplied by aspirating concentrated solutions from portable containers with steam as the aspirating medium. The hot service facility is equipped with suitable vacuum cleaning, spraying, etc., equipment for the decontamination operations.

The system is designed for collecting and reusing the decontamination solution. When finally spent, the solution is pumped to the radioactive-liquid waste system for disposal. Detergent and rinse streams collected in the hot service facility are also transferred to the liquid waste system for disposal.

5.5.2.6 Radioactive-Liquid Waste System. The radioactive-liquid waste system collects and monitors all aqueous wastes generated in the reactor plant. Liquids with activity below the maximum permissible concentration (MPC) are disposed of at the plant site by pumping into the circulating water discharge canal. Liquids with activities exceeding MPC are processed at the plant site to minimize the cost of disposal.

Liquid wastes generated in normal operation of the reactor plant are expected to be essentially free of activity and to occur infrequently and in limited quantities. Radioactive liquid wastes in appreciable quantities will be produced only by decontamination operations or as the result of accidents.

Liquids of low activity are collected in the reactor building sump. From the sump, the liquids are pumped through one of two filters (one standby) to one of the two liquid waste receivers. A standby sump pump is provided. When a convenient amount of liquid has been collected in a receiver, the incoming fluid is diverted to the second tank. The first tank is then isolated, and a sample is analyzed to determine the activity. If the liquid is below MPC it is pumped by the liquid waste transfer pump to the circulating-water discharge canal for disposal. If the activity exceeds MPC limitations, the liquid is retained for further processing.

Liquid wastes originating as the result of an accident (steam-generator tube failure) and/or decontamination operations, and which are known to be too radioactive for direct disposal, are collected directly into one of the liquid waste receivers. The liquid is pumped through one of two demineralizers and collected in a liquid waste monitor tank. Analysis of the demineralized liquid determines whether the activity has been reduced to a level permitting disposal or whether the liquid must be recycled through the demineralizer.

5.5.2.7 Radioactive-Gas Waste System. The radioactive-gas waste system is designed to handle all radioactive or potentially radioactive gases that must be vented from the reactor plant, except for the effluent from the plant ventilation system. Gas streams to be processed by this system include equipment vents, gases produced by regeneration of absorption beds in the helium purification system, and vent gases resulting from purging various items of process equipment.

The radioactive-gas waste system consists essentially of an inlet vacuum tank, two surge tanks, two compressors (one standby), two vent gas filters (one standby), two blowers (one standby), and associated piping and controls. In operation, all potentially radioactive gases entering the gas waste system are collected in an inlet header and directed through one of the vent gas filters. Filter effluent is continuously monitored for activity. If the activity of the effluent gas is less than a preset value, the gas is transferred via one of the gas waste blowers to the ventilation system exhaust filters for ultimate disposal to the plant stack. Should the gas activity exceed this preset value, the gas is automatically diverted to the inlet gas waste vacuum tank.

Inlet gases to the system that are known to be radioactive, such as the helium purification system regeneration gas, are routed directly to the vacuum tank.

When the pressure in the gas waste vacuum tank rises to about 11 psia as a result of gas input to the tank, one of the gas waste compressors is automatically started. The compressor transfers gas from the vacuum tank to one of the gas waste surge tanks.

Radioactive gases collected in the surge tanks are ultimately disposed of by venting in a controlled manner to the ventilation system exhaust filters and thence to the plant stack after analysis of the tank contents. This is accomplished by opening a vent line from each tank to the inlet of the gas waste blowers. A flow controller limits the rate of release of the tank inventory to the value established by the analysis.

System design parameters are

Gas waste compressor capacity, each, acfm	50
Gas waste blower capacity, each, acfm	80
Vacuum tank volume (operating pressure 11 psia), ft^3	500
Surge tank volume, each (operating pressure 50 psia), ft^3	700

5.5.3 Turbine Plant Auxiliaries

Turbine plant auxiliaries include the lubricating-oil purification system and the feedwater treatment system.

5.5.3.1 Turbine Lube-Oil Purification System. The turbine lube-oil purification system consists of a turbine lube-oil storage tank with clean and dirty oil compartments, a transfer pump, and a purifier (centrifuge). Additionally, a filter pump is provided to take suction from the turbine lube-oil reservoir and continuously bypass oil through a cartridge-type filter back to the reservoir. Both the purifier and filter are arranged to process lube oil from the turbine lube-oil reservoir or the storage tank.

5.5.3.2 Water Treatment. A full-flow condensate demineralizing system is provided that consists of five 25% demineralizer vessels, one of which is a spare. A regeneration system, a chemical injection system, and a makeup water treatment system are provided.

Accessory equipment for the condensate demineralizer systems includes: demineralizer outlet strainers, backwash resin trap, external regeneration facilities, and air-opened valves and controls for remote semiautomatic operation from the control cubicle. The chemical injection system for oxygen scavenging and pH control in the secondary water treatment system includes chemical pumps and solution tanks. The makeup water treatment system includes two cation units, two anion units, one degasifier, and rubber- or plastic-lined tanks.

5.5.4 Evaluation

The auxiliary and service systems were not evaluated in detail but were reviewed to assure that the necessary items were included. In all cases the systems appear to be adequate.

As a result of the evaluation the auxiliary turbine-generator was changed from a single unit to two 50% capacity units to provide the reliability required for emergency power provision. The auxiliary boiler that supplies steam to the auxiliary turbine-generator during emergency operation is provided with redundant auxiliary systems and is fired at a low level to allow rapid startup during periods when outside power is unavailable. Upon shutdown, sufficient steam is available from the turbine bypass flash tank to provide emergency power for approximately 30 min and thus allow time for startup of the auxiliary boiler.

Comparison of the helium purification system with that for the Fort St. Vrain plant indicates that the units are identical. At the outset this appeared unreasonable, since the coolant inventory and the flow rate are approximately three times as great. GGA states, however, that the primary function of the helium purification system is to provide purified helium flow for purging the seals of the circulators, control rod drives, instruments, and PCRV penetrations and to purify the reactor coolant during the depressurization of the primary coolant system. The purification system maintains chemical impurities at a level low enough to prevent graphite erosion. As a byproduct the system also reduces the coolant radioactivity. Based on the principal purified helium purge requirements (50 lb/hr per circulator and 5 lb/hr per control rod drive), the system requirements are approximately 800 lb/hr, which allows a reserve of approximately 17%. This appears to be adequate. Nevertheless, detailed accident analyses should be made when design is finalized and when leak and reaction rates are established. A third purification train may be desirable if the failure of an operating train would result in long reactor downtime.

Both the helium purification system and the reactor plant cooling water system are essential parts of the reactor containment system. The purification system is the only point where the primary coolant leaves the PCRV, while the reactor plant cooling system provides the PCRV integrity by maintaining the operating temperature at a permissible level. Thus the designs of these systems are critical, and careful attention must be given the operating requirements. This is discussed further in Chapter 6.

5.6 Turbine-Generator Systems

The turbine plant designs for the backup and reference plants are summarized in Table 5.7. The throttle steam for the backup design is at 2415 psia and 1000°F. The reference design specifies supercritical steam with throttle conditions of 3515 psia and 1050°F. Both designs have conventional tandem-compound turbine-generators with reheat. The turbine is of standard reheat design, except that steam from the high-pressure turbine exhaust (cold reheat) is expanded through the main circulator turbine drives before entering the reheaters. Thus the circulator turbine drives replace one or two stages in the exhaust end of the high-pressure turbine.

Startup, shutdown, and emergency conditions require the use of a special steam bypass system. The bypass system accepts steam-water fluid from the superheated steam header before and during startup of the main turbine and during shutdown. The system, which is designed to accept full-load steam flow following a turbine trip, consists of a flash tank, a back-pressure valve, a desuperheater, and the associated piping. Steam from the flash tank, supplemented by auxiliary boiler steam, is available

Table 5.7. Turbine Plant Characteristics for Backup and Reference Designs

	Backup Design	Reference Design
Turbine-generator		
Throttle conditions		
Pressure, psia	2415	3515
Temperature, °F	1000	1050
Enthalpy, Btu/lb	1461.2	1461.0
Flow, lb/hr	6.7465×10^6	6.7257×10^6
Reheat conditions		
Pressure, psia	567	568.6
Temperature, °F	1000	1050
Enthalpy, Btu/lb	1517.7	1544.4
Flow, lb/hr	6.6744×10^6	5.9755×10^6
Condenser conditions		
Pressure, in. Hg abs	1.5	1.5
Flow, lb/hr	4.5926×10^6	4.0081×10^6
Enthalpy, Btu/lb	1023.7	1036.2
Duty, Btu/hr	4.966×10^9	4.538×10^9
Turbine arrangement	TC 6F-33.5	TC 6F-30
Net station output, Mw	1001	1000.2
Net station heat rate, Btu/kwhr	8379	7911
Net station efficiency, %	40.73	43.14
Reactor thermal output, Mw	2457	2318
Station electrical auxiliary power, Mw(e)	17	18
Generator rating		
Total, kva	1,122,000	1,122,000
Terminal voltage, kv	24	24
Power factor	0.9	0.9
Feedwater conditions		
Final temperature, °F	410	510
Enthalpy, Btu/lb	389.3	499.6
Shaft-driven auxiliaries	None	None
Exhaust-steam-powered auxiliaries	Six helium circulators One 87.5%-capacity boiler feed pump	Six helium circulators One 87.5%-capacity boiler feed pump
Condensing system		
Circulating-water pumps	Two 45%, one 10%	Two 45%, one 10%
Flow rate, each, gpm[a]	Two at 215,000, one at 47,000	Two at 197,000, one at 44,000
Drive rating, hp[a]	Two at 1600, one at 400	Two at 1500, one at 275
Condenser type	Single-pass multi-pressure	Single-pass multi-pressure
Units	1	1
Tubes		
Material	Admiralty metal	Admiralty metal
Size, in.[a]	1	1
Length, ft	90	90
Design heat transfer surface, ft^2	400,000	370,000
Design cooling-water temperature, °F	57	57
Design saturation temperature, °F	91.7	91.7
Cooling-water flow, total, gpm[a]	477,000	438,000
Feedwater system		
Feedwater demineralizing	Full flow	Full flow
Deaeration	Yes	Yes
Feedwater heaters		
Total number	10	12
Number of banks	2	2
Drains	Cascade	Cascade
Boiler feed pumps		
Number	2	2
Capacity	One at 87.5%, one at 12.5%	One at 87.5%, one at 12.5%
Flow, lb/hr	6.75×10^6	6.73×10^6
Feedwater temperature, °F	410.0	510.5
Driver	Steam-turbine, electric motor	Steam-turbine, electric motor
Emergency boiler feed pumps		
Number	1	1
Capacity	12.5% pump	12.5% pump
Drive	Listed above	Listed above

[a]Assumed values.

for operating the helium circulator turbines, the boiler feed pump turbine, deaerating heater, main turbine seals, and the main condenser air ejector.

In addition to the main turbine-generator, each design provides two auxiliary turbine-generators operating on exhaust steam from the intermediate-pressure turbine during normal operation. During plant shutdown the auxiliary turbines utilize steam from an oil-fired package boiler to provide the plant emergency power requirements. Each auxiliary turbine-generator is rated at 4683 kw and is capable of supplying power for essential operations during a power failure. The package boiler provides 250-psig 600°F steam at 135,000 lb/hr.

Use of the primary heat transfer system for shutdown cooling requires a more reliable feedwater system than is normally provided in a conventional plant. To provide this reliability an emergency feedwater line from the boiler feed pumps to the steam generators is provided. In addition, an emergency condensate line from the condensate pumps to the steam generator is provided as a backup. Both designs provide a conventional feedwater system, with the backup design having six stages of feedwater heating in two trains to provide a final feedwater temperature of 410°F. The reference design employs seven stages of feedwater heating in two trains to provide a final feedwater temperature of 510°F. Each design provides one 87.5% turbine-driven boiler feed pump and one 12.5% electric motor-driven boiler feed pump. During normal operation, steam for the turbine-driven pumps is obtained from the exhaust of the intermediate-pressure turbine. At loads below approximately 50%, steam is obtained from the cold reheat line. The condensate pumps are two 50%-capacity electric-motor-driven pumps with two 12.5%-capacity motor-driven pumps provided as a backup for use during the loss of one 50%-capacity pump.

The condenser is a single-pass multipressure unit that provides an equivalent back pressure of 1.5 in. Hg abs. Admiralty-metal tubes approximately 90 ft long are used in the condenser. The inlet cooling water temperature is 57°F, with the condenser being designed for a cooling water temperature rise of 22°F. The circulating-water system, which is of conventional design, provides fresh river water at a pumping head of 25 ft.

A common crib house serves the circulating-water system, as well as service-water systems required for the reactor plant.

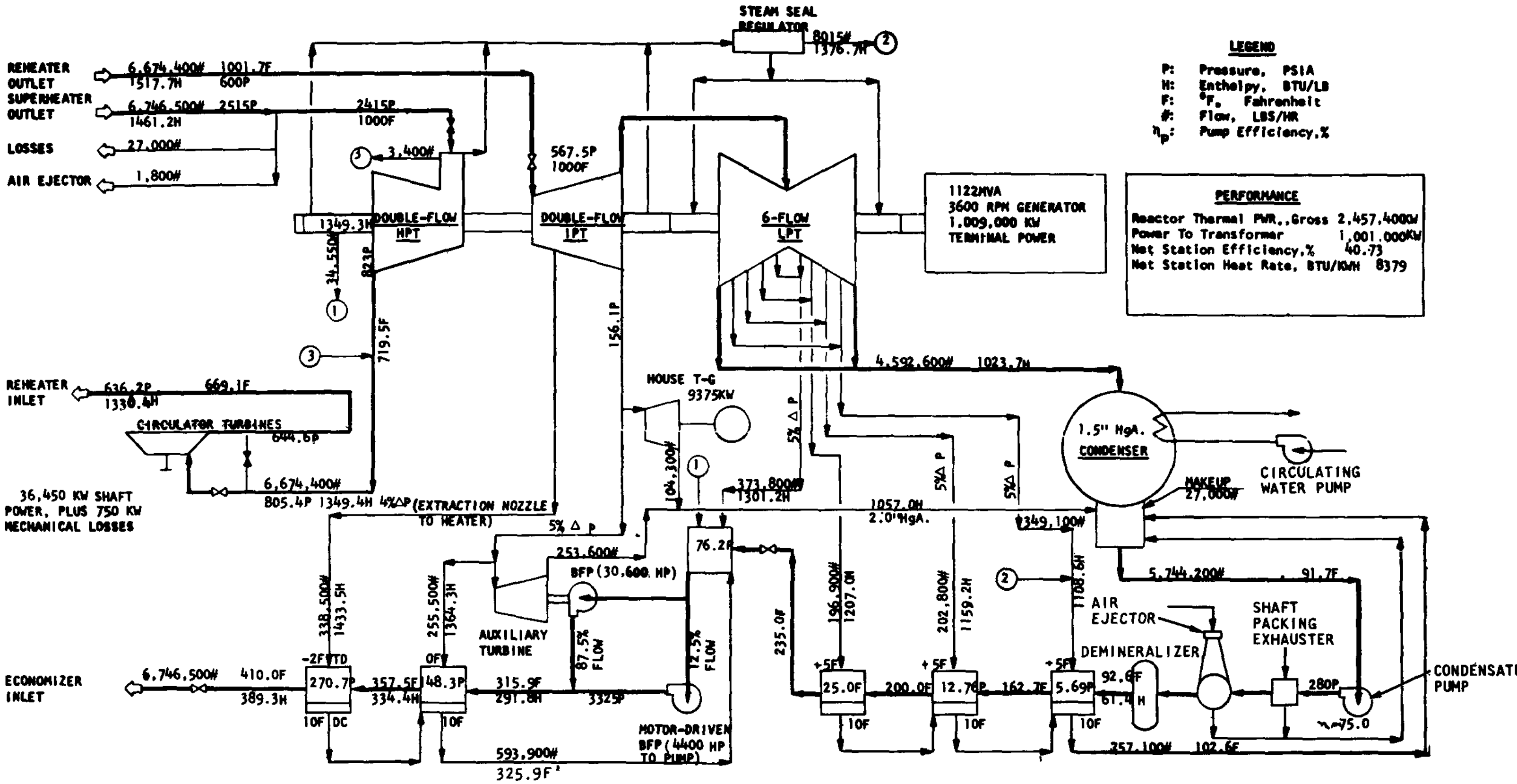
The heat balances shown in Figs. 5.8 and 5.9 were evaluated and found to be correct to within a fraction of 1%. The net station heat rates of 8379 and 7911 Btu/kwhr for the backup and reference designs, respectively, were confirmed.

The six-flow tandem-compound turbine-generators specified for both designs represent an extrapolation of approximately 65% over the largest tandem-compound unit currently in use. The specified exhaust flows are reasonable in all cases. General Electric has stated that they are prepared to build a tandem unit of this size but as yet have not sold one. The largest available rating on a 3600-rpm generator is currently 1280 Mva, which is more than adequate for both designs.

Both the backup and reference designs take advantage of the 5% overpressure operation guaranteed by the turbine-generator manufacturers. This, in effect, results in specifying a rating of approximately 5% less than the plant's design rating and utilizing the 5% guaranteed overpressure to achieve the plant rating. An approach of this type results in a reduction of approximately 3.5% in turbine-generator cost; ORNL allowed for this reduction in the capital cost evaluation. However, in practice, the utility may wish to have the additional margin in turbine-generator performance to take advantage of a possible improvement in the reactor performance that may be possible as operating experience develops.

The use of the 30-in. last-stage blade (LSB) unit in the reference design rather than a 33.5-in. LSB unit reduces the turbine cost by approximately 6% (0.02 mill/kwhr) but increases the heat rate by approximately 1% (0.03 mill/kwhr). These are only approximate values, and the overall economics must be analyzed to assess the value of the selection of a 30-in. LSB unit rather than a 33.5-in. LSB unit.

The utilization of the primary heat transfer system for decay heat removal requires both added redundancy and specialized equipment in both the feedwater and the steam systems. The reliability of these features is essential in providing both power and cooling water immediately after a reactor scram. This requires that careful attention be given to the sources of systematic failures in this equipment. Investigation of the



1000 MW HTGR PLANT

2400 PSIG, 1000/1000F, 1.5" Hg A.
TC6F - 33.5" TURBINE

Fig. 5.8. Heat Balance Diagram for Backup Design. (GGA illustration)

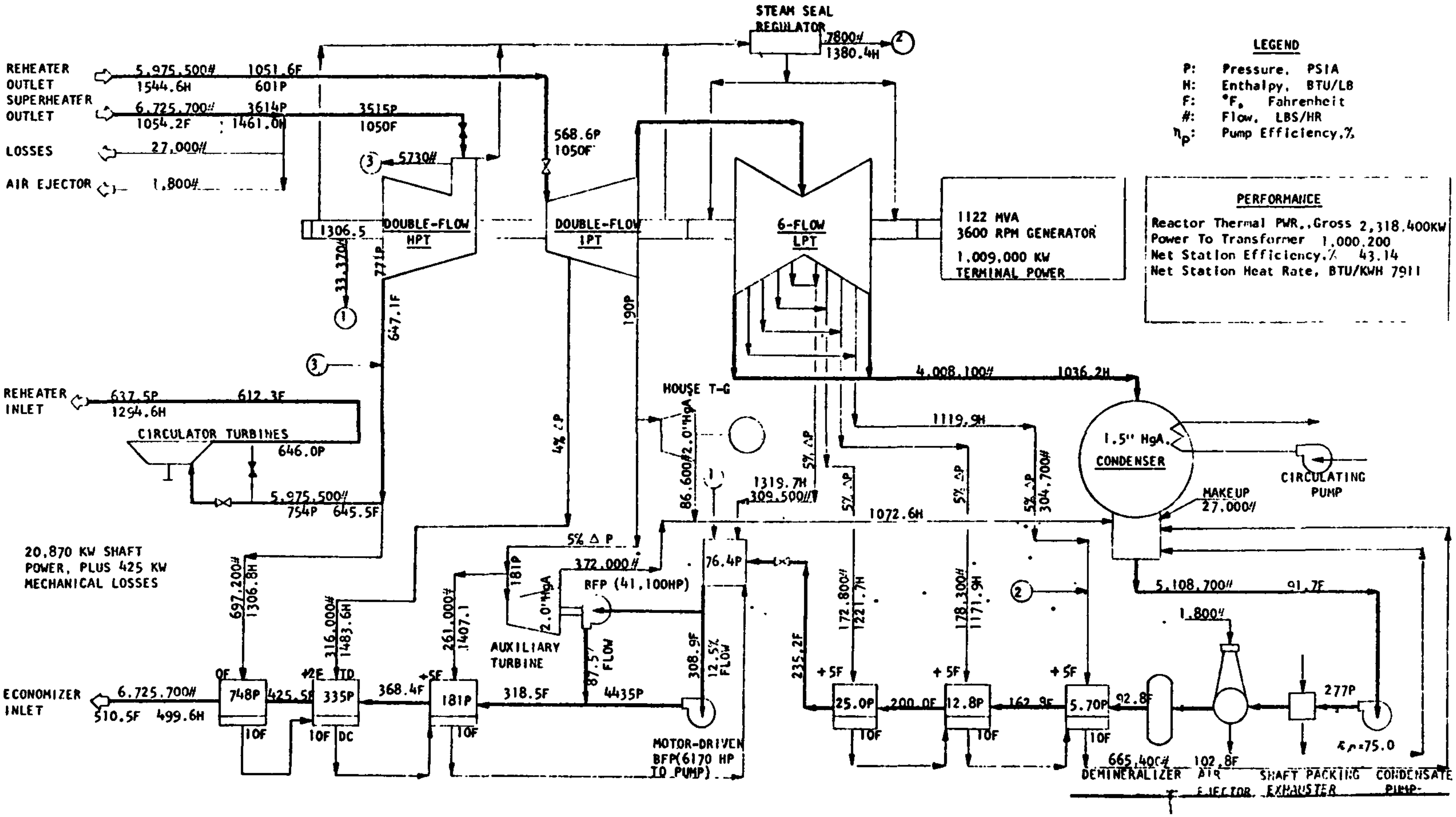


Fig. 5.9. Heat Balance Diagram for Reference Design (1000-Mw HTGR Plant, 3500 psig, 1050/1050°F, 1.5 in. Hg abs, TC6F 30-in. Turbine). (GGA illustration)

various modes of operation, the dynamic characteristics, and the instabilities of these systems, including the transients involved when the mode of operation is changed, is essential in proving the adequacy of the design. Such analyses were not made in this evaluation study.

5.7 Site and Buildings

5.7.1 Plant Site

The following site conditions were assumed for both the backup and reference design for the 1000-Mw(e) HTGR station.

5.7.1.1 General Characteristics. The site is of adequate size and located on a navigable river in an area of low-population density. It is 5 ft above river level, which is assumed to have negligible variation. The site occupies an area of grass-covered level terrain that is clear of existing structures and is situated within 30 to 50 miles of a population center. It is assumed that no easements are necessary. The land adjacent to the site is used for forest, livestock, or cultivated crop farming, except for railroads and highways.

5.7.1.2 Access. Secondary roads in good condition and a railroad spur are available to the site boundary. The river is navigable throughout the year for boats with up to a 6-ft draft. A barge unloading dock is located relatively close to the site for unloading heavy equipment.

5.7.1.3 Utilities. The river flow is fresh water and provides an adequate source of raw makeup and condenser cooling water for the ultimate station capability. The average maximum temperature is 75°F and the average minimum is 40°F. Condenser cooling water chlorination is required. Construction power, natural gas service, and communication lines are available to the site boundary.

5.7.1.4 Meteorology and Climatology. 1. Prevailing Wind Variation. Prevailing surface winds in the region surrounding the site blow from the south through west quadrant at speeds varying from 4 to 15 mph throughout the year. There are no large daily variations in wind speed, or direction. Observations of wind velocities at various altitudes indicate a gradual increase in mean speed and a gradual shift in prevailing wind direction from southwest near the surface to westerly aloft.

2. Temperature Ranges. The daily average temperature ranges between 40 and 60°F, with a design maximum of 90°F and a design minimum of 30°F.

3. Frequency of Temperature Inversions. Surface-based atmospheric inversions occur frequently during summer and early fall nights with clear skies and low wind speeds. These inversions are destroyed quickly by solar heating. Inversions occurring during winter or spring are more likely to extend into the daytime. Inversions occur most frequently when the winds flow from the north or west. Stagnation periods with steady light winds and a high frequency of inversions are most probable from August to October. A persistent inversion with its base between 1000 and 4000 ft, wind speeds less than 5 mph below 5000 ft, and clear skies that permit the formation of surface-based inversions at night are characteristic of these periods. The annual average percentage of time with inversions is 50%.

4. Frequency and Severity of Disturbances. A maximum wind velocity of 100 mph has been recorded at the site.

5. Snow Load. The snow loading design specification is 30 psf.

5.7.1.5 Hydrology. 1. Precipitation. The average annual rainfall at the site is over 27 in. per year.

2. Drainage. Natural drainage of the site is provided by the land contours. The subterranean water travels toward the river at a velocity of 300 ft per year. The maximum temperature is 75°F, with sufficient flow available to prevent exceeding the allowable temperature rise specified by state regulations. Dewatering and pilings are not required.

3. Ground Water. Ground water in the region collects mostly in the weathered layer of the shale above the bedrock. Adequate ground water for sanitary supply and plant makeup is available within 50 ft below grade. Most wells in the region are drilled to the shale layer.

5.7.1.6 Geology and Seismology. 1. Soil Profiles and Load-Bearing Characteristics. Soil profiles for the site show alluvial soil and rock fill to a depth of 8 ft, Brassfield limestone to a depth of 30 ft, blue weathered shale and fossiliferous Richmond limestone to a depth of 50 ft, and bedrock over a depth of 50 ft. The allowable soil bearing is 6,000 psf, and rock-bearing characteristics are 18,000 and 15,000 psf for Brassfield and Richmond strata, respectively. No underground cavities exist in the limestone.

2. Seismology. Zone 1 earthquake conditions, as designated by the Uniform Building Code, exist at the site.

5.7.1.7 Radioactive Waste Disposal. 1. Sewage. All sewage must receive primary and secondary treatment prior to being dumped into the river.

2. Volatile Wastes (Radioactive and Toxic Gas). Maximum permissible concentrations or dosages are as prescribed in AEC Standards for Protection Against Radiation, as published in the Code of Federal Regulations, Title 10, Part 20 (10 CFR 20).

3. Liquid Wastes. The maximum permissible activity of water entering the river is as prescribed in 10 CFR 20. The activity level of the liquid effluent is measured as it leaves the plant. No credit for dilution in the river is assumed.

4. Solid Wastes. Storage of solid wastes onsite for decay is permissible, but no ultimate disposal will be made onsite.

5.7.2 Plant Arrangement

Both the backup and reference design 1000-Mw(e) HTGR stations consist basically of

1. a reactor building approximately 162 ft long by 101 ft wide by 248 ft high for the backup design and 226 ft high for the reference design that contains
   a. an HTGR nuclear steam supply system within the prestressed concrete reactor vessel,
   b. a general operating area,
   c. fuel handling, storage, and shipping facilities,
   d. decontamination and radioactive waste disposal equipment,
2. a turbine building with approximate overall dimensions of 285 ft long by 159 ft wide by 98 ft high for the backup design and 267 ft long by 159 ft wide by 98 ft high for the reference design that contains
   a. a tandem-compound six-flow turbine-generator with condensing, feedwater, and other auxiliary systems,

   b. a general service area that provides space for a machine shop, auxiliary steam systems, and administration,
   c. reactor plant ventilation equipment and controlled personnel access to the reactor building,
   d. control room and area for miscellaneous electrical services,
3. a crib house and associated equipment.

Areas not requiring radiological control of access are entered through the general service structure adjacent to the main turbine building. Areas requiring radiological control of access are entered through the monitoring station located in the access-control area. Personnel decontamination rooms and clean-clothing storage are also provided in this latter area. Other facilities located in the access-control area adjacent to the reactor building include a health physics and first-aid station and locker room facilities.

Service facilities and auxiliary systems that are associated with the reactor are located in the reactor building. Positioned under the operating floor level in the reactor building are the fuel storage area, storage facilities for various pieces of equipment, the loading port for the fuel shipping cask, and a hot service facility for decontamination and/or servicing equipment that could become contaminated.

The radioactive-gas and radioactive-liquid waste systems are located in the reactor building. These systems are designed to accept all effluents from the reactor that could be contaminated. The gas waste system includes surge tanks to collect radioactive gaseous effluents for analysis and identification of radioisotopes prior to controlled release through filters to the plant stack. The liquid waste system provides for collection and monitoring of aqueous wastes, with subsequent processing as required to permit disposal of the effluent water at the plant site.

The plant arrangement is shown in Fig. 5.10.

5.7.3 Reactor Building

5.7.3.1 General. The reactor building is designed as a controlled leakage structure with a slightly negative internal pressure maintained by operation of the reactor plant ventilation system. Building design

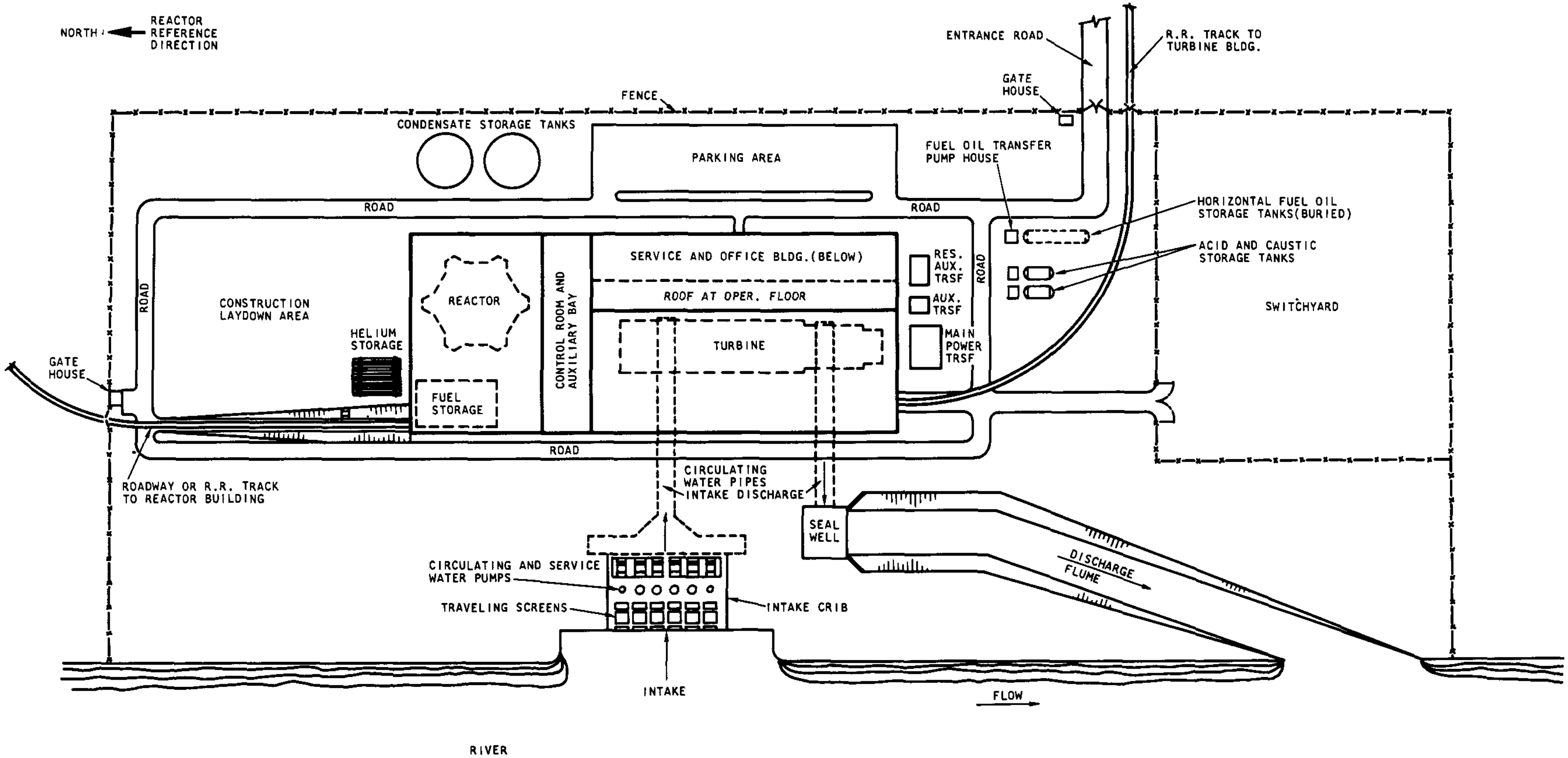


Fig. 5.10. Plot Plan for 1000-Mw(e) HTGR Station. (GGA illustration)

features include: (1) restricted leakage construction with conventional building materials through the use of appropriate construction joints and seals and (2) the elimination of essentially all windows and louvers. Leakage collected from all systems in the reactor building is released via the stack only after being processed by the reactor plant ventilation system.

An overhead crane of 120-ton capacity is provided in the reactor building.

Shielding is designed so that the plant can normally be operated and refueled without operating personnel receiving radiation doses in excess of 50% of the limits prescribed in 10 CFR 20 based on an 8-hr daily shift. The remaining 50% of the 10 CFR 20 limits are reserved for maintenance operations.

The reactor building is shown in Figs. 5.11, 5.12, and 5.13.

5.7.3.2 Reactor Plant Ventilation System. The reactor plant ventilation system provides filtered and heated outside air to all plant locations, with the exception of the control room, administration offices, and turbine room, which are separately ventilated. The ventilation system is composed of five separate ducting arrangements that supply the following areas:

1. access control area,
2. instrument room,
3. reactor building refueling floor,
4. reactor building lower level,
5. PCRV area.

The ventilation system is sized to provide a sufficient number of air changes for personnel comfort based on void air space and expected occupancy of each area. The access control area and the instrument room are supplied with a minimum of six air changes per hour. The refueling floor and lower reactor systems and PCRV areas are provided with one to two air changes per hour.

Pressures are normally maintained slightly negative with respect to the outside atmosphere.

A main exhaust header collects the discharge from the five exhaust headers serving the access control area, instrument room, refueling floor,

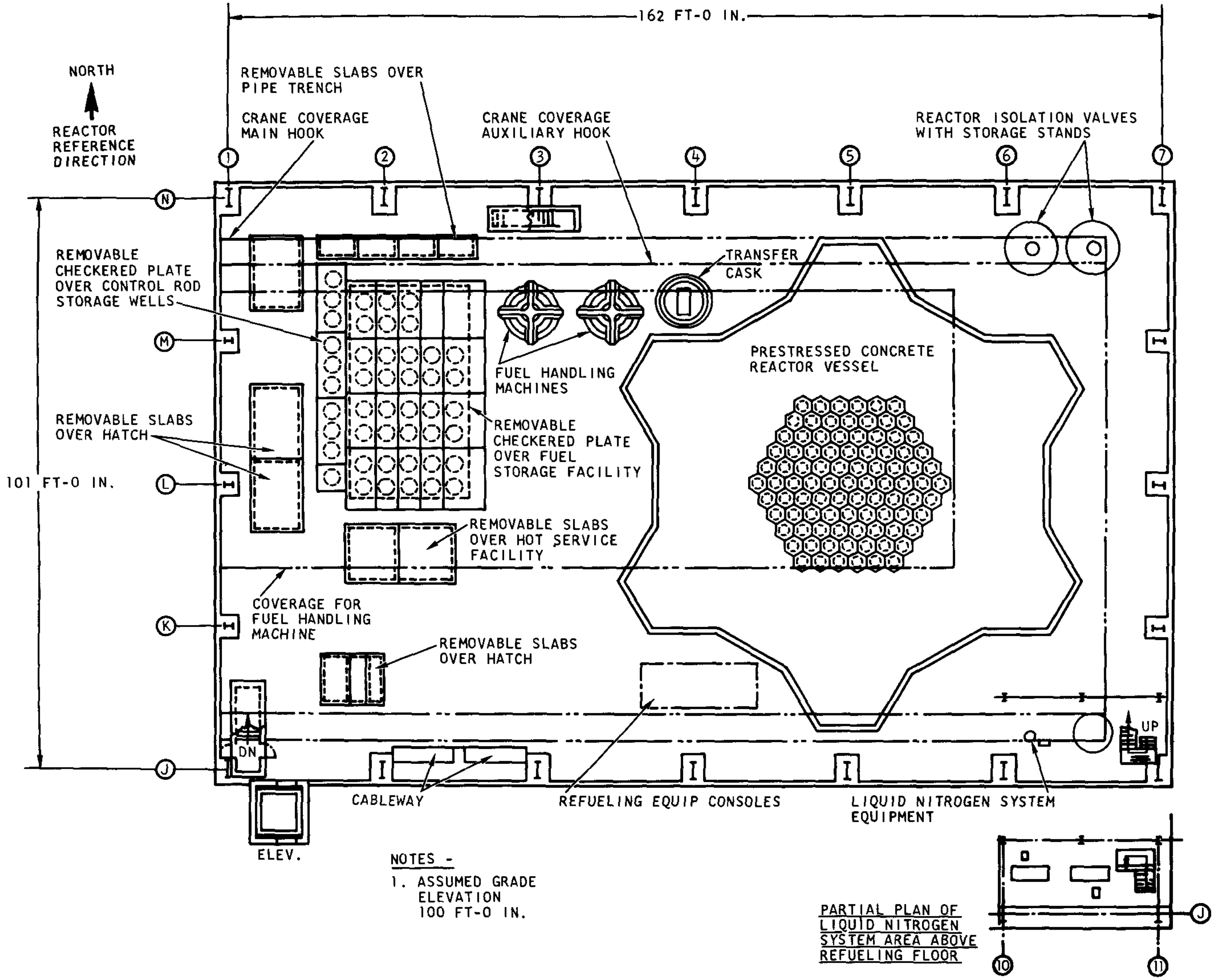


Fig. 5.11. Reactor Plant Arrangement, Refueling Floor. (GGA illustration)

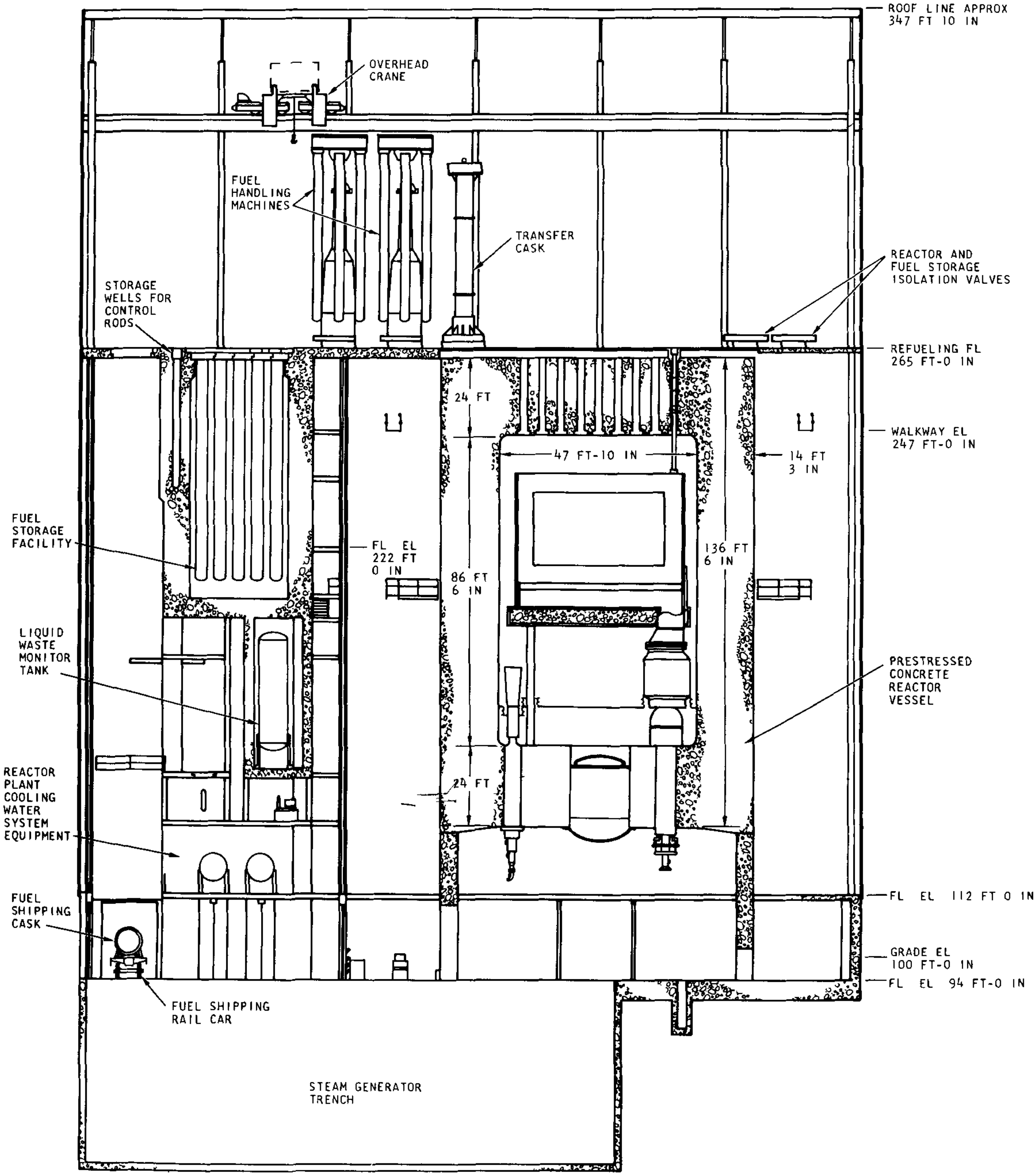


Fig. 5.12. Reactor Plant Arrangement, Section. (GGA illustration)

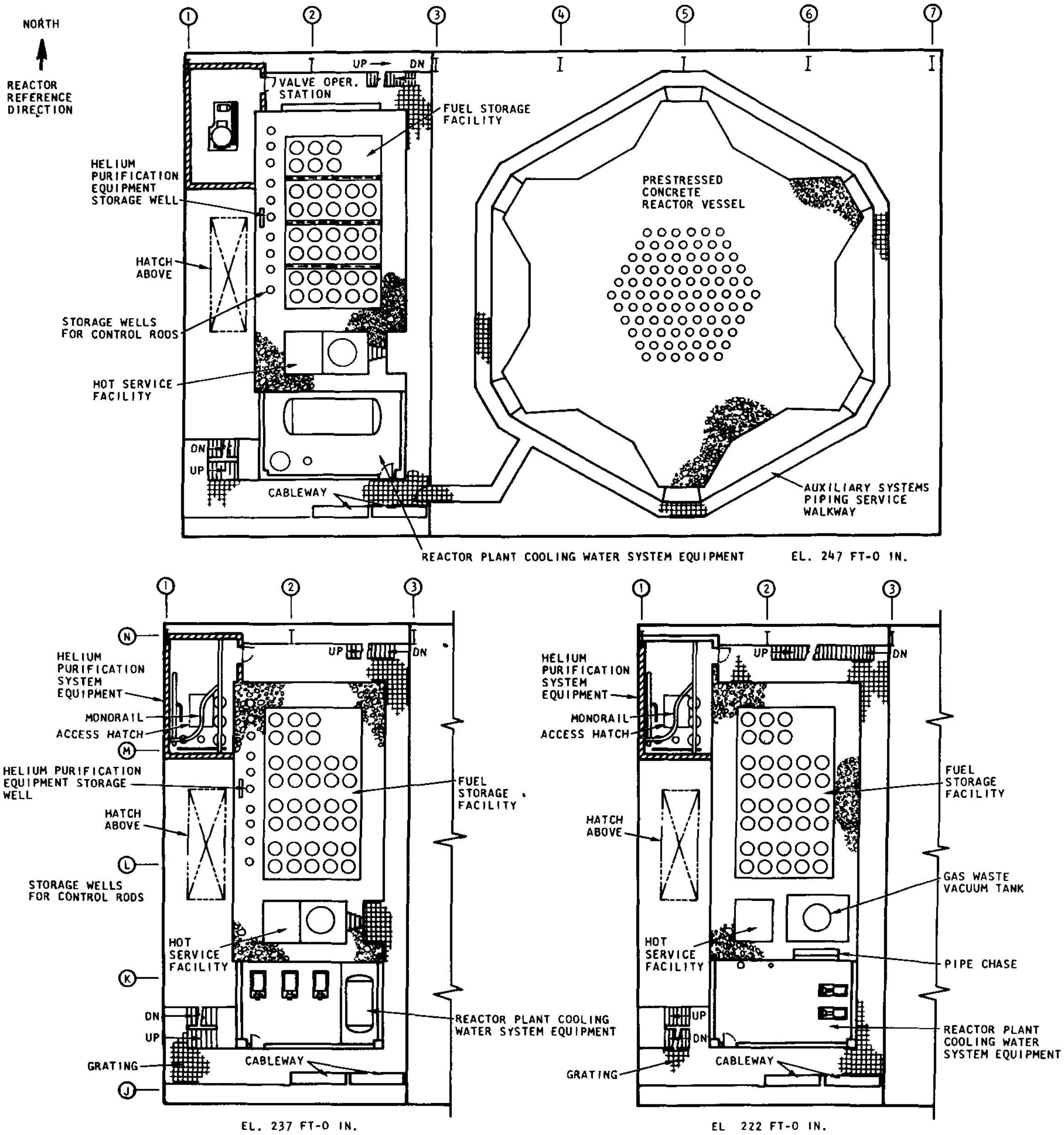


Fig. 5.13. Reactor Plant Arrangement, Twin Unit Plans. (GGA illustration)

PCRV area, and lower reactor systems. In addition, exhausts from the hogging and steam jet air ejectors, hot service facility, and gas waste system are discharged to the main header. From the main exhaust header, the air is drawn through five 25%-capacity filters (one standby) by five 25%-capacity exhaust fans (one standby). The plant exhaust filters incorporate a prefilter followed by an absolute filter and a charcoal filter to remove 99.9% of all particulates 0.3 μ or greater in size and 99% of all radioactive iodine. At least four of the five filters are always in service.

The plant exhaust air is monitored for radioactive particulates in the main exhaust ducts leading to the plant exhaust filters and for radioactive particulates, iodine, and noble gases prior to discharge to the atmosphere through the plant exhaust vent. It is expected that the plant exhaust filters will remove practically all but the noble gas contaminants from the air prior to being discharged. If, for any reason other radioactive gases or particulates should pass through the filters, radiation alarms from the monitors would be energized in the central control room and the air handlers would be throttled or stopped by the operator.

5.7.4 Turbine Building

5.7.4.1 General. The turbine building has reinforced-concrete foundations and structural-steel framing. Usual construction methods are specified. Rooms are provided to enclose the turbine lube-oil reservoir, turbine lube-oil storage tank, and purification system.

5.7.4.2 Heating and Ventilation. 1. Turbine Building Service Area. Perimeter rooms are heated by a hot-water system supplied with heat from steam taken either from the auxiliary boiler or from a low-pressure extraction point in the turbine-generator. An overhead duct system supplies year-round conditioned air to the administration areas, locker room, and shower room from a central air conditioner. The unit incorporates both heating and cooling coils, and provides proper outside air for ventilation. Cooling coils and spray are not employed on a common pipe arrangement.

Heating and ventilation are provided for other areas in the turbine building.

2. Control Room. An overhead duct system supplies year-round conditioned air from a central air conditioner. The unit incorporates both heating and cooling coils and provides proper outside air for ventilation.

3. Other Turbine Building Areas. Heating is supplied by hot-water-type unit heaters. The heaters are capable of maintaining 70°F when the outdoor temperature is 70°F. Roof exhaust fans supply ventilation air by pulling air in through windows and doors. Sufficient cooling capacity is provided to maintain an average temperature of 105°F with ambient outdoor conditions of 90°F.

4. Overhead Crane. The overhead crane has the following characteristics:

Capacity (main), tons	85
Span, ft	142

5. Turbine Building Drain System. A turbine building sump is provided. Two automatically operated motor-driven sump pumps (one standby) or a gravity drain system are specified. The choice is dependent on site conditions.

The turbine building arrangements are shown in Figs. 5.14 and 5.15.

6. Site and Building Evaluation. Site conditions are generally consistent with conditions for the hypothetical Middletown site. No rearrangement of layout for normalization purposes was found necessary.

The layout of the reactor plant was briefly reviewed. It appears reasonable and provides the necessary access for equipment removal.

Turbine building arrangements indicate a sufficient amount of laydown area. Assuming simultaneous overhaul of all sections of the turbine and removal of the generator, it will be necessary to remove the casings from the turbine building. The space provided is comparable to that of other concepts being evaluated.

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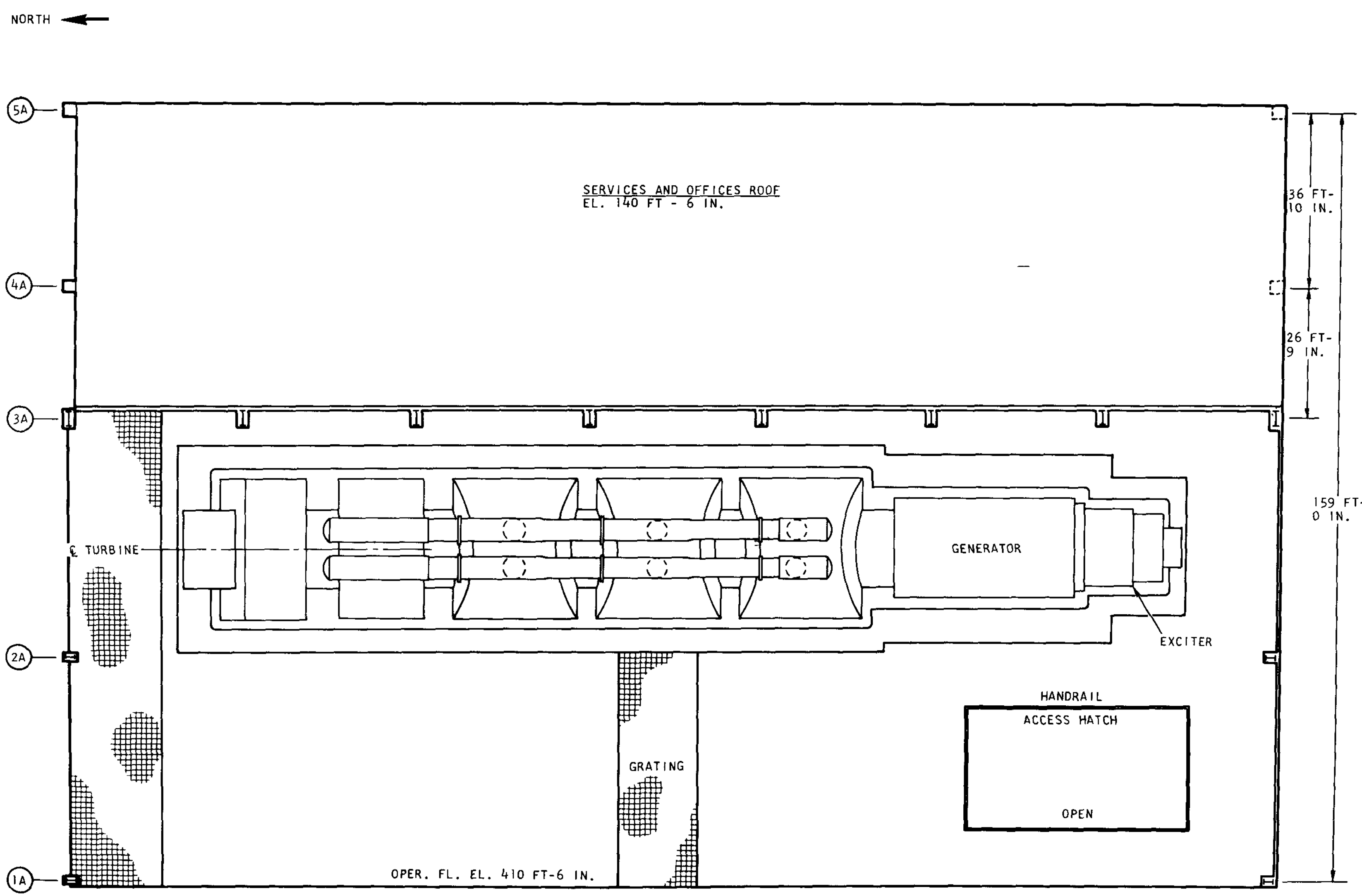

Fig. 5.14. Turbine Plant Arrangement, Operating Floor. (GGA illustration)

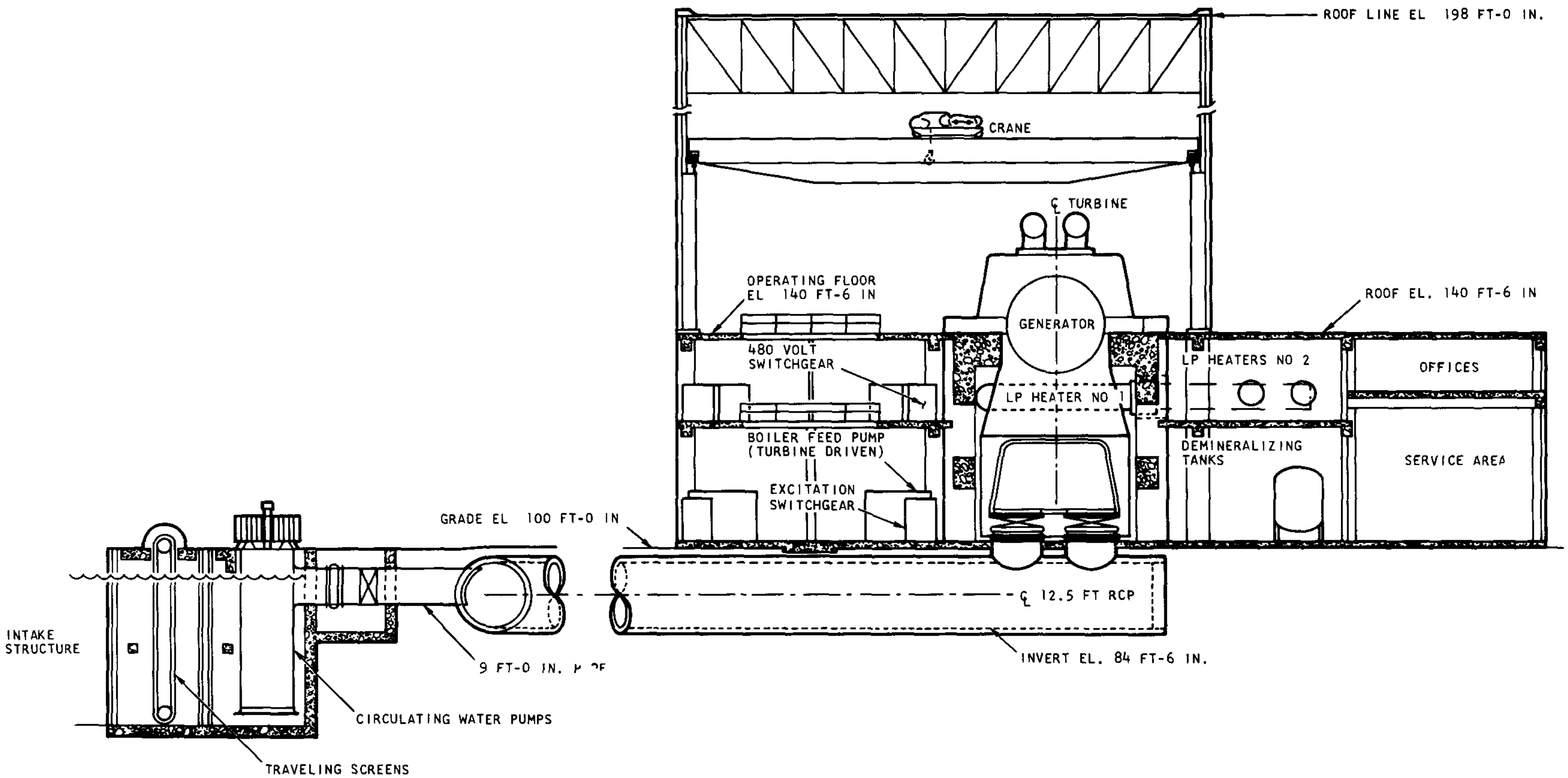

Fig. 5.15. Turbine Plant Arrangement, Section. (GGA illustration)

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6. ENGINEERED SAFETY FEATURES

The subject of engineered safety features is not specifically discussed in the reports on the twin 1000-Mw(e) HTGR stations; however, in principle, the safety features needed for a 1000-Mw(e) HTGR are the same as those for the Fort St. Vrain plant, which are covered in the PSAR and its amendments.[1] ORNL's review was made in the spring and summer of 1967 and it covered, except for some subsequent major revisions, the design as presented through Amendment 3 of the PSAR. Additional amendments were issued later, and a construction permit was issued in September 1968.

ORNL also reviewed written statements by GGA on compliance of the backup design with the 27 design criteria proposed by the AEC in 1965. These 27 criteria were superseded in 1967, and Amendment 10 of the Fort St. Vrain PSAR describes compliance of that plant with the revised, expanded criteria;[2] Amendment 10 was not available at the time ORNL's review was made. In addition to the GGA reports, ORNL also reviewed a study by Southern Nuclear Engineering, Inc.,[3] which discusses engineered safety features of a 3000-Mw(t) HTGR as a part of a nuclear desalination siting study.

6.1 Reactor Containment

Both the backup and reference designs, as well as the Fort St. Vrain design, have the prestressed-concrete reactor vessel, along with its primary closures, as the primary containment system. The secondary closures on the PCRV penetrations are considered by GGA to be secondary containment provisions. In addition, the reactor building serves as a vented confinement structure designed to accommodate and dissipate, without failure, the pressure associated with the maximum design leakage rate of the primary coolant system. Since the type of containment required for large HTGR's has not yet been established through the licensing process, the alternate of using a conventional pressure-tight containment building has been considered in the cost estimates.

6.2 Systems Essential to Containment

The reliability aspects of the systems whose operation is essential to the protection of the plant or the public were identified and evaluated. Most of these systems are associated with preserving the integrity of the containment system or preventing major damage to the reactor core. These systems can be broken down by function into the main groups of those for (1) shutting the reactor down, (2) shutting off sources of pressure buildup in the PCRV, (3) cooling the PCRV structure, and (4) forced cooling the core. The first three functions are essential for the preservation of containment integrity.

The PSAR indicates that forced cooling the core is not essential in the Fort St. Vrain design for the preservation of PCRV integrity. Depending on the PCRV leakage rate, operation of the air-cleanup system in the reactor building may become an essential system for protection of the public under some accident conditions. The core heat would be removed by conduction to the outer surface of the core, by radiation to the PCRV liner, and finally by the PCRV cooling coils. The temperatures would be low enough that the graphite core structure would remain intact. This mechanism of cooling has not yet been shown to be adequate for the larger 1000-Mw(e) designs, since the temperatures in the center of the core may go to values that would lead to failure of the graphite core structure. If careful study indicates that forced cooling of the core is not essential in the larger designs for protection of the public, this would be an attractive "selling" feature of the HTGR design, since the reliability requirements for this group of systems could be based on economic rather than safety considerations. However, economic considerations of preventing major damage to the reactor core may also require high reliability for the forced cooling systems. For this review, ORNL assumed that all four groups of systems are essential and that they must be designed to meet safety-system standards,[4] which require that a single failure not prevent the successful operation of the system. The designs presented for many of these systems include redundant active components, such as prime movers, pumps, valves, etc. In addition, all the corresponding instrumentation, controls, piping, and emergency power supplies are redundant and independent where necessary to meet the single-failure criterion.

The reports for the 1000-Mw(e) reference and backup designs do not present detailed designs for these systems, so for this review, these systems, as well as the instrumentation and controls, are considered to be based on the engineering features of the 330-Mw(e) Fort St. Vrain Nuclear Generating Station, as described in the PSAR.

A portion of these comments is directed to features such as the instrumentation in protective systems that could be easily modified in the detailed design stage of a 1000-Mw(e) HTGR plant and would involve little or no cost penalty, whereas modification of other features would involve more substantial changes. The ORNL reference cost estimate does not include a penalty for safety system equipment in addition to that specified by GGA, since it appears that the items involving major additional expense (mostly valves and piping) have not been required for some reactors built to date. However, ORNL made a rough estimate (see Section 7.1.2) of the maximum amount by which the capital costs could increase if all the additional items discussed below had to be included.

6.2.1 Reactor Safety System and Scram Mechanism

A reactor shutdown will be required to preserve the integrity of the prestressed-concrete reactor vessel (PCRV) in the event of cooling system failures or large steam-generator ruptures.

6.2.1.1 Safety System. The current reactor shutdown system design has two scram buses that each serve one-half the rod drive brakes. A separate logic matrix is used for each of these scram buses. It is understood that under many operating conditions in Fort St. Vrain, one-half the rods (on one scram matrix) are sufficient to shut the reactor down. Reliability might be improved by further splitting the rods into three or four groups so that the reactor could be shut down for all conditions after the failure of one matrix and its group of rods.

The protective system has six neutron-flux channels as inputs. These are spaced equally around the edge of the core and are arranged in the protective system logic in such a way that they give protection against large flux tilts that might arise from a rod runaway in one edge of the core, as well as the usual protection against overall high flux. GGA indicated that this spatial protection is not essential for the safety of the Fort St. Vrain reactor; however, it is felt that this is an attractive feature, and experience with this type of spatial protection will be useful in the development of protective systems for 1000-Mw(e) reactors where spatial protection may be more essential. The spatial protection offers the economic advantage of reducing the chance of localized damage to the core. The two sets of the flux instrumentation will be of different design, so this adds some diversity to the high flux protection.

The protection system does not have a scram on high reactor coolant outlet temperature or a scram on a high ratio of reactor power to coolant flow as is used on many power reactors. GGA transient studies[5] have shown that these types of protection signals are not needed for fuel overtemperature protection because of the high heat capacity, high temperature capabilities of the fuel, and the large negative temperature feedback features of the HTGR. The Fort St. Vrain design provides a scram on low flow indirectly through the loop shutdown and circulator shutdown protection systems.[5] A scram occurs if more than two circulators are tripped. In addition, the circulators are tripped on low speed, with the low-speed trip point varied as a function of the feedwater flow; and the circulators are tripped on low feedwater flow, with the low-flow trip point varied as a function of the circulator speed.

Since steam flow and hence feedwater are indicative of reactor power (if steam temperature and pressure are held within reasonable bounds), the variable circulator low-speed trip and the reactor scram insure that a reasonable amount of coolant flow is available to allow the reactor to continue to run at power. The scram on high reheat steam temperature is used in conjunction with the scram on shutdown of three circulators and the circulator shutdown on circulator low speed and low feedwater flow to limit fuel temperatures.

There are some reservations regarding this indirect approach, since the reactor overtemperature protection depends on the correct action of a combination of trip signals. This approach does give more direct protection based on the steam generator temperatures, which appear to be more critical than the fuel temperatures. There are also some reservations about tripping the circulators with the low speed and low feedwater flow trip points, since this cuts off core cooling; however, it protects the steam generators, which appear to be more vulnerable to damage than the core.

6.2.1.2 Control Rod Drives. The cable-and-drum control rod drives have an extremely slow scram insertion time (152 sec for the Fort St. Vrain design) as compared with other gas-cooled reactors (1 to 5 sec). This slow scram insertion speed (or even manual operation of the reserve shutdown system) seems to be adequate for the accidents examined in the PSAR and is related to the fact that there are no mechanisms that could produce rapid reactivity addition. The slow speed reduces the problem of slowing and stopping the rod at the end of its stroke and the complexity of the drive.

The dynamic-retarding method (capacitors across the motor windings) of controlling rod speed during the scram appears to be a nonstandard technique in motor control practice; however, it offers the attractive advantage of simplicity. This design has been tested under various operating conditions and component failure modes (such as failures in the loading capacitors) to demonstrate that the retarding system probably could not prevent a scram or significantly slow the insertion of more than a few rod pairs.

The falling rod must turn all the drive mechanism during a scram. The inertia and friction in the drive system will tend to lengthen the time to attain constant scram velocity; however, this is insignificant in comparison with the long scram insertion time. Tests performed in helium indicate that the frictional resistance of the drive mechanisms would be small enough that the somewhat unpredictable changes in friction would not prevent a scram.

The rods are scrammed by deenergizing and releasing the motor brake rather than by declutching the rod from the motor (in the usual manner). Thus, in addition to deenergizing the brake, the safety system is required to insure that no electrical power is applied to the drive motors in the direction of rod withdrawal to prevent the motors from overpowering the scram action.

The reactivity accidents examined in the original PSAR assume that only one rod can "runaway." Interlocks and a load sensor are used to prevent the withdrawal of more than one rod at a time. If the one-rod-runaway criterion is essential to the protection of the reactor, these circuits and devices must be considered to be part of the safety system and designed to those standards. They would have to cope with the gang rod control switches (probably intended only for rod insertion or withdrawal during startup) and three-phase motors that are capable of running backward if the phase of the electrical system is reversed. Studies reported in PSAR Amendment 2 indicate that the automatic scram system can adequately cope with an all-rod-runaway accident, so it may be possible to relax the one-rod-runaway criterion. The withdrawal of a large number of rods would, however, reduce the time available for manual actuation of the reserve shutdown system if the automatic shutdown system should fail.

The cable drums do not appear to have a mechanical stop when the rod reaches the fully inserted position. During a motor-driven insertion, failure of the limit switch that stops the drive motor when the rod reaches the fully inserted position would allow the motor to continue to rotate the drum. This would wind the cable up in the opposite direction and withdraw the control rod with the motor running in the insert direction. This problem is not serious if a systematic failure of the limit switches on several rod drives is made incredible.

6.2.2 Reserve Shutdown System

As discussed earlier, an infallible reactivity shutdown is required to preserve the integrity of the containment system. There is no inherent shutdown mechanism (although the negative temperature coefficient would limit the power and reactivity), so an applied shutdown mechanism that is different and independent from the control rods is needed. For this purpose, the design provides an independent reserve (or secondary) shutdown system that drops granules of poison into the core. The reactor is basically stable, and the high-capacity and high-temperature capability give a considerable longer time margin than in water reactors before a shutdown is absolutely required. However, this time margin appears to be only a matter of a minute or several minutes, as indicated by the 1 1/2-min delay in the one-rod-runaway accident in the PSAR. ORNL thinks it is undesirable to force the operator to make a decision to activate the reserve shutdown system in this short time interval under the emotional pressure of a possible accident situation and the economic consequences of an unnecessary shutdown, and that therefore it would be better to use a highly reliable automatic protection system to initiate the shutdown.

The reserve shutdown system has the reliability features of having individual gas-pressurizing bottles for each hopper and of being broken into two redundant groups with separate actuator control systems; however, it suffers from lack of ability to test the actual release of the granules under reactor operating conditions. A few of the hoppers are removed at the annual shutdown and the release is tested in the hot service facilities; however, this is not as conclusive as testing the release of the hoppers in the reactor. Some consideration might be given to easier methods of removing the boron granules when they are dropped into the core (such as those on the New Production Reactor) and to the possibility of testing the complete system in the reactor.

6.2.3 Automatic Loop Shutdown

A tube rupture in a steam-generator module would be the main source of overpressure in the PCRV, and it would be essential to shut off the feedwater and desirable to drain the water from the steam generator. The automatic loop shutdown protective system supplied for this purpose is complex and requires a considerable number of valves and controls to function correctly; however, the philosophy behind the design of the hardware and diverse backup devices appears to be good.

The automatic shutdown and the dump of coolant loops present a difficult problem because failure to shut down when necessary because of a large steam generator leak and spurious shutdowns of the reactor cooling system would both be undesirable from safety and economic considerations. The better of the two alternatives must be selected to form the basic criteria for the design of the protective system. GGA has based the design of this protection system on the criterion that the spurious shutdown of a coolant loop is a more serious problem than the failure to isolate a loop when isolation is actually needed during a large steam generator leak. In accordance with this design basis, an energize-to-trip logic along with local coincidence has been employed to reduce the possibility of spurious loop shutdowns at the expense of reducing the probability that the system will function when a shutdown is needed. Dual sets of logic and power supplies are used to increase the reliability of the system functioning when needed. This type of protection system design is used for the majority of the engineered safety features in current light-water-cooled power reactors.[6]

A failure in a circulator bearing system could also lead to the injection of water into the PCRV; however, the rate would be considerably smaller than that from a failure in a steam generator (~2 lb/sec versus 35 lb/sec). Amendment 2 to the Fort St. Vrain PSAR indicates that the detection of a leak from this source will result in automatic shutdown of the circulator, followed by isolation of the water and helium lines in the bearing and buffer seal systems. This action will also block the isolation of the steam generator when the moisture detector in that loop senses a high moisture level. This additional protection system increases the complexity of the automatic loop shutdown system but is worthwhile, since a steam generator dump is a very serious transient and false dumps should be avoided.

6.2.3.1 Feedwater Shutoff Valves. The feedwater isolation valves are critical items in the isolation of a faulty steam generator. In the Fort St. Vrain reactor, two valves are placed in series in the feedwater line. One is a block valve and can be tested only at scheduled shutdown, and the other is the flow control valve that is continually exercised but cannot be tested for complete closure except at shutdown, which is normally once a year. In the 1000-Mw(e) plant it may be necessary to place some of these isolation valves in a matrix arrangement to allow them to be tested during reactor operation. (Conventional steam turbines usually have two stop valves in parallel to allow them to be tested, or exercised, during operation.) There is an additional valve at the outlet of each feedwater pump that can be tested on-line.

The accident studies are based on fairly rapid closure of the feedwater valves. The effects of extending this time for closure should be examined to see if manual closure of the feedwater lines would be tolerable. This might reduce the requirements on the automatic operation of the feedwater isolation valves. Also PSAR Amendment 2 indicates that the steam-generator dump valves on the feedwater lines will divert a tolerable portion of the feedwater to the dump if the isolation valves fail to close. This will probably allow time for manual operation of the feedwater valves or pump shutoff.

6.2.3.2 Superheater Block-Check Valves. The block-check valve in the superheat header of each steam generator must close to prevent steam flow from the good steam generator from backflowing into the core through the faulty steam generator, although the steam inleakage rate from the steam end of a subheater rupture is much less than that from the feedwater end. These valves must close automatically to make isolation of a single steam generator completely effective; however, if they initially fail to operate correctly they can be manually driven shut, or all the steam generators can be dumped. The valves in the faulty steam generator must be closed before the others can be restarted. The Fort St. Vrain design provides only one valve for each steam generator. Again it may be necessary to use two valves in series or a matrix of valves for increased reliability and on-line testability in HTGR's of 1000 Mw(e).

6.2.3.3 Moisture Detectors. The dew-point moisture detectors are complex devices to be used in a protective system since they have several active components, such as refrigeration, heating, and optical systems; however, GGA has examined several types of detectors and, based on their testing and development work on the dew-point device, feels that this is the most suitable type. The selection of this type of instrument cannot presently be verified, but GGA's approach to the problem seems sound. There has been good experience with commercial dew-point moisture detectors in other types of applications. Reliability has been improved by providing for injection of moisture for on-line testing, placing each monitor in separate PCRV penetrations, and moving the photocells outside the PCRV. The moisture monitors are accessible during reactor operation from outside the PCRV.

It may be possible to reduce the requirements on the moisture detectors by depending on the simpler PCRV relief valves for the public safety protective action and to consider the moisture detectors as devices for "economic" protective action to prevent core damage. The moisture detectors would still need to be reliable to avoid core damage and inadvertent isolation of the steam loop.

6.2.4 PCRV Overpressure Protection

The overpressure protection of the PCRV appears to depend for a first line of defense on the complex automatic loop isolation system to eliminate the major cause of overpressure by shutting off the incoming water or perhaps manual action in a reasonable time. The automatic loop isolation system has diversity in actuating signals but a single type of protective action in shutting off the incoming water. The recent addition of the PCRV relief-valve for safety-valve system offers a second type of protective action that is needed for the important function of preventing PCRV overpressure that comes from water inleakage or any other (unknown) source. Relief valves are also a more basic protection mechanism than moisture detectors. The PCRV itself may act as a relief "valve" for cracks in the liner; however, economic considerations suggest the use of actual relief valves.

In the Fort St. Vrain design, the safety valves are vented through filters to the top of the reactor building. Some more restrictive measures might be necessary for the 1000-Mw(e) reactors.

6.2.5 Emergency Cooling

The HTGR design with the steam generators located beneath the core does not provide sufficient cooling by natural circulation to remove the afterheat, and it is necessary to provide forced cooling. Also the design does not provide any specialized forced-helium-circulation emergency-cooling loops,[7] but rather it relies on the multiplicity of main coolant loops and heat exchanger surfaces to provide forced-circulation afterheat cooling. The designer has gone a long way in making these systems reliable, such as providing several forms of circulator motive power; however, since the main loops are located inside the PCRV and exposed to the environment of the accident, careful attention must be given to sources of systematic failures in the circulators or steam generators.

The HTGR has a considerable advantage over water-cooled reactors in that it has a high heat capacity in the core, and the fuel elements have very good fission-product retention properties and do not experience sudden cladding failures as do metal-clad fuel elements. The adiabatic heatup rate for the core at full power with no heat removal is 5°F/sec, as compared with approximately 100°F/sec for current light-water reactors. The PSAR indicates that the Fort St. Vrain reactor can tolerate a complete loss of forced circulation for 30 min before the fuel temperature rises to its normal operating condition and then, if pressurized, requires the operation of only one of the four main circulators. After cooling is supplied for several hours, the forced cooling can be interrupted for 10 to 18 hr before it must be resumed. This time margin may be reduced somewhat following a depressurization accident. (The PSAR indicates that about 4.4% of total full-power mass flow is needed to level the temperatures off after the 30-min no-cooling period with a peak fuel temperature of 1880°F. During a maximum credible depressurization with a time constant of 1600 sec, a single circulator should produce about 4 to 6% of total full-power flow at the end of the 30-min no-cooling interval; however, this single circulator flow will drop to about 1.2% when pressure falls to atmospheric in another 1 1/2 hr.) This time margin during a depressurization may be reduced further in the 1000-Mw(e) designs, since a single circulator delivers only 16.7% of the total mass flow instead of the 25% in the Fort St. Vrain design; however, the time margins should still be in the region of 15 to 30 min. These time margins are important because they reduce the requirements on the emergency cooling system in comparison with those of many other reactors. This margin allows time for manual operations and for repair of equipment located outside the PCRV.

6.2.5.1 Primary Coolant Shutoff Valves. The shutoff valves for the coolant circulators are critical components, since they must open correctly when the circulator is running and they must close when the circulator is not running to prevent backflow from short circuiting the flow from the operating blowers and reducing (or possibly stopping) the flow through the core. The valves may be designed so that they cannot stay closed against a running circulator. The amount of backflow for various types of valve failures is discussed in PSAR Amendment 2.

6.2.5.2 Circulators. Only one of the four main circulators in the Fort St. Vrain design is required for afterheat cooling. The 1000-Mw(e) designs offer greater redundancy in circulators, since only one of the six main circulators is required for afterheat cooling.

6.2.5.3 Circulator Motive Power. The circulators have redundant forms of motive power that should be adequate, especially with the long time margins allowed. The circulators have steam-driven turbines that can run for about 1 hr (in the Fort St. Vrain design) after a scram on steam supplied from the reactor through the bypass tank system, if electrical power is available for auxiliary systems. After that time, steam can be supplied by the auxiliary package boiler, or the circulators can be driven by the water turbines on water supplied by the feedwater pumps, condensate pumps, feedwater pumps for the auxiliary boiler, or the firewater pumps.

The steam and water turbines must be designed to withstand any disturbances caused by switching from one form of motive power to the other.

6.2.5.4 Circulator Integrity. Since the main circulators are the only source of forced cooling, several situations must be examined to insure that it is not possible to damage or destroy all the circulators with temperature or pressure transients. If any of the following situations can lead to systematic damage of the circulators, either the mechanical design of the circulators must be improved or the instrumentation and control systems must be of safety-system standards.

1. Bearing Water System. The circulators probably would be destroyed[8] before they could coast to a stop if the bearing water supply was interrupted. If this is true, the bearing water systems become very essential and must be designed to safety-system standards. The design provides a separate bearing water system for each coolant loop with redundant active components on standby. A surge tank is supplied for each of the coolant loop systems to give some continuity in water supply; however, makeup water for all loops is supplied by a single emergency feedwater line and pressure regulator. A backup makeup water supply is provided from the condensate storage tanks; however, a single pump serves all the loops. The emergency feedwater line serves as a backup for the

normal bearing system by supplying water to the circulator bearings directly; however, a single line serves all the coolant loops. Since the bearing water systems are important to the circulator integrity, ORNL believes that additional redundancy should be provided in the piping, makeup water supply systems, and controls. As an alternative to redundancy in the systems serving the two circulators in a loop, it may be desirable to provide a separate bearing water system for each circulator.

The failure of the automatically controlled valves in the bearing drain system might prevent bearing lubrication or allow water to be pumped into the PCRV cavity or into the steam-turbine drives. PSAR Amendment 2 indicates that automatic isolation valves will be added to prevent water being pumped into the PCRV. The closing of these valves is delayed for 3 min after the circulator is tripped to allow time for circulator coastdown. Inadvertent closure of these isolation valves might lead to circulator damage.

2. Shaft Seal System. The rotating shaft seals are very complex; however, the Fort St. Vrain PSAR indicates that the failure of these seals would not harm the circulator or impair the cooldown of the reactor. This point should be carefully verified.

3. Parallel Operation of Circulators. The operation of axial flow circulators in parallel often presents problems with flow stability and surging. Axial flow circulators are usually more subject to these problems than are other types of circulators, and if they exist in this design, it may be necessary to employ elaborate control systems to hold all the circulators (or the circulators in one loop) within an allowable speed range. The circulators should be designed to withstand surging without deblading or suffering other damage.

4. Ability to Pump Heavy Gas Mixtures. The circulators should be designed so that they can pump mixtures of helium and dry steam for maintaining forced cooling after a steam-generator rupture.

5. Circulator Overspeed. Failure in a circulator speed controller or a steam-pipe rupture might lead to the destruction of a circulator. The circulators are designed to withstand an overspeed (of about 40%) corresponding to the rupture of the turbine exhaust steam pipe for some period of time. If the possible overspeeds can be destructive, the

protective devices should be of safety-system grade. The design provides four speed sensors for each circulator and two forms of overspeed protection: (1) the normal speed controller and control valve and (2) an overspeed trip circuit. One sensor serves the normal speed controller, and the other controls the stop valve through two dual two-of-three logic matrices. These devices provide redundancy in detection but not in valve closure.

6.2.5.5 Heat Exchangers. The steam generators used in normal operation provide multiple heat exchanger surfaces for afterheat and emergency cooling. In the Fort St. Vrain design, any one of four heat exchanger surfaces can provide adequate cooling after a scram. Two of these are provided by the economizer-evaporator-superheater sections in each of the two coolant loops. The other two are provided by the reheater sections in each of the two coolant loops. The 1000-Mw(e) designs offer more redundancy in heat exchanger surfaces with three cooling loops; however, the steam generators are only broken into two modules instead of the six in the Fort St. Vrain design and consequently there would not be as much opportunity of reusing part of the modules in a failed coolant loop.

6.2.5.6 Water Supplies. The steam generators are supplied with water from several sources. The Fort St. Vrain design has three feedwater pumps; whereas, the 1000-Mw(e) designs have only two feedwater pumps and somewhat less redundancy in pumping after a shutdown. Reactor steam can drive one of the turbine-driven feedwater pumps in the Fort St. Vrain design for about 1/2 hr after a scram; however, the turbine-driven feedwater pump in the 1000-Mw(e) designs is probably too large to be driven by reactor steam, and either steam from the auxiliary boiler or water from the electrically driven feedwater pumps would have to be used at an earlier time. The main feedwater pumps are backed up by the condensate pumps, feedwater pumps for the auxiliary boiler, and the firewater pumps.

The piping in the Fort St. Vrain design does not have the same redundancy as the pumping or sources of water. The design provides an emergency feedwater line with automatic switchover for the high-pressure piping between the boiler feedwater pumps and the steam generators. This redundancy is omitted for all the low-pressure piping ahead of the boiler feedwater pumps; however, an emergency condensate line can be used to bypass the feedwater pumps and supply water at a lower pressure from the condensate pumps, firewater pumps, or auxiliary boiler feedwater pumps to the steam generators. The use of a single emergency condensate line serving all the steam generators will compromise some of the redundancy provided by the sources that feed into this line.

6.2.5.7 Steam-Generator Integrity. Since there is no external protected emergency cooling system, several situations must be examined to insure that it is not possible to damage or destroy all the steam generators with temperature or pressure transients. If any of the following situations can lead to systematic damage of the steam generators, either the mechanical design of the steam generators must be improved or the instrumentation and control systems must be of safety-system standards.

1. Any Failure in the Plant Control System. For example, the automatic rampdown of the feedwater followed by the helium flow may be an essential action. The PSAR describes this as a "secondary safety action" and an "action to avoid possible damage."

2. Flow Instabilities in the Once-Through Boilers. Feedwater flow control and pressure control may be essential actions.

3. Reestablishing Feedwater Flow After a Loop Isolation

4. Switching to Steam Dump Operation

5. Switching to Water Feed from Condensate Pumps. The PSAR says the set point of the main steam bypass valves will be lowered slowly to reduce the thermal shock from the cooler condensate water.

6. Depressurization of PCRV. The steam generators must be able to withstand the maximum credible rate of depressurizing the PCRV. Amendment 3 of the Fort St. Vrain PSAR indicates that the main reactor components can withstand the forces of sudden depressurization. The steam generator tubes are stress analyzed for 0-psi external pressure, operating temperature, and full internal pressure. It would be difficult to provide tests to assure that the steam generator tubes could stand an increase in differential pressure (to 700 psi) at temperature throughout life. However, the tubes will be tested cold with helium at atmospheric pressure during shutdowns.

7. Failure of Primary Coolant Shutoff Valves to Close During Loop Isolation. Amendment 2 of the PSAR indicates that the temperature transients would not be excessive for such failures.

6.2.6 Emergency Power

The requirements for emergency electrical power are reduced in this type of reactor, since the reactor coolant circulators and possibly the boiler feedpumps (see Section 5.6) can be driven by steam produced in the reactor for about 1/2 hr; however, the circulators can operate on reactor steam for only a few minutes if electrical power for the auxiliary systems is not available. The circulators and feedpumps can also be powered by steam from the auxiliary boiler.

The 1000-Mw(e) designs do not provide any Diesel generators for emergency electrical power (unlike the Fort St. Vrain design). They do include a 9-Mw auxiliary or house generator that is on-line during normal operation and is driven by reactor steam taken from the intermediate pressure turbine. This generator is adequate to supply essential electric power for shutdown cooling, such as that needed for circulator bearing pumps, helium purification system pumps, PCRV cooling water pumps, a motor-driven feedwater pump, condensate pumps, etc. Since this generator is the only form of emergency power, it presumably can be driven by the auxiliary steam boiler during shutdown conditions. It will be necessary to add a standby auxiliary generator and a standby auxiliary steam boiler to provide adequate redundance for on-site emergency power.[2] GGA has stated that a 4.5-Mw unit would be adequate for the loads that are essential for afterheat cooling. Since most of the essential loads are redundant and would be split between the two sources of emergency power, it may be possible to reduce the size of the units still further. Also, reactor steam may be considered as one of the emergency power sources for some of the turbine-driven loads. For this evaluation, ORNL assumed that two 4.5-Mw units will be provided.

This type of reactor seems to have such a long time margin (1/2 hr) that it can do without forced cooling, so there may be a similar time margin before a large amount of electrical power is required. If some of the loads mentioned above cannot tolerate this long delay, it may be necessary to keep an auxiliary boiler in operation at all times or to provide battery sources of power.

6.2.7 PCRV and Core Support Structure Cooling

It appears to be essential to maintain water cooling to preserve the integrity of the PCRV structure and the core support structure. Failure of the core support structure might eventually allow the core to sag onto the steam generators. Both these structures can tolerate some interruption of cooling water, and this time margin (possibly several hours or days) would influence the requirements on the cooling systems. Two separate and redundant cooling systems are provided in the design. Each of these systems has redundant pumps and power supplies. The instrumentation does not appear to be redundant in these systems. However, there is sufficient time margin for manual control or replacement should the need arise. The piping headers are arranged so that failed tubes could be plugged, and the system could continue to be used.

The core support structure should be vented to prevent pressure buildup and possible explosions such as those experienced[9] with enclosed concrete shield plugs for beam holes in test reactors. Amendment 2 of the PSAR discusses a vent system.

6.2.8 Isolation of Lines Leaving the PCRV

In the event of pipe ruptures, several lines leaving the PCRV would have to be closed off or isolated to prevent the radioactive reactor coolant from entering the unshielded portions of the plant piping or entering the reactor building. It should be noted that by comparison with other types of reactors the integral design of the PCRV reduces the number of lines containing the primary coolant that leave the containment vessel and require provisions for isolation. This is one of the main advantages of using an integral design.

6.2.8.1 Reheater Steam Lines. A tube rupture in the reheater would allow the reactor coolant to enter the steam system. Radiation detectors on the reheater steam lines are arranged in two-of-three logic to give an automatic loop shutdown and to close a single isolation valve in the hot reheat header line. Depending on the consequences of failure, it may be necessary to replace the single isolation valve in each reheat header with two valves in series or a matrix of valves that would allow them to be tested (or exercised) during plant operation.

6.2.8.2 Helium Purification Lines. A pipe rupture plus failure of a closed valve (see mca described in the PSAR) in the helium purification systems could allow reactor coolant to enter the unshielded portions of the system or the reactor building. Double isolation valves are closed on high flow and radiation signals. The instrumentation controlling these valves should be of safety system grade, and provisions should be made for testing valve operation.

6.2.9 Exclusion of Air from Primary Coolant System

The air-graphite reaction is usually considered to be a serious problem[7] in gas-cooled reactors, and some reactor experiments have included a secondary containment shell with an inert gas between it and the primary container or a large inert-gas purging system. Although the HTGR design does not include either of these features, the PCRV has the advantages that the maximum credible leaks are small and that it seems impossible to have the simultaneous opening of two large flow paths to the outside of the PCRV that would be necessary for a significant flow of air into the core. Amendment 3 of the PSAR indicates that the normal purge gas to the PCRV penetrations, control rod drives, etc., is sufficient to prevent inleakage of air following the rapid failure of any penetration. These studies also indicate that the ingress of oxygen following a complete failure in the purge systems is not significant. The purge systems include standby helium compressors and can be supplied by either the helium purification systems, the helium storage system, or the nitrogen system.

6.2.10 Dynamic Containment System

The reactor building and ventilation system form dynamic secondary containment. The operation of the ventilation system is not essential for the protection of the public from the credible accidents in the Fort St. Vrain design; however, Amendment 3 of the PSAR indicates that it will become essential for the situation of extended loss of forced core cooling.

The system seems to be highly reliable, since it is in continuous operation and has standby blowers and filters, and it may be essential for the 1000-Mw(e) designs.

6.2.11 Monitoring Leaks in PCRV Penetrations

The limitation on the size of the maximum credible leak in the PCRV depends on the integrity of large penetrations. These large penetrations have two closures, with the interspace filled with helium. In order to claim "double" containment at these closures, it is necessary to detect the first leak. This is done by monitoring the total helium flow to the interspaces between the closures with redundant flowmeters. Manual means of observing the pressure changes in the interspace are used to determine which of the two closures is leaking.

For large steam generator units it may be difficult to provide adequate pressure relief to protect the penetration interspace cavity from overpressure caused by a steam line failure. The possibilities for reducing this problem by limiting steam line sizes or by providing other protection equivalent to double closures should be evaluated for these penetrations.

6.2.12 On-Line Refueling

The 1000-Mw(e) reference design specifies on-line refueling. This presents two additional safety problems that should be carefully considered.

6.2.12.1 Leak in the PCRV. The removal of a refueling plug during operation by the charge machine offers the possibility of opening up a large penetration hole in the PCRV. One or more valves will be required to make sure that the machine is functioning properly each time a refueling plug is removed. This system will require careful review, after it is designed, to determine whether it could experience depressurization rates that would lead to a more serious mca than has been previously identified.

6.2.12.2 Reactivity Effects. The movement of fuel during operation might offer the possibility of changes in reactivity. However, the refueling machine will only handle individual fuel blocks, which are worth a maximum of about ten cents of reactivity.

References

1. Public Service Company of Colorado, Fort St. Vrain Nuclear Generating Station, Preliminary Safety Analysis Report, Docket 50-267.

2. USAEC, General Design Criteria for Nuclear Power Plant Construction Permits, Federal Register, July 11, 1967.

3. C. Rogers McCullough et al., Siting Studies of Two Large Dual Purpose Desalination Reactors, Report SNE-33, Southern Nuclear Engineering, Inc., June 16, 1967.

4. Institute of Electrical and Electronics Engineers, Proposed Criteria for Nuclear Power Plant Protection Systems, IEEE No. 279, Aug. 30, 1968.

5. Personal communications from M. M. Dye and W. Lones, Gulf General Atomic, to H. G. O'Brien, Oak Ridge National Laboratory, February 1968.

6. H. G. O'Brien and C. S. Walker, Protection Instrumentation Systems in Light-Water-Cooled Power Reactor Plants, USAEC Report ORNL-NSIC-29, Oak Ridge National Laboratory, 1969.

7. G. C. Robinson, Emergency Cooling Systems in Gas-Cooled Reactors, Nucl. Safety, 6(4): 425–432 (Summer 1965).

8. Personal communications from A. Harris, General Atomic, to D. D. Cannon, Oak Ridge National Laboratory.

9. Reactor Safety Operating Experiences Bulletin, High-Pressure Release in an Eight-Inch Tube of the Puerto Rico Research Reactor, ROE: 66-3, USAEC, Oct. 25, 1966.

7. COSTS

7.1 Capital Costs

Capital cost estimates for the backup and reference design HTGR's are summarized in Table 7.1, and a more detailed breakdown of the cost is given in Table 7.2. All cost data were normalized to the construction cost levels prevailing in June 1967, with no escalation allowance. The totals are $123 million for the backup design and $122 million for the reference design under private financing. Methods employed in arriving at these estimates are similar to those used and described in other ORNL reactor evaluation studies[1,2] and consist essentially of the following steps:

1. The design parameters evaluated and discussed in Chapter 5 on the engineering review of the plant design are used as the bases for capital cost estimates.

2. Capital cost data presented by the designer are evaluated and compared with cost data on analogous systems or components of other reactor concepts being evaluated.

3. The designer's cost estimates are adjusted or new cost estimates are made when it appears necessary to do so for cost normalization.

4. Independent estimates of major components of the plant are made by using unit cost data developed for the ORNL evaluation studies.

5. Cost estimates of systems not clearly defined (such as auxiliary and service systems) are determined by analogy with systems of other reactor plants that serve similar functions and for which costs have been developed.

6. Direct construction costs are arranged and displayed in accordance with the AEC classification of accounts given in Table 106-2 of TID-7025, AEC Guide to Nuclear Power Plant Evaluation.

7. Indirect costs are based on methods outlined in TID-7025; however, percentages applied in determining these costs have been changed. The percentages used, which are applied to accumulative totals, are shown in Table 7.1.

Table 7.1. Estimated Total Capital Cost of 1000-Mw(e) HTGR Power Plant

	Basis of Indirect Cost (% of accumulated cost)	Backup Design	Reference Design
Direct construction cost			
Account			
21 — Structures and improvements		$ 8,255,000	$ 7,090,000
22 — Reactor plant equipment		48,117,000	49,245,000
23 — Turbine-generator units		25,625,000	24,233,000
24 — Accessory electric equipment		3,815,000	4,015,000
25 — Miscellaneous power plant equipment		1,250,000	1,250,000
Total direct construction cost		$ 87,062,000	$ 85,833,000
Indirect construction cost			
General and administrative	6.1	5,311,000	5,236,000
Subtotal		$ 92,373,000	$ 91,069,000
Miscellaneous construction	1	924,000	911,000
Subtotal		$ 93,297,000	$ 91,980,000
Engineering design and inspection			
Architectural and engineering services	5.1	4,758,000	4,691,000
Subtotal		$ 98,055,000	$ 96,671,000
Nuclear engineering	2.1	2,059,000	2,030,000
Subtotal		$100,114,000	$ 98,701,000
Startup costs		850,000	823,000
Subtotal		$100,964,000	$ 99,524,000
Contingency	10	10,096,000	9,952,000
TOTAL DIRECT AND INDIRECT CONSTRUCTION COST		$111,060,000	$109,476,000
Customer cost			
Interest during construction of investor-owned plant	10.8	11,994,000	11,823,000
TOTAL DEPRECIABLE CAPITAL COST		$123,054,000	$121,299,000
20 — Land and land rights		360,000	360,000
TOTAL INVESTOR-OWNED CAPITAL COST		$123,414,000	$121,659,000
Interest during construction of publicly owned plant	7.2	7,996,000	7,882,000
TOTAL DEPRECIABLE CAPITAL COST		$119,056,000	$117,358,000
20 — Land and land rights		360,000	360,000
TOTAL PUBLICLY OWNED CAPITAL COST		$119,416,000	$117,718,000

Table 7.2. Estimate of Direct Construction Cost of 1000-Mw(e) HTGR Plant

	Backup Design	Reference Design
Account		
21 — Structures and improvements		
211 Ground improvements	$ 1,000,000	$ 1,000,000
212 Buildings		
212A Reactor building	3,950,000	2,950,000
212B Turbine-generator including control room	2,015,000	1,900,000
212C Office and service building	540,000	540,000
212D Waste disposal	In 212A	In 212A
212E Fuel handling	In 212A	In 212A
212F Intake structures	700,000	650,000
212G Gate house and miscellaneous structures	50,000	50,000
Total cost, item 212	$ 7,255,000	$ 6,090,000
219 Reactor structure	None	None
Total cost, account 21	$ 8,255,000	$ 7,090,000
22 — Reactor plant equipment		
221 Reactor equipment		
.1 Pressure vessel including internal shields	$11,755,000	$ 9,400,000
.2 Reactor controls	2,995,000	2,265,000
.3 Reflector	2,400,000	2,400,000
.4 Cranes and hoists	200,000	50,000
Total cost, item 221	$17,350,000	$14,115,000
222 Heat transfer systems		
.1 Reactor coolant system circulators	$ 2,577,000	$ 2,320,000
.2 Steam generators	8,930,000	9,000,000
.3 Primary coolant receiving, supply and treatment	1,035,000	900,000
.4 Coolant inventory	225,000	150,000
.5 Auxiliary steam supply (oil-fired)	350,000	350,000
Total cost, item 222	$13,117,000	$12,720,000
223 Nuclear fuel-handling and storage equipment		
.1 Fuel-handling machine, service, and viewing equipment	$ 1,500,000	$ 2,500,000
.2 Fuel storage	1,200,000	720,000
.4 Gasks, special tools, etc.	500,000	500,000
Total cost, item 223	$ 3,200,000	$ 3,720,000
224 Fuel processing and fabrication equipment	Not included	Not included
225 Radioactive waste treatment and disposal	750,000	750,000
226 Instrumentation and control		
.1 Reactor	$ 1,500,000	$ 1,500,000
.2 Secondary and auxiliary systems	1,250,000	1,250,000
.3 Other reactor plant instruments and controls	250,000	250,000
Total cost, item 226	$ 3,000,000	$ 3,000,000
227 Feedwater supply and treatment		
.1 Raw water and makeup supply	$ 400,000	$ 400,000
.2 Purification (primary coolant)	850,000	775,000
.3 Feedwater heaters	1,510,000	1,800,000
.4 Feedwater pumps and drives (except condensate)	1,310,000	1,875,000
Total cost, item 227	$ 4,070,000	$ 4,850,000
228 Steam, condensate, and feedwater piping	$ 5,800,000	$ 9,260,000
229 Other reactor plant equipment	830,000	830,000
Total cost, account 22	$48,117,000	$49,245,000

Table 7.2 (continued)

	Backup Design	Reference Design
23 – Turbine-generator unit		
231 Turbine-generator	$20,192,000	$19,103,000
232 Circulating water systems	1,650,000	1,500,000
233 Condensers and auxiliaries	2,553,000	2,300,000
234 Central lubricating system	80,000	80,000
235 Instruments and controls	400,000	400,000
236 Piping	550,000	550,000
237 Auxiliary equipment for generator	750,000	750,000
238 Other equipment	125,000	125,000
Total cost, account 23	$25,625,000	$24,233,000
24 – Accessory electric equipment		
241 Switchgear	$ 500,000	500,000
242 Switchboards	375,000	375,000
243 Protective equipment	100,000	100,000
244 Structures	150,000	150,000
245 Conduit	300,000	300,000
246 Power and control wiring	1,300,000	1,320,000
247 Station service	450,000	630,000
248 Emergency power (auxiliary turbine-generators)	640,000	640,000
Total cost, account 24	$ 3,815,000	$ 4,015,000
25 – Miscellaneous power plant equipment	$ 1,250,000	$ 1,250,000
TOTAL DIRECT CONSTRUCTION COST	$87,062,000	$85,833,000

The cost data supplied by the designer were evaluated by comparing them with those of an independent cost estimate prepared by ORNL. During the evaluation the designer was contacted and discrepancies between the two estimates were discussed, but complete agreement was not reached on the over-all cost. GGA estimates of total (investor-owned) capital cost are $114.06 million for the backup design and $112.23 million for the reference design. These estimates are about 8% lower than ORNL's in both cases. A discussion of the ORNL direct construction cost estimates follows.

7.1.1 Direct Construction Cost

Account 211 - Ground Improvements

Ground improvements, which consist essentially of site roads and yard facilities, site access railroads, and barge docking provisions, are estimated at $1,000,000. This cost is used for all 1000-Mw(e) reactor plants for which the reference AEC Middletown site conditions are assumed.

Account 212 - Structures

The reactor buildings, including shielding, were estimated to range in cost from \$0.85 to \$1.35/ft^3, depending on the area considered. Turbine building costs normally range from 60 to 80 cents/ft^3, with a differential of 20 cents between direct and indirect cycles. A cost of 60 cents/ft^3 was used. The office and service building was estimated at \$1.50/ft^3. The intake structure cost includes \$200,000 for waterfront improvements, which is not included in account 211. Radioactive waste and fuel-handling facilities are housed within the reactor building, and thus no separate structures are included for these facilities.

There is no allowance in the cost of structures for a secondary containment vessel around the PCRV, since preliminary safety evaluations indicate no need for it with either design. If such a vessel is required, it has been estimated that an incremental cost of \$2,800,000 for the reference design and \$3,500,000 for the backup design would be added to the total cost of structures.

Account 22 - Reactor Plant Equipment

Reactor equipment (account 211) includes the pressure vessel, control rods, shielding, reactor building ventilation system, reflector, and the reactor plant cranes. The vessel cost represents approximately 70% of this account. For the backup design the ORNL unit cost estimates were

Prestressed concrete structure, \$/yd^3	390
Vessel liner including cooling coils, \$/lb	1.25
Penetration liners and anchors, \$/lb	2.00
Internals, \$/lb	
Steel	1.00
Graphite	1.50
Insulation, \$/ft^2	25

With these unit costs, a vessel cost of \$11,755,000 was obtained. For the reference design the same unit costs were used except that the unit cost for the prestressed structure was reduced to \$350/yd^3 to reflect the elimination of the cross head and circumferential tendons. The cost of wire wrapping, estimated to be \$1,500,000 was then added in. This resulted in a vessel cost of \$9,400,000.

Control rod costs include the cost of the rod and drive assembly and, for the backup design, the variable orifice and drive assembly.

The rod pair and drive assembly was estimated at $21,600 each, the variable orifice and drive at $8,000 each, and the poison injection system at $300,000. Thus, the total cost for the backup design is $2,995,000. The reference design eliminates the variable orifices, which reduces the cost to $2,267,000.

Reflector cost was estimated at a unit cost of $1.50/lb for the graphite blocks for a total of $2,400,000 for each design. The plant ventilation system discharges effluent from a duct at the top of the reactor building and thus eliminates use of a stack, which would normally be included in account 212, at a cost of approximately $75,000.

The heat transfer system (account 222) includes the helium circulators, steam generators, and the coolant receiving, storage, and treatment facilities. Steam-generator costs were estimated at $70/ft^2 of heat transfer surface for the backup design, including the shrouds and PCRV headers. As a result of the evaluation the reference design surface areas were increased by 5%, and the cost was estimated at $83.50/ft^2. The difference in unit cost stems from the difference in module size, operating pressure, and the tube spacing. An auxiliary packaged boiler and its auxiliaries for each design was estimated at $350,000.

Helium receiving, storage, and purification, including the initial charge of helium, were estimated by using the results of previous reactor studies and the design parameters.

Nuclear fuel-handling equipment (account 223) includes two refueling machines, four isolation valves, an auxiliary transfer cask, a reactor viewing device, and 36 fuel storage wells for the backup design. The reference design, with on-line refueling, has only one refueling machine, a reactor viewing device, and 22 storage wells. Both designs require remote tooling to be used in conjunction with the reactor viewing device for reactor maintenance. A breakdown of the ORNL estimate is

	Backup Design	Reference Design
Refueling machine	$1,500,000	$2,500,000
Fuel storage	1,200,000	720,000
Special tools	500,000	500,000
Total	$3,200,000	$3,700,000

A breakdown of the ORNL estimate for instrumentation and controls is

	Backup Design	Reference Design
Reactor controls	$1,500,000	$1,500,000
Secondary and auxiliary systems	1,250,000	1,250,000
Plant systems	250,000	250,000
Total	$3,000,000	$3,000,000

Feedwater supply and treatment equipment (account 227) costs were based on similar cycle costs developed in earlier studies.

Steam, condensate, and feedwater piping (account 228) is unconventional in that redundant lines are required to provide the necessary reliability for circulator motive power and decay-heat removal. Costs were based on conventional cycle costs with appropriate adjustments. The resulting cost breakdown is

	Backup Design	Reference Design
	2400 psi/1000°F/1000°F	3500 psi/1050°F/1050°F
Main steam	$1,820,000	$4,330,000
Reheat	1,560,000	2,210,000
Feedwater and condensate	2,220,000	2,520,000
Miscellaneous	200,000	200,000
Total	$5,800,000	$9,260,000

Other reactor plant equipment (account 229) is made up of the following for both designs:

Nitrogen system	$150,000
Decontamination facilities and miscellaneous	350,000
Circulator removal equipment	330,000
Total	$830,000

Account 23 - Turbine-Generator Units

Turbine-generator costs (account 231) were estimated for a unit using a 5% overpressure allowance to obtain rated output. These basic costs were used with the 7% discount[3] applied rather than 10%, which was in effect prior to May 1, 1967. The turbine costs are based on the

General Electric list price for a single unit rated at 3.5-in.-Hg back pressure and 3% makeup. Adjusting for 1.5-in.-Hg back pressure and 0% makeup results in a reduction of 2.9% in the required rating. Utilizing the 5% overpressure gives a further reduction of 5%. Based on these factors the unit rating was reduced approximately 8% for estimating the cost. This resulted in a rating of 931 Mw. A breakdown of turbine-generator costs is given in Table 7.3.

Table 7.3. Turbine-Generator Cost

	Backup Design	Reference Design
Design		
Type	TC 6F-335 LSB/3600 rpm[a]	TC 6F-30 LSB/3600 rpm[a]
Station net rating, Mw(e)	1001	1000
Station gross rating, Mw(e)	1009	1009
Turbine rating, 0% makeup, 1.5 in. Hg abs, 0% overpressure	959	959
Guaranteed or book turbine rating with 3% makeup and 3.5 in. Hg abs	931	931
Turbine inlet pressure, psia	2400	3500
Turbine inlet temperature, °F	1000	1050
Turbine reheat, °F	1000	1050
Generator rating at 90% power factor	1121	1121
Bypass system rating, % valves wide open	100	100
Base cost turbine rating, Mw(e)/Mva	750/900	650/780
Cost, account 231		
Base cost (June 1967)	$16,400,000	$14,300,000
Turbine cost adder at $4.50/kw	815,000	1,265,000
Generator cost adder at $5/kva	1,105,000	1,705,000
Pressure correction	600,000	0
Inlet temperature correction	0	330,000
Reheat temperature correction	0	330,000
BFP extraction cost adder	128,000	128,000
Total book price	$19,048,000	$18,028,000
Selling price at 93% book price	$17,715,000	$16,766,000
Additional costs		
Bypass system (no piping)	$ 1,227,000	$ 1,097,000
Erection	900,000	890,000
Foundation	350,000	350,000
Account 231 total	$20,192,000	$19,103,000

[a]Read: Tandem-compound six-flow (exhaust) 30-in. last-stage blading, 3600 revolutions per minute.

Circulating water system (account 232) and condenser (account 233) costs were estimated for the following design: 57°F inlet water temperature (26°F ΔT), 100 ft to river intake from turbine building, two-pass single-pressure condenser with 1-in.-OD 22-KWG stainless steel tubes. The designated heat rejection rate is to be used as a cost normalization parameter.

Account 24 - Accessory Electric Equipment

Accessory electric equipment (account 24) costs were based on the costs of a similar plant design with the same auxiliary power requirements. This account includes the auxiliary turbine-generator. As a result of the evaluation the auxiliary turbine-generator was changed from a single unit to two half-size units. This resulted in a cost increase of $120,000.

Account 25 - Miscellaneous Power Plant Equipment

Cost estimates for this account were based on an evaluation of requirements typical of 1000-Mw(e) reactor power stations. A normalized cost of $1,250,000 has been estimated for this account. Items included in this account are compressed air and vacuum cleaning systems, cranes and hoisting equipment not included in the turbine or reactor plant costs, general use service water systems, machine tools, and other miscellaneous power plant equipment defined in Table 105-2 of TID-7025.

Based on a 0.8 load factor and 13.7% per year charges against the total capitalization, the capital contribution to power production cost (exclusive of fuel inventory) is 2.41 mills/kwhr(e) for the backup design and 2.38 mills/kwhr(e) for the reference design.

7.1.2 Capital Cost Uncertainties

In the discussion in Chapters 5 and 6 on the design of systems and components for the HTGR plants, ORNL indicated several areas of uncertainty, particularly in the design of the PCRV, the containment systems, and the steam generators. Changes in these designs could result in increases in the cost estimates presented above. The possible extent of these cost increases is discussed below.

In addition to the cost uncertainties associated with possible design changes there are, of course, uncertainties in the cost estimates for equipment, such as the helium circulators for example, for which there is presently little or no construction experience. The 10% contingency allowance of about $10,000,000 provides for cost uncertainties of this kind. The design uncertainties and the extent of their associated cost increases are discussed below.

7.1.2.1 Prestressed-Concrete Pressure Vessel. The estimate for the PCRV is based on a normal construction schedule for the vessel. Problems with quality control, tolerance requirements, climatic conditions, method of installation, and the fabrication of the liner and penetrations could add time to the construction schedules and increase the cost of labor. Actual construction experience with the Fort St. Vrain PCRV and the PCRV's for the first few 1000-Mw(e) reactors will be required before a more accurate estimate can be made for later plants. However, a cost variation of 20% from the estimated value would be reasonable to expect.

There is a further uncertainty in the cost of the thermal barrier for the walls of the PCRV and core support structure. The estimate of $25/ft^2 for the insulation or thermal barrier is based on the development of a satisfactory, relatively low cost barrier for future HTGR plants. Since the backup design requires about 20,000 ft^2 of thermal barrier and the reference design about 17,000 ft^2, the uncertainty in the cost estimate for this item may be as much as one million dollars.

7.1.2.2 Reactor Building. The reactor building cost is based on the confinement concept wherein the structure is subjected to a differential pressure of only a few inches of water. The acceptance of the confinement concept is discussed in Section 6.1.

Use of a pressure-containing secondary containment structure surrounding the entire PCRV would increase the cost by approximately $2,800,000 for the reference design and $3,500,000 for the backup design. The added cost for secondary containment does not take into consideration possible cost tradeoffs, such as reduced requirements for individual penetration containment, which could reduce the above incremental cost addition.

7.1.2.3 Steam Generators. Section 5.3 points out the uncertainty of the steam generator heat transfer calculations. ORNL and GA calculations were in reasonable agreement for both designs. However, until actual data on heat transfer coefficients for the proposed tube arrangements are obtained, a margin of uncertainty in the design and cost estimate exists. A difference of ±10% from the calculated areas would result in a capital cost uncertainty of about one million dollars for each plant.

7.1.2.4 Additional Equipment Related to Containment. Section 6.2 discusses a number of items of equipment that might be needed to provide assurance against the possibility of overpressure and failure of the PCRV. In the absence of detailed designs and safety analyses for the 1000-Mw(e) HTGR's, ORNL based cost estimates on the equipment specified in the PSAR for Fort St. Vrain. If a more stringent interpretation of containment criteria were to be applied to the 1000-Mw(e) designs, as discussed in Section 6.2, the principal additional costs would be for a pressure-relief valve on the PCRV and for additional valves, piping, and instrumentation in the feedwater and steam systems. ORNL estimated that a pressure-relief valve, together with associated piping and filters might cost a total of $100,000. Additional feedwater and steam valves, piping, and instrumentation would cost a maximum of $300,000 for the backup design and $350,000 for the reference design.

7.2 Operation and Maintenance Costs

Annual operation and maintenance costs for the HTGR, given below, are consistent with the costs estimated for other advanced converter reactors.[4] A permanent staff of 83 people is assumed, with an allowance of 10% for payroll fringe benefits and 14% for home office general and administrative expenses. The repair and maintenance materials and contract services are assumed to be about the same for each of the advanced converter plants. Insurance premiums include commercial third-party liability coverage at $240,000 per year and federal indemnity at $30/Mw(th) per year. Helium losses equivalent to 20% of one inventory of helium per year amount to 4700 lb per year for the backup design and 3120 lb per year for the reference design at $9.60/lb. The breakdown of annual operation and maintenance costs is given in Table 7.4.

Based on a 0.8 load factor, the operation and maintenance contribution to power-production costs is 0.30 mill/kwhr(e) for the backup plant and 0.29 mill/kwhr(e) for the reference plant.

Table 7.4. Operation and Maintenance Costs for a 1000-Mw(e) HTGR

	Annual Operation and Maintenance Cost	
	Backup Design	Reference Design
Total payroll	$ 771,000	$ 771,000
Repair and maintenance materials and contract services	730,000	730,000
Administrative and general	210,000	210,000
Insurance	317,000	310,000
Coolant makeup	45,000	30,000
	$2,073,000	$2,051,000

7.3 Fuel-Preparation and Fabrication Costs

7.3.1 Fuel-Preparation (Conversion) Costs

Fuel-preparation costs for the HTGR were estimated on the same basis as for the HWOCR evaluation.[2] This represents a somewhat closer look at the problem than was made for the advanced converter evaluation.[1] However, it must be realized that cost estimates are being made for processes at production rates measured in tons per day as extrapolated from engineering development experience at rates of only kilograms per day. Thus there is substantial uncertainty in the present estimates, perhaps on the order of ±50%.

Fuel preparation includes those operations necessary to convert makeup and/or recycle material to the proper chemical and physical form needed for fuel fabrication. In this case, sol-gel oxide microspheres are being prepared from virgin thorium as the nitrate, recycled ^{233}U as the nitrate, and makeup ^{235}U as the hexafluoride. Either two or three kinds of microspheres are to be made for the fuel: $^{233}UO_2$–ThO_2 (remote preparation) and $^{235}UO_2$ (hooded preparation), or ThO_2 (hooded preparation),

$^{233}UO_2$–ThO_2 (Th/U ratio = 3, remote preparation), and $^{235}UO_2$ (hooded preparation).

The same cost estimate is presented for both schemes at this point in time, however. The two-particle scheme is "simpler" but not necessarily cheaper, since it requires that all the thorium go through remote preparation.

The fuel-preparation plant is assumed to be an integral part of either the spent-fuel processing plant or the fuel-fabrication plant, or both, in that no site costs are included and sharing of supporting services and service personnel is assumed. This is an economically desirable arrangement, especially since the growth of the gamma-active daughters of ^{232}U into the recycled ^{233}U makes it worthwhile to conduct preparation and fabrication steps promptly after the ^{233}U is purified.

Table 7.5 summarizes the fuel-preparation cost estimates. In addition to the 15,000-Mw(e) equivalent size used as the reference case in the earlier evaluations, plant sizes equivalent to HTGR industries up to 120,000 Mw(e) are presented. These estimates will be used by the Fuel Recycle and Systems Analysis Task Forces by taking into consideration the ultimate number of HTGR's calculated by the linear-programming optimization code, the rate of growth calculated, and the degree of "fragmentation" (number of competing fuel-preparation plants) assumed. Cost numbers are presented for fixed-charge rates on capital investment of 22% per year and 30% per year. The 22% figure was used as a reference value in the earlier evaluations, but some recent estimates were based on a higher value.

Table 7.5 is based on throughput rates calculated from an assumed refueling rate of 10.53 kg (U + Th) per Mw(e) per year. The same cost-vs-throughput-rate relationship can be used for other reactor refueling rates by interpolation on Fig. 7.1. For more precise calculations, the following empirical equations can be used:

$$\text{Capital investment (\$)} = 5.82 \times 10^6 \left(\frac{\text{MT/year}}{260}\right)^{0.458}$$

$$\text{Annual operating cost (\$/year)} = 1.26 \times 10^6 \left(\frac{\text{MT/year}}{260}\right)^{0.357}$$

Table 7.5. Fuel-Preparation (Conversion) Cost Estimates for HTGR

Basis: 22% or 30% fixed-charge rate on capital investment
260 production days per year
Virgin $Th(NO_3)_4$ plus recycle $^{233}UO_2(NO_3)_2$ yields ThO_2 microspheres and $UO_2 \cdot ThO_2$ microspheres
$^{235}UF_6$ makeup yields UO_2 microspheres
0.8 reactor load factor and a throughput rate of 10.53 kg/Mw(e) per year

	Industry Size			
	15,000 Mw(e)	30,000 Mw(e)	60,000 Mw(e)	120,000 Mw(e)
Throughput				
MT/year	158	316	632	1264
MT/day	0.608	1.22	2.43	4.86
Capital investment, $\$10^6$	4.60	6.40	8.70	12.0
Annual operating cost, $\$10^6$	1.05	1.35	1.72	2.20
Total annual cost, $\$10^6$				
At 22% fixed-charge rate	2.06	2.76	3.63	4.84
At 30% fixed-charge rate	2.43	3.27	4.33	5.80
Total unit cost, \$/kg[a]				
At 22% fixed-charge rate	13.05	8.73	5.75	3.83
At 30% fixed-charge rate	15.38	10.35	6.85	4.59
Total unit cost, mill/kwhr				
At 22% fixed-charge rate	0.0196	0.0131	0.0087	0.0058
At 30% fixed-charge rate	0.0231	0.0156	0.0103	0.0069

[a]Unit costs are based on amount of U + Th charged to reactor.

7.3.2 Fuel-Fabrication Costs

The estimation of fuel-fabrication costs for the HTGR is difficult because of the limited data and experience available for this fuel element. The fuel element design is essentially that of the proposed Fort St. Vrain reactor.[5,6] Therefore, a large amount of research, development, and engineering design effort is being directed to establishing the adequacy of the concept. However, the fuel element has not yet been fabricated on any scale nor have the final specifications been established.

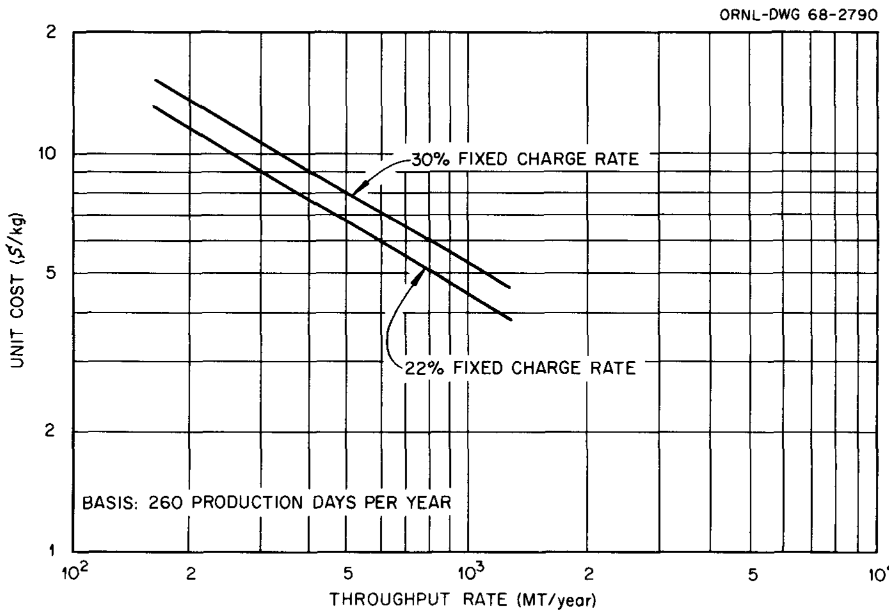

Fig. 7.1. Fuel-Preparation Cost for HTGR.

Any estimate of its cost must therefore be considered as a general, or "ball-park," type.

In this study we have considered two cases: fuel consisting of three types of particles — fissile, fertile, and mixed fissile-fertile; and, as an alternate, two types of fuel particles — fissile and mixed fissile-fertile.

The ORNL cost analysis is based on the fuel element parameters given in Table 7.6. The fabrication flowsheet is shown in Fig. 7.2, which is the process projected for recycle of HTGR fuel in the Thorium-Uranium Recycle Facility (TURF). The starting fuel material is sol-gel oxide microspheres, while output of the fabrication plant is completed fuel elements ready for shipment to the reactor site. In this evaluation only virgin thoria was considered for the fertile particles, with the coating being performed in equipment mounted in vented hoods. The fissile particles, containing only makeup ^{235}U, are processed in the same manner but in separate equipment. The mixed fissile-fertile particles contain virgin thoria and recycle ^{233}U in a 4:1 ratio. It was assumed that activity levels of the recycle uranium would be sufficient to dictate fabrication

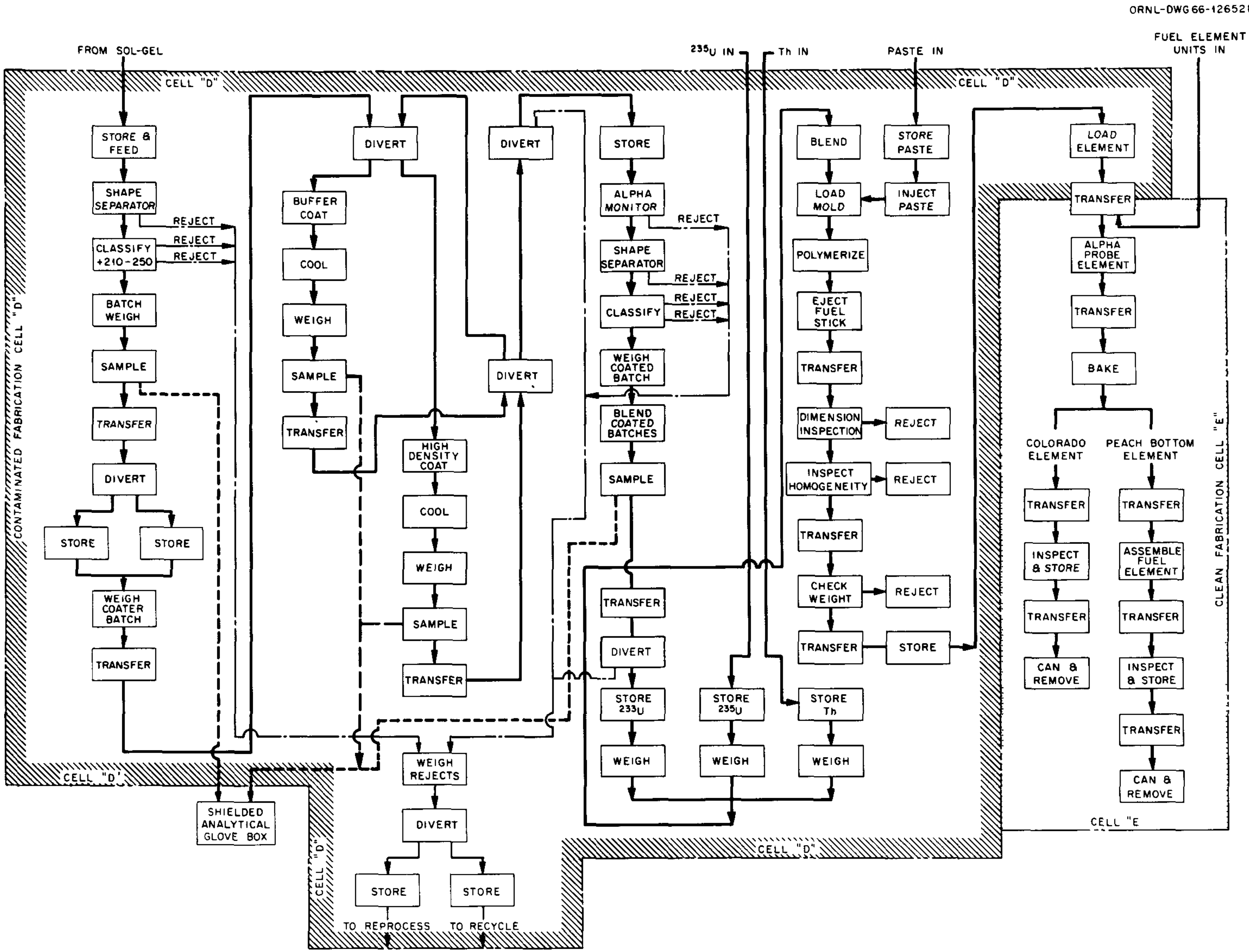

Fig. 7.2. Process Flow Diagram for Recycle HTGR Fuel in TURF.

in a remote facility similar to TURF. The above comments apply specifically to the concept of utilizing three types of particles: fissile, fertile, and fissile-fertile. For the two-particle concept, the fertile

Table 7.6. HTGR Fuel Element Parameters

	Backup Design	Reference Design
Refueling schedule, kg/yr		
Thorium	9757	9770
^{233}U, recycle	243	230
^{235}U, recycle	29	26
^{235}U, makeup	255	185
Total uranium	649	532
Fuel block height, in.	31.2	15.6
Fuel hole diameter, in.	0.455	Same
^{235}U makeup enrichment, % ^{235}U in uranium	93	Same
Fertile particle		
Material	Th + ^{233}U + ^{235}U	Same
Diameter, μ	350	Same
Buffer coating thickness, μ	30	Same
Isotropic coating thickness, μ	100	Same
Fissile particle		
Material	^{235}U	Same
Diameter, μ	150	Same
Buffer coating thickness, μ	90	Same
Isotropic coating thickness, μ	60	Same
Fuel composition	Oxide	Same
Fuel geometry form	Sticks, not bonded to block	Same
Diametral gap	1% of volume	Same
HTGR fuel element drawing	SK-977 (R1801) (8/17/66)	Same
HTGR control rod fuel element drawing	SK-976 (R1801) (8/17/66)	Same
Fuel particle packing fraction	0.6	Same

particle is eliminated and the recycle ^{233}U is dispersed in all the thorium-bearing particles.

The fuel-management scheme for this study was to refuel one-fourth of the core each year, with an annual fabrication requirement of 10,300 kg of heavy metal. If the fabrication plant operates 260 days per year, a daily throughput of 40 kg of heavy metal is necessary to supply one 1000-Mw(e) reactor. For the reference case of a fifteen-reactor economy, an average daily throughput rate of 600 kg of heavy metal is required.

For this study ORNL extrapolated costs from those reported earlier.[7] Costs included in this work are capital and operating expenses of the fabrication plant for coating particles with pyrolytic carbon, forming them into fuel bodies or "sticks," loading and assembling the fuel elements, and purchasing fuel element structural components. Specifically excluded are costs of nuclear fuels, inventory charges, fuel losses, and scrap recovery. These items are included in other parts of the fuel-cycle cost analysis.

The costs calculated for this study are shown in Table 7.7. The costs of fabricating fuel for the two cases under reference conditions and ground rules are \$106 per kg of heavy metal for the three-particle fuel and \$109 per kg for the two-particle fuel.

Thus, it appears slightly cheaper to fabricate with three types of particles than with two types because of the need for remote fabrication of a much larger quantity of material when the recycle uranium is dispersed in all the thorium-bearing particles. However, this difference is quite small and may not be significant.

Two fuel element designs were submitted for this evaluation: the reference design with a fuel element length of 15.6 in. and the backup design of 31.2 in. ORNL concluded that any savings in fabricating a lesser number of fuel elements for the backup design would be offset by additional costs of drilling deeper holes in the hex block of the backup design. ORNL can see no significant difference in fabrication costs for the two designs.

The scale of production has a pronounced effect on fabrication cost, especially at relatively small throughputs. Figure 7.3 is the ORNL projection of fabrication cost for the oxide fuel as a function of throughput in the range of 100 to 1000 kg of heavy metal per day. The dot on the curve is the reference case for this evaluation; that is, a fabrication cost of $106 per kg of heavy metal at a production rate of 600 kg per day.

Table 7.7. Fuel-Fabrication Costs for HTGR Backup Design (Oxide Fuels) at a 600-kg/day Fabrication Rate

	Three-Particle Fuel	Two-Particle Fuel
Loading, %		
Fissile	2.45	2.45
Fissile-fertile	18.90	97.55
Fertile	78.65	
Total	100.00	100.00
Fabrication rate, kg heavy metal/day		
Fissile	14.7	14.7
Fissile-fertile	113.4	585.3
Fertile	471.9	
Total	600.0	600.0
Cost of coating particle, $/kg of heavy metal in particle		
Fissile	130	130
Fissile-fertile	66	36
Fertile	25	
Coating costs, $/kg of heavy metal in fuel element		
Fissile particles	3.19	3.19
Fissile-fertile particles	12.47	35.12
Fertile particles	19.65	
Assembly, $/kg of heavy metal in fuel element	42.00	42.00
Hardware, $/kg of heavy metal in fuel element	29.00	29.00
Total	106.31	109.31

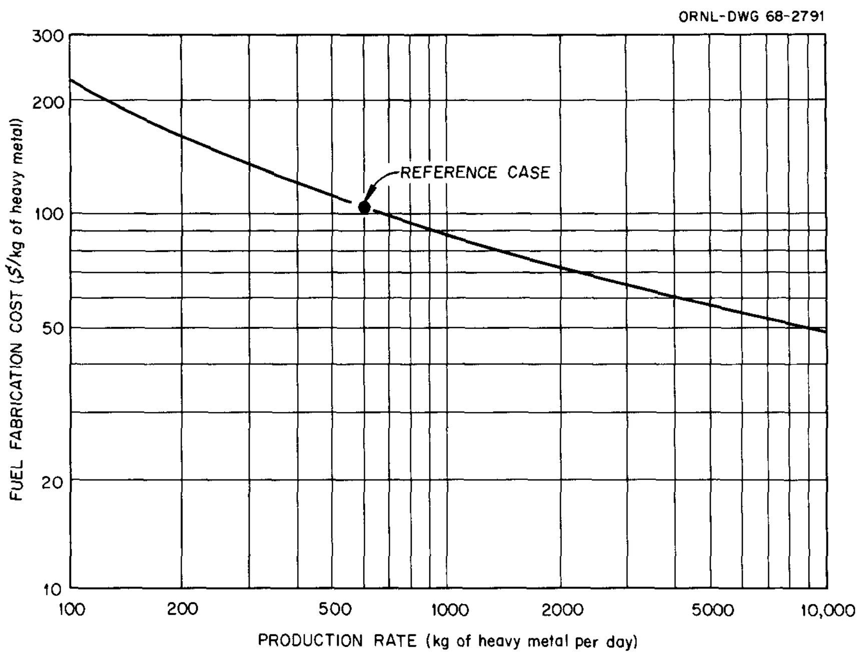

Fig. 7.3. **Fuel-Fabrication Costs as Function of Plant Production Rate for HTGR Fuel Elements with Oxide Fuel and Three Types of Particles.**

7.4 Fuel-Processing Costs

Spent-fuel-processing costs for the HTGR were estimated on the same basis as for the advanced converter evaluation,[1] as modified for comparison with the HWOCR evaluation,[2] except for the following differences:

1. The capital and operating costs were revised somewhat to better reflect differences in fuel type, burnup, and recycle scheme.

2. The "bred-U recycle" scheme called "GA Type II recycle" in Ref. 1 was assumed, with the to-be-recycled bred ^{233}U recovered separately from the to-be-sold $^{235-236}U$. A crush-burn-leach head-end process was assumed, with a "uranium isotope separation" step based on the ^{235}U particles being coated with SiC (or, alternatively, based on a size difference or a difference in the burn-leach behavior of the ^{235}U particle and the ^{233}U-Th particle or particles).

3. The thorium was assumed to be recovered in the nitrate form, but stored for 10 to 15 years, to permit decay of the ^{228}Th and its gamma-active daughters before recycling. The thorium storage charge is itemized separately but is included in the total unit processing cost.

After crushing and burning, the ^{233}U and the thorium and their associated fission products are dissolved together and then separated from each other by solvent extraction. The recovered thorium is stored, and the recovered ^{233}U is recycled immediately to the fuel-preparation plant. The $^{235-236}$U particle is dissolved separately, after grinding (or grinding and reburning if necessary), and the $^{235-236}$U is separated from its associated fission products by solvent extraction. All the fission products are assumed to be disposed of as a high-level waste. The high-level waste charge is itemized separately but included in the total unit processing cost. The high-level waste cost estimate was calculated for perpetual acidic-solution storage in stainless steel tanks by using the TASCO computer code;[8] however, this cost is believed also to be sufficient to cover reduction to solid form and ultimate disposal in a salt-mine storage facility.

Table 7.8 summarizes the fuel-processing cost estimates. In addition to the 15,000-Mw(e)-equivalent size used as the reference case, plant sizes equivalent to HTGR industries up to 120,000 Mw(e) are presented. The unit costs are presented for the 22% per year fixed-charge rate in capital investment used as a reference value and also for 30% per year.

Table 7.8 is based on throughput rates calculated from an assumed refueling rate of 10.53 kg U + Th per Mw(e) per year. The same cost-vs-throughput-rate relationship can be used, as an approximation, for other reactor refueling rates (other burnup or thermal-efficiency values) by interpolation from Fig. 7.4. For more precision, other cases can be calculated from the following empirical cost-scaling equations:

$$\text{Capital investment (\$)} = 39.5 \times 10^6 \left(\frac{\text{MT/year}}{260}\right)^{0.35} ,$$

$$\text{Annual operating cost (\$/year)} = 0.084 \times (\text{capital investment}) + (0.26 \times 10^6)\left(\frac{\text{MT/year}}{260}\right) ,$$

High-level fission-product waste disposal charges (\$/year) =

$$(2.66 \times 10^6)\left[\frac{(\text{MT/year})(\text{Mwd/MT})}{(56{,}000)(365)}\right]^{0.844} ,$$

$$\text{Thorium waste storage charge (\$/year)} = (0.49 \times 10^6)\left(\frac{\text{MT/year}}{260}\right)^{0.844}$$

The capital investment includes site and startup costs and working capital. The capital investment is annualized by multiplying by the fixed-charge rate to cover the costs of return on investment, recovery of investment (depreciation), interim replacements, income taxes, property taxes, and property insurance. All other costs are included in the annual operating cost, except the fission-product and thorium waste charges.

Table 7.8. Spent-Fuel-Reprocessing Cost Estimates for HTGR's of 15,000- to 120,000-Mw(e) Capacity

Basis: Single-purpose, central, aqueous processing plants sized to match the assumed amount and type of reactor indicated; 0.8 reactor load factor; 260 production days per year for processing; throughput rate, 10.53 kg per Mw(e) per year

	Industry Size			
	15,000 Mw(e)	30,000 Mw(e)	60,000 Mw(e)	120,000 Mw(e)
Throughput rate				
MT/year	158	316	632	1264
MT/day	0.61	1.22	2.43	4.86
Capital investment, $\$10^6$	33.1	42.3	53.9	68.6
Annual operating cost, $\$10^6$	2.94	3.87	5.16	7.02
Annual high-level waste charge, $\$10^6$	1.68	2.77	5.02	9.60
Annual thorium storage cost, $\$10^6$	0.29	0.52	0.94	1.68
Total annual cost, $\$10^6$				
At 22% fixed-charge rate	12.19	16.46	22.98	33.39
At 30% fixed-charge rate	14.84	19.84	27.29	38.88
Total unit cost, $/kg				
At 22% fixed-charge rate	77.2	52.1	36.4	26.4
At 30% fixed-charge rate	93.9	62.8	43.2	30.8
Total unit cost, mill/kwhr				
At 22% fixed-charge rate	0.116	0.078	0.055	0.040
At 30% fixed-charge rate	0.141	0.094	0.065	0.046

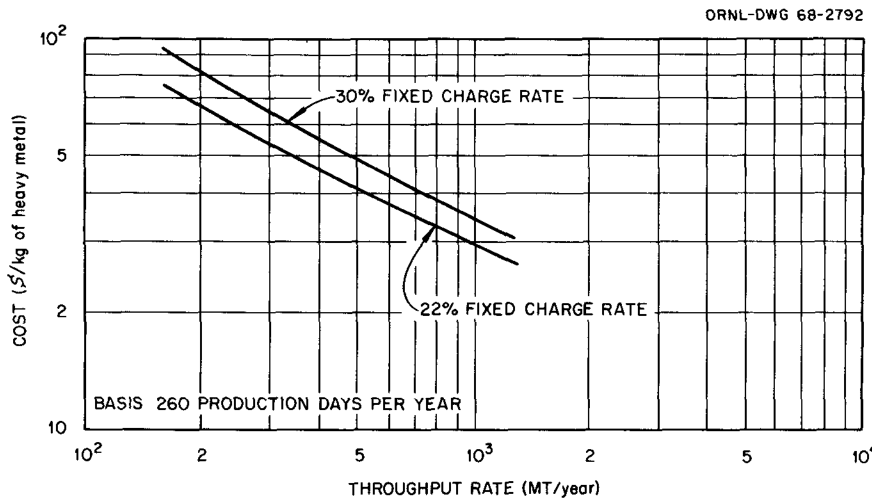

Fig. 7.4. Spent-Fuel-Reprocessing Costs for HTGR.

If the $^{235-236}U$ is not to be recovered, the capital investment can be reduced about 10% and the annual operating reduced proportionately (as per dependence on capital investment in the equation given above). In the reference case [15,000 Mw(e), 22% fixed-charge rate] the incremental cost of recovering the 450 kg ^{235}U (2340 kg total U) per year, which is "retired" from the equilibrium cycle, is about $\$1.01 \times 10^6$ per year; which appears to be economically justifiable.

These reprocessing cost estimates may have an absolute accuracy of about ±30%. (The much-more-detailed DuPont estimates[9] in DP-566 claimed only about ±20%.) It is believed that the relative accuracy for purposes of comparison with other reactors in the current evaluation study is better, perhaps ±10%.

7.5 Fuel-Shipping Costs

Among the component costs of the fuel cycle are the costs of shipping fuel elements to and from the reactor. These costs and the methods used in their estimation are discussed in this section. The fuel shipments considered are of three types:

1. shipment of fresh (unrecycled) fuel elements from the fabrication plant to the reactor,

2. shipment of irradiated (spent) fuel elements from the reactor to the chemical-reprocessing plant (assumed to be at the same site as the fabrication plant),
3. shipment of gamma-active recycled fuel elements from the fabrication plant to the reactor.

The methods of calculation were previously described in the advanced converter evaluation[10] and in other documents.[11,12] The bases and assumptions used are listed below:

1. The fabrication and chemical reprocessing plants are at the same site, which is a distance of 1000 miles from the reactor site.

2. Shipments are by rail. Both sites have railroad sidings and facilities for handling 120-ton casks. Round-trip time is 16 days.

3. Rail freight rates are: full cask, \$0.0193 per lb; empty cask, \$0.0181 per lb.

4. Insurance against damage to cask and contents is at the rate of 0.0005 times the value of the shipment.

5. The spent-fuel shipping cask is used to carry recycled fresh fuel back to the reactor on the return trip.

6. The graphite block fuel assemblies are shipped in the fully assembled condition.

7. Individual canning of fuel assemblies is not required.

8. Handling costs are \$1000 per round trip.

9. Casks are purchased at a cost of \$1.25/lb of cask weight. Fixed charges on casks are 15% per year, including recovery of investment, return on investment, taxes, and ordinary maintenance.

10. It is assumed that it will not be necessary for a courier to accompany the shipment.

11. Shipments are designed to comply with 10 CFR 71 and with ICC Order 70. The 120-ton cask is assumed to have the exclusive use of the vehicle. The maximum dose rate is 10 mr/hr at a distance of 6 ft from the vehicle.[13,14]

Spent fuel elements were assumed to be shipped by rail in steel-shell lead-shielded casks weighing about 120 tons each. A cross section of the cask is shown in Fig. 7.5. The 2-in.-thick outer shell has cooling fins for heat dissipation to the atmosphere, and no mechanical cooling system

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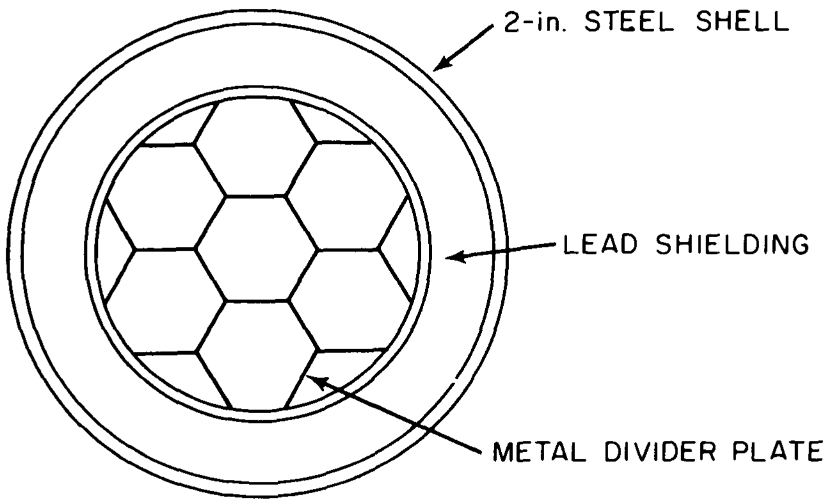


Fig. 7.5. Cross Section of Shipping Cask.

or liquid coolant is used. The cask is air filled rather than water filled. The cask is opened by means of a flanged and bolted cover at one end. Cask dimensions are

	Dimension
Inside diameter of inner shell	48 in.
Outside diameter of outer shell	68 in.
Shield thickness	7.1 in.
Inner cavity length	18 ft

At the full rate of heat dissipation of 30 kw, the fuel element surface temperature is about 500°F and the outer shell temperature is about 200°F for an ambient air temperature of 130°F.

Each shipment consists of one cask carrying 49 elements of the backup design or 98 elements of the reference design. The rate of spent-fuel production from one 1000-Mw(e) reactor can be handled by one cask. On the return trip, the cask is assumed to carry recycled fresh fuel elements from the fabrication plant to the reactor.

Optimal cooling time before shipping spent fuel was calculated to be about 90 days. This included consideration of shielding requirements, inventory charges, and rate of ^{233}U production by decay of ^{233}Pa.

Fresh (unrecycled) fuel elements, which require no shielding, are individually packed in foam-lined boxes to prevent mechanical damage during transit. Shipment by rail was assumed, since it appears to be slightly more economical than shipment by truck.

Fuel-shipping costs at 90 days cooling are listed in Table 7.9. Per kilogram of heavy metal, these costs are several times higher than those of light-water reactors. This is due primarily to the fact that a relatively small quantity of heavy metal is carried in a large volume of graphite; the weight of the cask is governed largely by the volumetric requirements of the inner cavity. The same considerations affect the fresh unrecycled fuel-shipping costs, although to a lesser degree.

Table 7.9. HTGR Fuel-Shipping Costs

	Shipping Cost[a] ($/kg U + Th)	
	Until 1980	After 1980
Fresh (unrecycled) fuel	2.50	2.20
Spent fuel[b]	17	16
Recycled fuel	8	7

[a]Costs given are per kg of U + Th charged to the reactor.

[b]90 days cooling time before shipping.

Because fuel is carried on both legs of the round trip, it was necessary to make an arbitrary assignment of costs between spent fuel and recycled fuel. About two-thirds of the total cost was assigned to the spent fuel and one-third to the recycled fuel. The rationale behind this was that the spent fuel requires more shielding and hence should bear more of the cost. This is admittedly arbitrary, and any other way of splitting costs would work as well. Actually the costs must be considered as a whole; the only reason for separating them is to follow conventional practice.

No distinction was made between shipping costs for the backup and reference designs, since the designs are essentially identical per unit of length. Costs after the year 1980 were assumed to drop slightly as a result of industry growth and higher total rates of shipment.

As mentioned earlier, the high shipping costs primarily result from a relatively small quantity of heavy metal being carried in a large volume of graphite. This suggests the possibility of separating the fuel particles from the blocks prior to shipment. The particles would then be packed in steel containers which, in turn, would be packed in the cask. The graphite blocks would be shipped separately for disposal. This procedure has not been used in the present study, mainly because the state of knowledge of separation procedures was not deemed adequate to predict the costs involved. Also, separate shipment would most likely eliminate the possibility of using the same cask for shipping recycled fresh fuel back to the reactor. This would tend to offset some of the saving in cost. However, this subject still remains open for future studies.

7.6 Fuel-Cycle Costs

Fuel-cycle costs were calculated, as prescribed by the ground rules, on a present-value basis over a 30-year reactor operating history. In addition, ORNL made calculations of the equilibrium cycle on a present-value basis for comparison with other fuel-cycle cost calculations of equilibrium cycles. Since there can be some variation in the way in which such calculations are made, the ORNL calculational procedure is described below in some detail.

7.6.1 Calculation of Average Lifetime Fuel-Cycle Costs

To obtain average 30-year-lifetime costs, the present value (value discounted to reactor startup) of all future costs is determined and divided by the discounted amount of the energy sold during the life of the plant. This levelized cost represents the fixed price that must be received per unit of electrical energy in order to pay for all the costs associated with the fuel cycle. It was assumed that neither the reactor load factor nor the unit costs of purchased materials varied during the lifetime of the reactor.

The advantage of present-value discounting is that it implicitly includes the effect of time displacements between investments and returns. The levelized cost includes applicable interest charges caused by these time displacements.

In making the fuel-cycle cost calculation ORNL first determined the direct cost, which is the contribution that an item would make if interest charges and taxes were zero. The direct-cost contribution to the fuel-cycle cost is obtained by summing all the money invested in an item during the reactor history and dividing by the total energy sold, with no discounting. Thus, for the same total investment, the direct-cost contribution is the same regardless of whether the money is spent at the start of the history, at the end of the history, or in smaller payments spaced during the history.

For the computation of the interest cost it must be considered that some items contribute to the outstanding indebtedness of a utility company. These costs must be financed out of capital funds, and the charge rate applied to them must include taxes. In this study ORNL assumed that the capital charge rate applied to fuel purchase was 13.2% per year; applied to fuel fabrication, 12.8% per year; and applied to coolant purchase, 13.3% per year. Other costs of the fuel cycle may be covered out of current revenues and treated as operating costs rather than as capital investment. Since operating costs are paid before taxes, the applicable rate is simply the net cost of borrowing money. In this study ORNL treated spent-fuel shipping and reprocessing in this manner, with a charge rate of 7.2% per year. This assignment of charges to capital was recently revised to reflect current economic conditions. A detailed discussion of the constituents of these charge rates is given in Ref. 10. For all items ORNL calculated the present-value discount factor with a rate of 6% per year. The discounting was calculated with semiannual compounding. For the calculations ORNL assumed that income from energy generated during a six-month accounting period is received at the end of the period. Other costs and credits were taken at the time they occurred, with discounting to reactor startup.

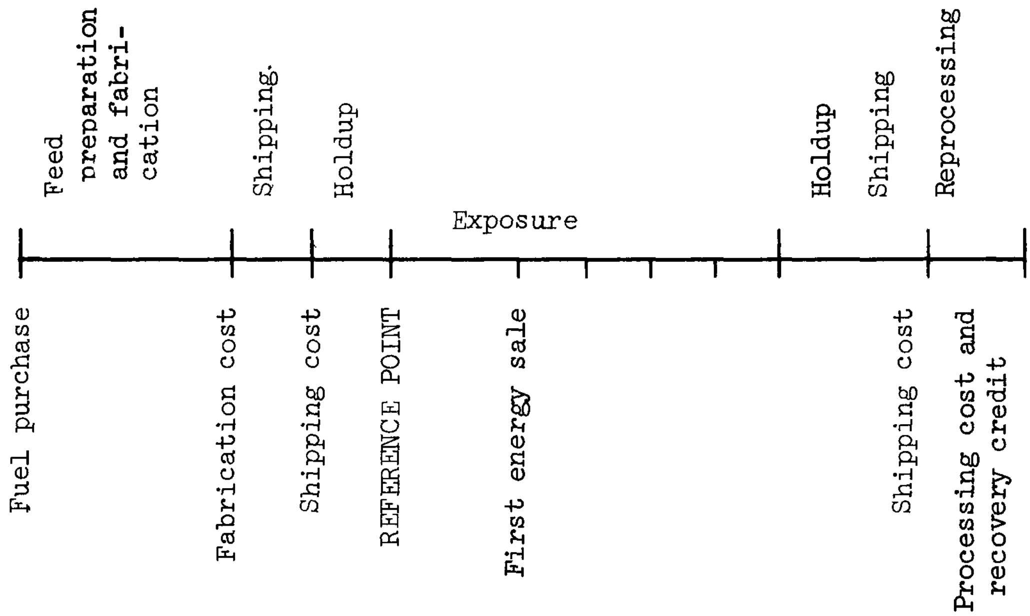
To calculate the interest charges associated with an item of cost ORNL determined the total discounted present value of all direct costs and credits for the item over the reactor lifetime and divided this by the amount of energy delivered, also discounted over the same period. The result is the total cost for the item, including interest charges if the interest rate is equal to the discount rate. The interest charge is the difference between the above total cost and the direct cost multiplied by the ratio of the capital charge rate to the discount rate.

The interest charge on an item may be either positive or negative, depending on whether the investment is made before or after revenue is received. In any case it is convenient to have the results presented in a form in which direct costs and interest costs are separate so that the effect of any changes in direct unit costs or interest rates can easily be determined.

With discounting to a fixed point in time (for example, reactor startup), the interest cost depends strongly on whether the investment occurred at the start or end of the history due to time displacement between expenditures and receipts. It is noteworthy that with 6% discounting and 13.2% charges, credit for fuel or unused fabrication at the end of a 30-year life makes essentially no net contribution and could be neglected. The contribution to direct cost is largely offset by a nearly equal interest charge of opposite sign.

7.6.2 Calculation of Equilibrium Fuel-Cycle Cost

In the calculation of the equilibrium fuel-cycle cost, a present-value discounting method is used that takes account of time displacements between cost and revenues in a typical cycle; that is, a cycle late enough in the reactor lifetime that it is not greatly affected by startup assumptions (although feed and discharge rates for such a cycle are not always at a true equilibrium). This calculation may be viewed as a history of a particle of fuel fed to the reactor and followed through individual steps, including exposure, as shown schematically below. Note that the present-value reference point is at the start of exposure, a lead time and a postexposure holdup are indicated that allow for out-of-core inventory, and revenue from energy sale is credited at the end of each accounting



period. Actually, the last-cycle results reported are based on the feed at the start of the refueling interval and the discharge at the end of this interval, which gives slightly different results than actually following feed material through its entire exposure when the fuel cycle has not reached equilibrium.

7.6.3 Total Fuel-Cycle Costs

Fuel-cycle costs are given in Table 7.10 for the reference design. For comparison this table also gives the fuel-cycle costs obtained from GA mass balances and ORNL unit costs. Table 7.11 shows the effects of perturbing the fuel processing and fabrication plant size and the effect of varying uranium ore cost. Fuel-cycle costs for the backup design are given in Table 7.12. Fabrication costs throughout were based on a carbon-to-thorium ratio of 200. The slightly different fuel element required for a ratio of 210 might increase the fuel-cycle costs by about 0.01 mill/kwhr(e).

7.7 Total Power Costs

The power costs are calculated, as specified in the ground rules, by using a fixed charge on depreciating capital of 13.7% per year for

Table 7.10. Fuel-Cycle Cost for the HTGR Reference Design

	From GA Mass Balances	From ORNL Mass Balances		
Reference exposure, Mwd/MT	62,600	62,600	66,000	43,000
Fuel plant throughput, MT/year	162	162	154	236
Carbon-to-thorium atom ratio	200	200	210	210
Unit costs, $/kg				
Fabrication	115	115	117	101
Processing	76	76	78	61
Shipping	23	23	23	23
Last cycle costs, mills/kwhr(e)				
Burnup	0.262	0.267	0.281	0.234
Fabrication	0.176	0.176	0.171	0.219
Processing	0.109	0.109	0.106	0.126
Shipping	0.034	0.034	0.032	0.048
Inventory	0.410	0.474	0.450	0.424
Fabrication interest	0.057	0.057	0.055	0.051
Processing interest	−0.017	−0.017	−0.016	−0.018
Total	1.031	1.100	1.079	1.084
30-year average costs, mills/kwhr(e)				
Burnup	0.280	0.289	0.301	0.255
Fabrication	0.180	0.180	0.175	0.222
Processing	0.111	0.111	0.108	0.128
Shipping	0.035	0.035	0.033	0.050
Inventory	0.337	0.366	0.345	0.395
Fabrication interest	0.078	0.078	0.076	0.068
Processing interest	−0.014	−0.014	−0.014	−0.012
Total	1.007	1.045	1.024	1.106

Table 7.11. Perturbed Fuel-Cycle Cost[a] for the HTGR Reference Design

HTGR fuel exposure, Mwd/MT	66,000	43,000
Variation with static fuel plant size		
Fuel Plant Capacity [Mw(e) of HTGR's]	**Fuel-Cycle Cost [mills/kwhr(e)]**	
7,500	1.161	1.267
15,500 (reference)	1.024	1.106
22,500	0.967	1.041
30,000	0.934	1.007
60,000	0.873	0.941

Variation with ore cost

U_3O_8 Ore Cost ($/lb)	^{235}U Value[b] ($/g)	Fuel-Cycle Cost [mills/kwhr(e)]	
4	9.143	0.927	1.012
8 (reference)	11.174	1.024	1.106
12	13.204	1.162	1.248
20	17.266	1.396	1.484
30	22.342	1.690	1.780
50	32.495	2.277	2.370

[a]Unperturbed basis same as Table 7.10.

[b]$26/kg for separative work, $2.70 U_3O_8 to UF_6 conversion, 93%.

Table 7.12. Fuel-Cycle Cost for the HTGR Backup Design

Reference exposure, Mwd/MT	50,100	66,500	82,600[a]
Fuel plant throughput, MT/year	203	153	123
Carbon-to-thorium atom ratio	200	200	200
Unit costs, $/kg			
Fabrication	106	118	128
Processing	67	78	88
Shipping	23	23	23
Last cycle costs, mills/kwhr(e)			
Burnup	0.342	0.382	0.431
Fabrication	0.205	0.172	0.150
Processing	0.129	0.113	0.103
Shipping	0.044	0.033	0.027
Inventory	0.515	0.550	0.610
Fabrication interest	0.054	0.055	0.060
Processing interest	−0.016	−0.017	−0.019
Total	1.273	1.288	1.362
30-year average costs, mills/kwhr(e)			
Burnup	0.356	0.400	0.452
Fabrication	0.217	0.188	0.164
Processing	0.129	0.115	0.103
Shipping	0.045	0.035	0.028
Inventory	0.435	0.462	0.496
Fabrication interest	0.072	0.073	0.081
Processing interest	−0.012	−0.015	−0.015
Total	1.242	1.258	1.309

[a]Early cycle had short exposures, so 30-year costs are underestimated for this case.

investor-owned financing. These charges to capital are added to the operation and maintenance and the fuel-cycle costs to give the total annual cost. Annual power production is assumed to be 0.8 times the full-power design capacity. The costs calculated under investor-owned financing are given in Table 7.13.

Table 7.13. Power-Production Cost for Investor-Owned Financing

	Cost [mills/kwhr(e)]	
	Backup Design	Reference Design
Capital	2.41	2.38
Operation and maintenance	0.30	0.29
Fuel cycle	1.26	1.02
Total	3.97	3.69

Complete secondary containment, if it were required, could add as much as 0.07 mill/kwhr(e) to the backup design or 0.05 mill/kwhr(e) to the reference design.

If the GA estimates of capital costs and the GA mass balances are used, together with ORNL estimates of fuel-fabrication, processing, shipping, and operation and maintenance cost, the costs indicated below are obtained for the investor-owned reference design.

	Cost [mills/kwhr(e)]
Capital	2.19
Operation and maintenance	0.29
Fuel cycle	1.01
Total	3.49

References

1. M. W. Rosenthal et al., A Comparative Evaluation of Advanced Converters, USAEC Report ORNL-3686, Oak Ridge National Laboratory, January 1965.

2. P. R. Kasten et al., An Evaluation of Heavy-Water-Moderated Organic-Cooled Reactors, USAEC Report ORNL-3921, Oak Ridge National Laboratory, January 1967.

3. Electrical World, May 15, 1967.

4. NUS Corporation, Guide for Economic Evaluation of Nuclear Reactor Plant Designs, draft, November 1967.

5. GA drawings No. 90-SK976(R1801) and No. 90-SK977(R1801), Aug. 17, 1967.

6. GA drawings No. 90-SK976(R1801) and No. 90-SK977(R1801), June 10, 1966.

7. A. L. Lotts, D. A. Douglas, and R. L. Pilloton, Refabrication Technology and Costs for High-Temperature Gas-Cooled Reactor Fuels, USAEC Report ORNL-TM-1115, Oak Ridge National Laboratory, May 1965.

8. J. O. Blomeke et al., Cost of Permanent Disposal of Power-Reactor Fuel-Processing Waste in Tanks, USAEC Report ORNL-2873, Oak Ridge National Laboratory, September 1965.

9. W. H. Farrow, Jr., Radiochemical Separations Plant Study, Part II. Design and Cost Estimates, USAEC Report DP-566, Savannah River Plant, March 1961.

10. Oak Ridge National Laboratory, An Evaluation of Advanced Converter Reactors, USAEC Report WASH-1087, February 1968.

11. R. Salmon, A Computer Code for Calculating the Cost of Shipping Spent Reactor Fuels, USAEC Report ORNL-3648, Oak Ridge National Laboratory, August 1964.

12. R. Salmon, Estimation of Fuel Shipping Costs for Nuclear Power Cost-Evaluation Purposes, USAEC Report ORNL-3943, Oak Ridge National Laboratory, March 1966.

13. Code of Federal Regulations, Title 10, Part 71, as published in Federal Register, Vol. 31, No. 141, July 22, 1966.

14. Interstate Commerce Commission Order No. 70, November 18, 1965, amended April 14, 1966, as reported in Federal Register, Vol. 31, No. 83, April 29, 1966.

8. RESEARCH AND DEVELOPMENT REQUIREMENTS

Research and development requirements for the backup and reference designs were discussed by GGA in letters prepared for this evaluation.[1-3] The ORNL working group reviewed the requirements and made changes to provide uniformity with the evaluation of other reactor concepts. These requirements were restricted to a limited scope of research and development for which the need could be clearly projected. The research and development costs projected by GGA and ORNL for this scope of work specifically for the backup and reference design plants total $92,000,000 and do not include work on Peach Bottom, the present and future research and development required for Fort St. Vrain, and work on UHTREX. The AEC base program effort was not included unless specifically identified as being applicable to the backup and reference design plants.

It has been shown that a high rate of plant installation by utilities is necessary for the success of a major reactor system and that a broad backup of industrial capability must also be developed to achieve this rate. Thus the development of the HTGR concept into a competitive position with the currently accepted light-water reactor plants will require development of industrial capability for supplying components, fuel fabrication, reprocessing and recycle services, and development of the safety related technology applicable to a variety of plant sizes and locations.

Total costs for developing all phases of this concept, including the required industrial capability cannot be accurately predicted. However, comparing the experience, status, and cost for the development of water reactor technology to the current status of the development of HTGR technology and considering the need for industrial capability, the Division of Reactor Development and Technology estimates that future HTGR development costs would probably range in the hundreds of millions of dollars.

The initial GGA schedule requires completion of the first 1000-Mw(e) backup plant in 1973 and the first reference plant in 1975. Since the submission of this proposed schedule, the required startup date of the Fort St. Vrain reactor has slipped into 1973. Corresponding experience with other nuclear, as well as conventional, plants incorporating developmental and scaleup requirements indicates that a considerably longer time is required for design, development, and construction for plants than that planned in the initial GGA schedule for HTGR. Based on prior experience, therefore, the Division of Reactor Development and Technology has concluded that more time will be required to achieve large-scale commercial status for HTGR systems than indicated in the initial schedule projections.

References

1. Letter from A. J. Goodjohn, General Atomic, to C. J. Raseman, Brookhaven National Laboratory, Dec. 28, 1966.

2. Letter from A. J. Goodjohn, General Atomic, to C. J. Raseman, Brookhaven National Laboratory, Feb. 8, 1967.

3. Letter from A. J. Goodjohn, General Atomic, to R. S. Carlsmith, Oak Ridge National Laboratory, May 25, 1967.

9. COMMENTS OF REVIEWERS

Pertinent reviewer's comments not reflected in the body of this report are abstracted below:

1. The Gulf General Atomic reports upon which the evaluation is based should be made public documents because the engineering information contained in the evaluation barely merits a conceptual classification.

2. The relative status of the backup and reference designs needs clarification. Which concept is GGA promoting for their 1000-Mw(e) HTGR's? Is it either concept? In any case, we agree with the AEC that the commercial availability dates of 1976 and 1978 for the backup and reference designs are optimistic.

3. It probably will be evident to the reader that a position of optimism was taken in the evaluation. The evaluation of the fuel system is an example of this. Important aspects were not discussed, or were treated in only a preliminary manner. Optimistic design margins for fuel performance were estimated, although demonstration of performance at the proposed burnups has not been experimentally confirmed. While there is reason to think that burnable poisons may be used in the backup design, the effects of neither burnable poisons nor power-shaping poisons are included in the estimates for neutron performance. Similarly, the potential influence of safety on economics is not made clear in the summary.

4. Evolutionary design improvements incorporated in the HTGR design during the last 2 1/2 years have not been included. Subsequent plant layouts for the large HTGR have specifically considered the ease of removability of steam generators. This led to the multicavity design which, unfortunately, was not available early enough for evaluation in this study.

5. Under fuel loading allowance, the coolant temperature rise should be reduced by at least a factor of 5 for both the reference and backup designs. Under flow shunting both the coolant temperature rise and the film temperature difference should be reduced by at least a factor of 2 for the reference design and by about 20% for the backup design. These corrections reduce the maximum in the reference design to about 2450°F.

6. These paragraphs [Section 4.2.1] were apparently inserted prior to the completion of the P-18 capsule. In this capsule, BISO particles

operated to a fast fluence in excess of 8×10^{21} nvt at temperatures up to 1300°C and burnups up to 20%. <u>All</u> particles that were within manufacturing specifications survived. As P-18 contained 15 samples each made up of a thousand or more particles, there is little question that BISO particles can meet the requirements set forth in this design.

7. The GGA schedule calls for startup of Ft. St. Vrain in either late 1971 or early 1972. The schedule has not slipped to 1973. The current schedule for starting up a 1000-Mw(e) reactor of the reference design type has slipped from 1976 to 1978.

8. [Concerning research and development costs, Chapter 8] The development of industrial capability for supplying components for fuel services is undertaken and included in the price structure of the materials produced. Only the reactor development program utilized by GGA and ORNL is applicable. The Task Force ground rules, as originally stipulated by RDT, excluded the industrial facility cost.

9. Many of the questions that can be raised about the fuel and core design are under extensive study for the Fort St. Vrain reactor. There is, however, one particular problem which should be discussed in more detail because it has broad implications on the requirements of other systems. The steam-graphite reaction is mentioned only in the summary. The specified steam leak rate of 0.04 lb/hr will certainly cause no concern about graphite removal. To detect this minute a leak would require more complex moisture-detection equipment than is implied in Section 6.2.3.3. Since there is no means of eliminating moisture between the steam generators and the core region, it will be necessary to isolate and repair leaking portions of the steam generator as mentioned in Section 5.3.2.3. It is not readily apparent that the stringent leak rate requirements will not have a severe effect on plant availability. This is not assessed in the report. If the low leak rate, or even a slightly higher one, can be maintained, there will still be a buildup of impurities in the gas systems. Even small amounts of carbon deposited in instrument lines, etc. could have an effect on plant availability.

10. The possibility of eliminating the requirement for emergency core coolant mentioned in Section 6.2 is not consistent with current design practice. This assumes that gross fuel failure can be tolerated in an

accident. In current reactor designs this is not considered to be an economically realistic approach. Apparently graphite temperatures remain low enough without emergency cooling to maintain structural integrity. More must be known, however, about the change in graphite stresses to estimate the structural capability of the graphite in going from normal operating to emergency shutdown conditions.

11. It would be helpful in Section 5.2.2 if the economic incentive for considering on-line refueling were pointed out. This is going to be a very complex operation and costly to develop; thus, the incentive should be strongly emphasized.

12. The value of the report would be enhanced if informed comment on Pu or low-enrichment fueling could be added.

APPENDICES

Appendix A

GROUND RULES

At the first meeting of the Advanced Converter Task Force on August 22, 1966, a set of economic ground rules was adopted for the studies. In the period since that date there have been a number of changes in prices and in financing conditions. In order to reflect current conditions as accurately as possible and at the same time provide cost data consistent with those being used by the Systems Analysis Task Force, ORNL used the following revised fixed charge rates:

Depreciating capital	13.7%
Coolant purchase	13.3%
Fuel purchase	13.2%
Fuel fabrication	12.8%
Fuel shipping and reprocessing	7.2%
Separative work	$26/kg
Reference ^{233}U price	$13.05/g

The changes in fixed charge rates reflect the higher cost of money which is currently in effect, the decrease in separative work cost is in accord with the current (1969) AEC price schedule. The revised U^{233} reference price was specified so that the ratio of U^{233} to U^{235} prices would realistically reflect the value of these two isotopes in an HTGR.

Capital costs were estimated as of June 1967, without escalation. For the remainder of the cost bases ORNL used the ground rules as originally stated and as quoted on the following pages.

It should be noted that these ground rules are not the same as the ones adopted by the Systems Analysis Task Force (WASH 1100) or the ones adopted by the Light-Water Reactor Task Force (WASH 1082).

REACTOR EVALUATION STUDIES — GROUND RULES AND COST BASES

1. Source of Data

The designers of each reactor concept are asked to provide a detailed description of a 1000-Mw(e) reactor plant, which in their opinion will lead to the lowest power cost. Effect of variations in size from 400 Mw(e) to 4000 Mw(e) should also be considered.

2. General Provisions

The time period under consideration in the studies will be from 1970 to 2020. For each reactor design an estimate should be made as to the year in which the first 1000-Mw(e) reactor will be started up. The reference design may assume successful completion of current development programs. However, in the cases in which such assumptions are made, a design should also be specified which is based entirely on current technology. For the purposes of capital cost calculations and system analysis it will be assumed that the plant lifetime is 30 years. Fuel cycles will be calculated on basis of 30-year present worth of costs (levelized costs).

3. Power Cost Components

A. Fuel-Cycle Cost

The fuel-cycle cost will be resolved into the following components: (1) burnup cost, if any, (2) credit for fissile material sold, (3) fabrication cost, including fuel preparation cost, (4) processing cost, if any, including ultimate waste disposal, (5) shipping cost, (6) fixed charges on fissile and fertile inventories, and (7) interest charges on operating capital invested in fabrication, processing, inventories of special nuclear materials, and shipping.

B. Reactor Plant Capital Cost

Capital costs will be estimated for each reactor plant. In estimating the capital costs it is to be assumed that the equipment and system have been fully developed and that the plant is one of a number of the same type to be built. However, discount credit for quantity orders of equipment is not assumed.

Capital cost breakdowns are to be arranged in accordance with the system of accounts given in the AEC Handbook "Guide to Nuclear Power Cost Evaluation," TID-7025 (Vol. I). Indirect costs appropriate to the reactor size will be estimated based on the breakdown used in TID-7025 (Vol. I). However, new estimates are to be made of percentages applied for each indirect cost item to reflect recent experience. Fixed charges on the reactor plant are taken as constant over the plant life at 12% per year.

C. Reactor Plant Operating Cost

Operating and maintenance costs will be estimated for each reactor plant.

D. Development Cost

Research and development costs for the reactor concept will be estimated by years, for the period from the present through the year in which the first commercial 1000-Mw(e) plant is scheduled to start up. R & D will include the net cost of construction and operation of any reactor experiments or reactor prototypes, i.e., the total cost less any anticipated revenues from power produced by these experiments. Revenues should be based on sale of power at costs for 1000-Mw(e) plant. Interest on fuel and D_2O should be assumed at 5% for "noncommercial" plants.

4. Financing Conventions

Private ownership of fuel and of fabrication and reprocessing plants is assumed. The reference values of fixed charges, interest rates, and material prices are indicated in the following paragraphs.

Ownership of fissile and fertile materials during fabrication and processing, as well as when on the reactor site, is considered to be vested in the reactor plant. Inventory charges on fissile and fertile inventories are to be computed using a reference value of 10% per year.

Interest charges on the fabrication cost of fuel elements are computed in the same way as the fixed charges on fuel. For this purpose the fuel elements are assumed to depreciate linearly with time over the period of irradiation.

The reference discount factor for computing present worth in systems analysis is 6%. The discount period and the period of interest payments are assumed to be semiannual.

Heavy water is to be treated as a nondepreciating asset and inventory cost computed using 10% per year charges on its value.

5. Value of Materials

A. The values of the following are to be considered as fixed for the purpose of these studies:

1. The value of unirradiated enriched uranium is based on a separative work cost of \$30/kg.
2. Conversion of U_3O_8 to UF_6: \$2.70 per kg of uranium.
3. D_2O: \$17.50/lb, until new plants have to be built.

B. The values of the following will vary with time, depending on such factors as power demand, availability of resources, types of power plants built, etc. Reference values are given below for current or near-term use, and an anticipated range of values for the longer term.

1. Natural uranium as U_3O_8: reference value of \$8.00/lb U_3O_8: anticipated range \$5.00 to \$50.00/lb.
2. Unirradiated thorium as ThO_2: reference value of \$5.00/lb ThO_2; anticipated range of \$5.00 to \$30.00/lb.
3. Depleted uranium of low enrichment: value corresponding to its enrichment based on total uranium present, with no additional penalty for ^{236}U content.
4. Highly enriched uranium containing ^{233}U: the value of the fuel mixture is computed from its isotopic composition by assigning the ^{235}U the same value per gram it has in 90% enriched uranium (reference value \$12/gram), assigning the contained ^{233}U the same value as the ^{235}U.
5. Plutonium: reference value from 0.8 to 1.5 times the value of enriched ^{235}U.

In order that the potential of the reactors be adequately evaluated, it will be necessary to have the optimum fuel cycle characteristics for each over the range of the above material values.

6. Reactor Plant

The electric station is to have a net capability of 1000 Mw(e). More than one reactor per station is permissible if indicated by the economics. The condenser pressure is assumed to be 1 1/2-in. Hg abs. The plant factor will be assumed to be 90% of the reactor availability.

Fuel is considered to be received at the reactor site 45 days before loading. The cooling time before shipment of irradiated fuel will be established for each concept.

The loss rate of heavy water is to be estimated by the reactor designer.

7. Fuel Fabrication Plant

The fabrication plant is considered to be located at the same site as the chemical processing plant. There is common use of utility facilities such as electrical power substations, water supplies, steam heating systems and natural gas lines, access roads, and waste treatment and disposal facilities. Since the cost of providing such site preparation can vary by large amounts depending upon the actual site selected, no allowance is made for these costs in our calculations.

The plant is assumed to be designed for fabricating a single type of fuel element and to be capable of serving a nuclear industry of specified capacity of the reactor type being studied. However, the computerized technique does have the capability of considering fabrication of more than one type of fuel element in the same plant when the designs are similar. It is suggested that below one metric ton per day throughput, the fabrication plant be considered as dual or multipurpose in those cases where a similar type fuel element is in commercial existence or is included in the particular study being considered. A "turn around" penalty shall be added when fuel elements of different design, enrichments or fuel rod geometries are involved.

The fixed charge rate on depreciating capital will be taken as 22% per year.

Losses during fabrication are assumed to be 0.2% per cycle.

The fabrication plant capacity will be assumed to be 125% of the average throughput to compensate for fluctuations in amounts of fuel being handled.

The fuel element design is to be specified in detail by the proponents.

The cost of fuel element fabrication is assumed to remain constant for a given production rate, with respect to time; i.e., no escalation allowances are provided.

Hold-up time of fuel material within the fabrication plant will be established for each concept.

8. Reprocessing Plant

Amounts and types of spent reactor fuel produced will be estimated as a function of time for a specified power growth rate. From these processing "demand" curves an estimate will be made of the economically optimal schedule of processing plant sizes and types required. Capital and operating costs, including waste disposal costs, for these plants will be estimated and appropriate present-worth-average unit processing costs will be calculated. Both single- and multi-purpose plants will be considered in the optimization. Stockpiling of fuel, i.e., delaying processing to permit building a larger plant at a later date, will be permitted if the overall cost of inventory plus processing is reduced. The use of NFS, and the NFS price formula, will be assumed to apply for the first 1.0 MT/day of processing load. Costs will be expressed in 1966 dollars, with no allowance for escalation.

A value of 22%/yr will be used for the fixed-charge rate on capital investment, including depreciation, cost of money, taxes, etc.

Operating costs will include labor, materials, and all other expenses not included in capital charges, except as noted below.

Fuel inventory in processing will be estimated, and considered in optimization studies, but inventory charges (interest) will not be included in the reported processing cost (i.e., will be reported separately). Losses of fissile and fertile materials will be assumed to be 1%; but the value of the material lost will not be included in the reported processing cost. Fuel materials preparation costs (conversion costs) will be estimated and reported separately, though in some cases it may be logical to carry out these chemical conversion steps in the processing plant, or at the same site (i.e., processing charges are based on nitrate-solution products).

Processing plants will be assumed to have an on-stream capability of 85% (310 days/year).

Economic plant life will be assumed to be 15 years, when estimating the potential load curve for new plants as old ones are retired.

9. Fuel Preparation

Fuel preparation is defined to include the preparation of ceramic grade oxide powder, arc-fused oxide fragments, sol-gel oxide fragments, sol-gel oxide fragments containing carbon, or thorium metal powder or sponge, as appropriate. Enriched uranium contained as UF_6 or plutonium obtained as nitrate are assumed to be converted to the proper form for inclusion in the fuel as a part of the fuel preparation steps. The facilities required for fuel preparation are assumed to be shared with those for reprocessing or fabrication as suitable, and costs are estimated using the financing conventions applied to the other plants. However, the conventional practice of including the fuel preparation cost with fuel fabrication cost will be followed in tabulating the fuel cycle cost components.

10. Shipping

It is assumed that fabrication and reprocessing are performed at the same site, which initially is located 1000 miles from the reactor. An estimate of average shipping distances for later years when more reprocessing plants are available will be made. Casks are assumed to cost $1.00 per lb of cask weight. The maximum cask weight is taken as 110 tonnes fully loaded. The cask utilization factor is estimated for each concept. The cask handling fee is $500 per round trip, and the cask life is 30 years.

The freight rates assumed are the following: loaded cask, \$0.0193 per lb; empty cask, \$0.0181 per lb. Insurance against property loss is charged each shipment at 0.05% of the value of the fabricated fuel elements (including the fuel) and the cask. The cost of liability insurance is included in the charges against the reactor, processing, and fabrication plants.

Appendix B

MASS BALANCES FOR BACKUP DESIGN

The mass balance for a 30-year core history for the backup design with ^{235}U makeup is given in Table B.1. The cycle has discrete refueling of one-fourth of the core at a time. Reference time is given at 100% load factor.

Table B.1. Mass Balance for Backup Design

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
Time,[b] years	0	1.6	2.4	3.2	4.0	4.8	5.6	6.4	7.2	8.0	8.8	9.6	10.4	11.2	12.0
Fresh makeup feed, kg															
^{232}Th	40,716	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180
^{235}U	2,297	593	456	355	270	284	315	291	294	287	287	281	278	277	275
^{238}U	149	50	30	23	17	18	20	19	19	19	19	18	18	18	18
Recycle feed,[c] kg															
^{233}U			193	234	256	267	256	276	279	281	283	283	287	288	289
^{234}U			17.8	29.8	41.3	51.7	41.3	62.1	70.6	77.4	83.0	77.3	88.2	92.3	95.7
^{235}U			2.1	4.8	8.3	12.0	8.2	15.9	19.5	22.8	25.7	22.9	28.7	31.0	33.0
^{236}U			0.1	0.4	1.0	1.8	1.0	3.0	4.5	6.1	7.9	6.1	9.8	11.9	13.9
Discharge, kg															
^{232}Th		9,866	9,714	9,565	9,418	9,565	9,560	9,556	9,552	9,551	9,552	9,556	9,558	9,560	9,561
^{233}U		195	236	259	270	259	279	282	284	286	286	290	291	292	293
^{234}U		18.2	32.6	43.8	54.2	44.5	64.4	72.5	79.2	84.8	79.2	90.1	94.3	97.7	101
^{235}U		173	145	104	77.2	130	85.7	80.0	61.8	66.4	68.6	71.9	75.6	77.2	79.6
^{236}U		42.7	81.2	91.9	100	115	70.4	56.8	46.1	49.9	52.5	52.8	55.2	56.2	58.3
^{238}U		25.4	31.0	31.5	31.8	39.6	23.2	18.1	13.7	14.4	16.0	14.8	15.0	14.6	14.6
^{239}Pu		0.69	0.90	0.93	0.93	1.13	0.65	0.50	0.38	0.41	0.46	0.43	0.43	0.42	0.43
^{240}Pu		0.30	0.39	0.41	0.43	0.53	0.31	0.24	0.18	0.19	0.21	0.20	0.20	0.20	0.20
^{241}Pu		0.27	0.43	0.48	0.50	0.60	0.35	0.27	0.20	0.21	0.24	0.22	0.23	0.22	0.22
^{242}Pu		0.10	0.26	0.42	0.56	0.53	0.32	0.25	0.19	0.20	0.22	0.20	0.20	0.20	0.19

[a]For fueling events 16 through final loading, see Table B.1 (continued).

[b]At full-power operation.

[c]99% of fertile particle discharge delayed 0.8 full-power years.

Table B.1 (continued)

Fueling event	16	17	18	19	20	21	22	23	24	25	26	27	28	29	Final loading
Time,[a] years	12.8	13.6	14.4	15.2	16.0	16.8	17.6	18.4	19.2	20.0	20.8	21.6	22.4	23.2	
Fresh makeup feed, kg															
^{232}Th	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	10,180	
^{235}U	281	275	275	275	273	279	274	275	275	274	279	276	276	277	
^{238}U	18	18	18	18	18	18	18	18	18	18	18	18	18	18	
Recycle feed,[b] kg															
^{233}U	290	291	292	293	294	294	295	296	296	297	297	297	298	298	
^{234}U	98.4	95.7	101	103	105	106	105	108	109	110	110	110	111	112	
^{235}U	32.7	33.1	36.4	37.7	38.8	39.7	38.9	40.7	41.4	42.0	42.5	42.1	43.0	43.4	
^{236}U	15.9	13.8	18.0	20.1	22.1	23.9	21.9	25.9	27.7	29.4	31.0	29.3	32.7	34.2	
Discharge, kg															
^{232}Th	9,562	9,564	9,565	9,566	9,568	9,569	9,570	9,570	9,571	9,572	9,573	9,573	9,574	9,575	39,195
^{233}U	294	925	296	297	297	298	299	299	300	300	300	301	301	302	1,216[c]
^{234}U	97.7	103	105	107	108	107	110	111	112	112	112	113	114	115	457
^{235}U	77.4	80.8	82.4	83.7	86.0	84.4	86.7	87.7	88.5	90.1	89.1	90.5	91.1	91.5	592
^{236}U	55.3	59.1	61.0	62.7	65.5	66.5	66.5	68.3	69.8	72.2	69.8	73.3	74.8	76.1	274
^{238}U	14.3	14.2	14.1	14.0	14.4	14.0	14.0	14.0	14.0	14.0	14.0	14.1	14.1	14.0	62
^{239}Pu	0.42	0.42	0.42	0.42	0.43	0.42	0.42	0.42	0.42	0.43	0.43	0.43	0.43	0.43	1.8
^{240}Pu	0.19	0.19	0.19	0.19	0.20	0.19	0.19	0.19	0.19	0.20	0.19	0.19	0.19	0.19	0.72
^{241}Pu	0.22	0.22	0.22	0.22	0.22	0.22	0.22	0.22	0.22	0.23	0.22	0.22	0.22	0.22	0.69
^{242}Pu	0.19	0.19	0.19	0.18	0.19	0.18	0.18	0.18	0.18	0.18	0.18	0.18	0.18	0.18	0.36

[a] At full-power operation.

[b] 99% of fertile particle discharge delayed 0.8 full-power years.

[c] Includes ^{233}Pa.

Appendix C

MASS BALANCES FOR REFERENCE DESIGN

Mass balances are given for the reference design in Table C.1 for a carbon-to-thorium ratio of 210. This table is for a 1.2-year first core life at full power and then continuous fueling approximated with discrete makeup for periods of 0.1333 years at full power. To reduce the amount of data presented, representative results obtained for a span in time are given. Results obtained by GGA are given in Table C.2 for comparison. Additional reference design mass balances are given in Tables C.3 through C.7. For ^{233}U feed or makeup, a uranium mixture was selected which might be representative of that from HTGR fertile-particle discharge.

A slight discrepancy is noted in the mass-balance tables. Small amounts of ^{234}U and ^{236}U isotopes were considered to be present in the enriched ^{235}U feed. Discharge data include these contributions but they were neglected in the feed data.

Also, these histories were actually calculated out to only about 15 years. Thus the estimates of final loadings are not exactly representative for a 30-year reactor lifetime with regard to relative amounts of the uranium nuclides.

Table C.1. Mass Balance for HTGR Reference Design

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12
Time,[b] years	0	1.2	1.333	1.467	1.600	1.733	1.867	2.000	2.133	2.267	2.400	2.533
Fresh makeup feed, kg												
^{232}Th	38,780	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{235}U	1,764	117	100	101	101	101	101	59.0	56.9	54.7	52.7	50.8
^{238}U	114	7.6	6.5	6.5	6.5	6.6	6.6	3.8	3.7	3.5	3.4	3.3
Recycle feed,[c] kg												
^{233}U								26.0	27.6	29.0	30.3	31.4
^{234}U								2.1	2.4	2.8	3.1	3.5
^{235}U								0.20	0.25	0.32	0.38	0.45
^{236}U								0.01	0.014	0.019	0.025	0.032
Discharge, kg												
^{232}Th		1,576	1,572	1,568	1,563	1,559	1,555	1,550	1,546	1,542	1,538	1,533
^{233}U		26.2	27.9	29.3	30.6	31.7	32.8	33.7	34.5	35.2	35.8	36.4
^{234}U		2.44	2.78	3.12	3.46	3.81	4.15	4.55	4.89	5.22	5.55	5.87
^{235}U		24.5	22.2	20.1	18.3	16.7	15.3	17.4	16.0	14.7	13.5	12.4
^{236}U		6.3	6.68	7.00	7.29	7.53	7.73	9.88	10.1	10.2	10.4	10.5
^{238}U		3.4	3.35	3.32	3.28	3.25	3.22	3.98	3.94	3.91	3.87	3.83
^{239}Pu		0.076	0.075	0.074	0.074	0.074	0.092	0.092	0.092	0.091	0.091	0.090
^{240}Pu		0.034	0.036	0.036	0.037	0.037	0.037	0.046	0.046	0.046	0.046	0.045
^{241}Pu		0.026	0.028	0.030	0.032	0.033	0.034	0.043	0.044	0.044	0.045	0.045
^{242}Pu		0.008	0.010	0.013	0.015	0.018	0.020	0.028	0.031	0.034	0.036	0.039
^{237}Np		0.31	0.36	0.41	0.46	0.51	0.56	0.76	0.82	0.88	0.93	0.99
^{238}Pu		0.100	0.083	0.104	0.127	0.151	0.177	0.256	0.29	0.33	0.37	0.41
^{228}Th[d]		2.6×10^{-5}	3.3×10^{-5}	4.1×10^{-5}	5.0×10^{-5}	6.0×10^{-5}	7.1×10^{-5}	8.2×10^{-5}	9.3×10^{-5}	1.1×10^{-4}	1.2×10^{-4}	1.3×10^{-4}

[a] For fueling events 13 through final loading, see Table C.1 (continued).

[b] At full-power operation, time shown to the start of a period.

[c] 99% of fertile particle discharge delayed 0.8 full-power years.

[d] With thorium recycle.

Table C.2. Mass Balance Obtained by GA for the Reference Design

Time[a]	Mass (kg)							
	Thorium In	^{233}U Recycled	^{235}U Recycled	Total U Recycled	^{235}U Makeup	Thorium Out	^{235}U Retired	Total U Retired
0	41,300				1,690			
1.2–2.0	10,300				550	10,000	116	180
2.0–2.8	10,300	170	2	187	280	9,900	69	144
2.8–3.6	10,300	200	4	236	230	9,700	42	127
3.6–4.4	10,300	220	7	263	170	9,600	29	131
4.4–5.2	10,300	220	10	286	180	9,700	50	159
5.2–6.0	10,300	220	8	269	190	9,700	25	81
6.0–6.8	10,300	230	13	302	180	9,700	20	65
6.8–7.6	10,300	230	16	318	180	9,700	14	48
7.6–8.4	10,300	230	19	329	180	9,700	15	51
8.4–9.2	10,300	230	21	339	180	9,700	16	54

[a]Years at full power.

Table C.3. Mass Balance for HTGR Reference Design, Low Exposure

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
Time,[b] years	0	0.800	0.889	0.978	1.067	1.155	1.244	1.333	1.422	1.511	1.600	1.689	1.778	1.867	1.956
Fresh makeup feed, kg															
^{232}Th	38,780	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{235}U	1,586	94.3	76.5	77.8	78.5	78.9	79.1	80.8	80.4	80.0	46.9	44.6	42.3	40.1	37.9
^{238}U	103	6.1	5.0	5.0	5.1	5.1	5.1	5.2	5.2	5.2	3.0	2.9	2.7	2.6	2.5
Recycle feed,[c] kg															
^{233}U											20.4	21.9	23.3	24.6	25.8
^{234}U											1.25	1.47	1.69	1.91	2.14
^{235}U											0.087	0.11	0.139	0.170	0.204
^{236}U											0.003	0.004	0.006	0.008	0.010
Discharge, kg															
^{232}Th		1,588	1,585	1,582	1,579	1,576	1,573	1,570	1,568	1,565	1,562	1,559	1,556	1,553	1,550
^{233}U		20.6	22.1	23.5	24.8	26.0	27.1	28.2	29.1	30.0	30.8	31.5	32.2	32.8	33.4
^{234}U		1.58	1.78	2.00	2.22	2.44	2.67	2.97	3.21	3.43	3.66	3.90	4.13	4.40	4.63
^{235}U		27.7	25.9	24.1	22.6	21.1	19.7	23.1	21.6	20.2	18.9	17.8	16.7	17.5	16.5
^{236}U		4.65	4.97	5.27	5.53	5.77	5.99	7.74	8.15	8.17	8.37	8.54	8.69	9.88	10.1
^{238}U		3.13	3.11	3.09	3.07	3.05	3.03	3.76	3.74	3.71	3.69	3.66	3.64	4.04	4.02
^{239}Pu		0.064	0.064	0.064	0.065	0.065	0.065	0.081	0.081	0.081	0.081	0.081	0.081	0.089	0.089
^{240}Pu		0.025	0.027	0.028	0.029	0.030	0.031	0.039	0.040	0.040	0.040	0.041	0.041	0.046	0.046
^{241}Pu		0.014	0.016	0.019	0.020	0.022	0.024	0.031	0.033	0.034	0.035	0.036	0.036	0.041	0.042
^{242}Pu		0.003	0.004	0.005	0.006	0.007	0.009	0.013	0.014	0.016	0.018	0.020	0.022	0.026	0.028

[a]For fueling events 16 through final loading, see Table C.3 (continued).

[b]At full-power operation, time shown to the start of a period.

[c]99% of fertile particle discharge delayed 0.8 full-power years.

Table C.3 (continued)

Fueling event	16	17	18	19	20	21	22	23	24	25	26–30[a]	31–35	36–40	41–45	46–50
Time,[b] years	2.044	2.133	2.400	2.489	2.578	2.667	2.756	2.844	2.933	3.022	3.467	3.556	4.000	4.444	4.889
Fresh makeup feed, kg															
^{232}Th	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{235}U	35.9	33.9	32.2	30.3	28.1	26.5	24.9	23.4	21.9	20.5	25.2	22.8	27.1	26.9	22.5
^{238}U	2.3	2.2	2.1	2.0	1.8	1.7	1.6	1.5	1.4	1.3	1.6	1.5	1.8	1.7	1.5
Recycle feed,[c] kg															
^{233}U	26.9	27.9	28.8	29.7	30.5	31.2	31.9	32.5	33.1	33.6	34.9	36.3	33.9	36.9	37.6
^{234}U	2.37	2.61	2.84	3.08	3.31	3.55	3.79	4.03	4.27	4.50	5.21	6.36	4.81	6.89	7.78
^{235}U	0.241	0.280	0.322	0.366	0.413	0.461	0.512	0.565	0.619	0.675	0.853	1.17	0.739	1.30	1.58
^{236}U	0.013	0.016	0.020	0.024	0.029	0.034	0.040	0.047	0.054	0.062	0.090	0.152	0.073	0.186	0.262
Discharge, kg															
^{232}Th	1,547	1,544	1,541	1,538	1,535	1,532	1,529	1,526	1,523	1,520	1,543	1,543	1,542	1,541	1,541
^{233}U	33.9	34.4	34.8	35.2	35.6	35.9	36.2	36.4	36.6	36.8	34.3	34.2	37.9	38.5	39.0
^{234}U	4.86	5.09	5.32	5.55	5.81	6.04	6.26	6.48	6.70	6.92	5.19	5.26	8.02	8.98	9.84
^{235}U	15.5	14.5	13.7	12.9	12.9	12.2	11.5	10.8	10.2	9.68	15.6	15.5	8.45	6.93	5.90
^{236}U	10.2	10.3	10.4	11.3	11.3	11.4	11.5	11.5	11.6	11.6	11.3	11.5	5.70	4.41	3.44
^{238}U	4.00	3.96	3.94	3.91	4.17	4.14	4.11	4.08	4.05	4.03	4.34	4.42	2.10	1.56	1.14
^{239}Pu	0.088	0.088	0.087	0.086	0.091	0.090	0.090	0.089	0.088	0.087	0.092	0.092	0.043	0.032	0.024
^{240}Pu	0.046	0.046	0.046	0.045	0.049	0.048	0.048	0.048	0.048	0.047	0.050	0.051	0.024	0.018	0.013
^{241}Pu	0.042	0.042	0.043	0.043	0.046	0.048	0.045	0.045	0.045	0.045	0.045	0.045	0.021	0.016	0.011
^{242}Pu	0.030	0.032	0.034	0.036	0.041	0.043	0.045	0.047	0.049	0.050	0.036	0.037	0.017	0.013	0.009

[a]For each of five fuelings hereafter.

[b]At full-power operation, time shown to the start of a period.

[c]99% of fertile particle discharge delayed 0.8 full-power years.

Table C.3 (continued)

Fueling event	51–55	56–60	61–65	66–70	71–75	76–80	81–85	86–90	91–95	96–100	Thereafter	Final loading
Time,[a] years	5.333	5.778	6.222	6.667	7.111	7.556	8.000	8.444	8.889	9.333	9.778	
Fresh makeup feed, kg												
^{232}Th	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	
^{235}U	21.5	21.1	21.2	20.9	20.7	20.4	20.1	19.9	20.3	20.5	21.0	
^{238}U	1.4	1.4	1.4	1.4	1.3	1.3	1.3	1.3	1.3	1.3	1.3	
Recycle feed,[b] kg												
^{233}U	38.2	38.6	38.9	38.7	38.8	39.4	39.6	39.7	39.8	39.9	40.0	
^{234}U	8.77	9.65	10.4	9.82	9.82	11.3	11.9	12.5	13.0	13.4	16.0	
^{235}U	1.92	2.23	2.53	2.31	2.31	2.89	3.16	3.40	3.62	3.81	5.2	
^{236}U	0.370	0.494	0.631	0.522	0.523	0.816	0.986	1.16	1.34	1.53	3.0	
Discharge, kg												
^{232}Th	1,541	1,541	1,541	1,541	1,542	1,542	1,542	1,542	1,542	1,543	1,543	37,860
^{233}U	39.3	39.1	39.2	39.8	40.0	40.1	40.2	40.3	40.4	40.5	40.4	974
^{234}U	10.6	10.0	10.0	11.5	12.1	12.7	13.2	13.6	13.1	13.1	16.2	400
^{235}U	6.81	7.13	7.13	7.07	7.22	7.41	7.67	7.89	7.61	7.62	9.0	350
^{236}U	4.10	5.44	4.39	4.09	4.11	4.21	4.38	4.55	4.27	4.24	6.0	120
^{238}U	1.34	1.51	1.49	1.27	1.21	1.18	1.18	1.18	1.16	1.15	1.15	29.1
^{239}Pu	0.028	0.032	0.031	0.027	0.026	0.025	0.025	0.025	0.025	0.025	0.025	0.54
^{240}Pu	0.015	0.017	0.017	0.015	0.014	0.014	0.014	0.014	0.013	0.013	0.013	0.23
^{241}Pu	0.014	0.015	0.015	0.013	0.012	0.012	0.012	0.012	0.012	0.012	0.012	0.16
^{242}Pu	0.011	0.012	0.012	0.010	0.010	0.009	0.009	0.009	0.009	0.009	0.009	0.074

[a]At full-power operation, time shown to the start of a period.

[b]99% of fertile particle discharge delayed 0.8 full-power years.

Table C.4. Mass Balance for HTGR Reference Design with ^{233}U in Initial Loading and Makeup

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16–18[b]	19–21	22–24
Time,[c] years	0	1.200	1.333	1.466	1.600	1.733	1.866	2.000	2.133	2.266	2.400	2.533	2.666	2.800	2.933	3.066	3.466	3.866
Fresh makeup feed, kg																		
^{232}Th	38,860	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619
^{233}U	1,296	80.8	70.0	71.2	72.2	73.0	73.8	37.8	37.1	36.3	35.6	34.9	34.2	26.6	26.1	25.0	21.6	17.3
^{234}U	471	29.4	25.4	25.9	26.2	26.5	26.8	13.7	13.5	13.2	12.9	12.7	12.4	9.67	9.49	9.09	7.85	6.29
^{235}U	271	16.9	14.6	14.9	15.1	15.2	15.4	7.89	7.72	7.57	7.43	7.28	7.13	5.55	5.44	5.22	4.51	3.61
^{236}U	204	12.7	11.0	11.2	11.3	11.5	11.6	5.94	5.83	5.70	5.59	5.48	5.37	4.18	4.10	3.93	3.39	2.72
Recycle feed,[d] kg																		
^{233}U								34.8	35.6	36.3	37.0	37.6	38.2	43.4	43.5	43.5	44.4	46.3
^{234}U								9.02	9.22	9.41	9.61	9.79	9.98	16.4	16.4	16.4	17.8	22.0
^{235}U								3.64	3.59	3.56	3.54	3.53	3.53	6.52	6.47	6.32	7.2	8.95
^{236}U								3.62	3.65	3.67	3.69	3.71	3.73	7.45	7.48	7.52	8.6	11.3
Discharge, kg																		
^{232}Th		1,581	1,576	1,572	1,568	1,563	1,559	1,555	1,551	1,547	1,543	1,539	1,535	1,535	1,527	1,519	1,507	1,496
^{233}U		35.1	35.9	36.7	37.4	38.0	38.5	43.8	43.9	43.9	44.0	44.0	44.0	46.8	46.6	46.1	47.1	46.2
^{234}U		9.11	9.31	9.51	9.70	9.89	10.1	16.5	16.6	16.6	16.6	16.7	16.7	15.9	22.1	21.9	25.2	25.5
^{235}U		3.67	3.63	3.60	3.58	3.57	3.57	6.59	6.54	6.50	6.46	6.43	6.40	9.04	8.97	8.83	9.9	10.6
^{236}U		3.66	3.87	3.71	3.73	3.75	3.77	7.53	7.55	7.58	7.60	7.62	7.64	11.4	11.5	11.5	14.0	15.3

[a] For fueling events 25 through final loading, see Table C.4 (continued).

[b] For each of three fuelings.

[c] At full-power operation, time shown to the start of a period.

[d] 99% of fertile particle discharge delayed 0.8 full-power years.

Table C.4 (continued)

Fueling event	25–27	28–30	31–35[a]	36–40	41–45	46–50	51–55	56–60	61–65	66–70	71–75	76–80	81–85	86–90	Thereafter	Final loading
Time,[b] years	4.266	4.666	5.066	5.733	6.400	7.066	7.733	8.400	9.066	9.733	10.400	11.066	11.733	12.400	13.066	
Fresh makeup feed, kg																
^{232}Th	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	1,619	
^{233}U	17.2	24.7	24.4	25.5	23.6	24.0	20.9	23.6	25.1	25.5	23.8	24.3	21.0	23.4	23.0	
^{234}U	6.25	8.98	8.87	9.27	8.58	8.73	7.60	8.58	9.13	9.27	8.65	8.84	7.64	8.51	8.36	
^{235}U	3.59	3.15	5.09	5.32	4.92	5.01	4.36	4.92	5.24	5.32	4.96	5.07	4.38	4.88	4.80	
^{236}U	2.70	3.88	3.83	4.01	3.71	3.77	3.28	3.71	3.94	4.01	3.74	3.82	3.30	3.68	3.61	
Recycle feed,[c] kg																
^{233}U	45.8	46.0	47.9	47.3	47.4	47.2	46.8	48.2	48.5	48.7	48.7	49.0	48.8	49.6	50.0	
^{234}U	22.9	25.5	25.5	24.0	21.8	23.1	23.7	27.7	27.7	27.2	25.7	26.4	26.0	28.7	28.5	
^{235}U	9.30	10.6	10.8	10.0	8.52	9.21	9.63	11.8	11.8	11.6	10.7	11.2	11.0	12.6	14.0	
^{236}U	12.7	15.2	14.1	12.9	10.2	12.1	14.2	19.0	18.2	17.2	14.5	16.2	17.3	22.0	25.0	
Discharge, kg																
^{232}Th	1,520	1,521	1,521	1,521	1,521	1,522	1,522	1,523	1,523	1,523	1,524	1,524	1,524	1,525	1,525	37,671
^{233}U	47.0	47.6	47.8	47.7	47.7	47.3	48.7	49.0	49.0	49.3	49.7	49.4	50.2	50.5	50.5	1,330
^{234}U	25.0	23.8	22.2	23.6	23.1	23.5	27.8	27.0	25.9	26.7	26.7	26.0	28.9	28.9	29.2	761
^{235}U	10.3	9.88	8.71	9.46	9.19	9.54	11.9	11.4	10.8	11.3	11.1	11.1	12.7	12.7	14.1	370
^{236}U	14.0	12.7	10.6	12.5	12.1	14.1	19.2	16.8	14.8	16.5	16.3	17.2	22.1	20.5	25.3	560

[a] For each of five fuelings hereafter.

[b] At full-power operation, time shown to the start of a period.

[c] 99% of fertile particle discharge delayed 0.8 full-power years.

Table C.5. Mass Balance for HTGR Reference Design with ^{233}U in Initial Loading

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Time,[b] years	0	1.200	1.333	1.467	1.600	1.733	1.867	2.000	2.133	2.267	2.400	2.533	2.667	2.800	2.933	3.066	3.200	3.333	3.467
Fresh makeup feed, kg																			
^{232}Th	38,780	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{233}U	1,291																		
^{234}U	469																		
^{235}U	269	122	104	105	105	105	105	49.7	48.7	47.6	46.6	45.6	44.7	33.3	32.9	32.4	31.9	31.4	30.9
^{236}U	203																		
^{238}U		7.9	6.7	6.8	6.8	6.8	6.8	3.2	3.2	3.1	3.0	3.0	2.9	2.2	2.1	2.1	2.1	2.0	2.0
Recycle feed,[c] kg																			
^{233}U								34.6	35.4	36.2	36.8	37.4	38.0	43.2	43.2	43.2	43.2	43.2	43.2
^{234}U								8.98	9.18	9.37	9.57	9.76	9.94	16.3	16.4	16.4	16.4	16.5	16.5
^{235}U								3.62	3.57	3.54	3.52	3.52	3.52	6.50	6.45	6.41	6.37	6.34	6.31
^{236}U								3.61	3.63	3.66	3.68	3.69	3.71	7.42	7.44	7.47	7.49	7.51	7.53
Discharge, kg																			
^{232}Th		1,577	1,573	1,569	1,565	1,560	1,556	1,552	1,548	1,544	1,540	1,536	1,532	1,528	1,524	1,520	1,516	1,512	1,508
^{233}U		35.0	35.8	36.5	37.2	37.8	38.4	43.6	43.6	43.7	43.7	43.7	43.7	46.3	46.1	45.8	45.6	45.3	45.1
^{234}U		9.08	9.27	9.47	9.66	9.85	10.0	16.5	16.5	16.6	16.6	16.6	16.7	22.1	22.0	21.9	21.8	21.7	21.6
^{235}U		3.65	3.61	3.58	3.56	3.55	3.55	6.56	6.52	6.47	6.44	6.40	6.38	9.00	8.91	8.83	8.76	8.69	8.62
^{236}U		3.65	3.67	3.69	3.71	3.73	3.75	7.49	7.52	7.54	7.57	7.59	7.61	11.4	11.4	11.4	11.4	11.5	11.5
^{238}U																			
^{239}Pu																			
^{240}Pu																			
^{241}Pu																			
^{242}Pu																			

[a]For fueling events 20 through final loading, see Table C.5 (continued).

[b]At full-power operation, time shown to the start of a period.

[c]99% of fertile particle discharge delayed 0.8 full-power years.

Table C.5 (continued)

Fueling event	20	21	22	23	24	25	26–30[a]	31–35	36–40	41–45	46–50	51–55	56–60	61–65	Thereafter	Final loading
Time,[b] years	3.600	3.733	3.866	4.000	4.133	4.267	4.400	5.067	5.733	6.400	7.067	7.733	8.400	9.067	9.733	
Fresh makeup feed, kg																
^{232}Th	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	
^{233}U																
^{234}U																
^{235}U	21.3	21.0	20.6	20.1	19.7	19.2	39.0	38.1	34.3	33.1	32.2	26.6	44.0	34.9	32.0	
^{236}U																
^{238}U	1.4	1.4	1.3	1.3	1.3	1.2	2.5	2.5	2.2	2.1	2.1	1.7	2.9	2.3	2.2	
Recycle feed,[c] kg																
^{233}U	45.9	45.6	45.3	45.1	44.9	44.6	45.2	39.0	42.2	42.6	43.0	43.0	42.9	42.2	42.8	
^{234}U	21.9	21.8	21.7	21.6	21.5	21.3	25.4	6.71	13.0	13.6	17.0	19.5	21.3	21.3	17.0	
^{235}U	8.91	8.83	8.75	8.68	8.61	8.53	10.5	1.31	4.12	4.36	6.04	7.32	8.25	3.47	5.20	
^{236}U	11.3	11.3	11.3	11.3	11.4	11.4	15.1	0.17	4.05	4.19	7.87	11.4	14.8	1.02	3.50	
Discharge, kg																
^{232}Th	1,504	1,500	1,496	1,495	1,488	1,484	1,517	1,517	1,516	1,516	1,516	1,516	1,515	1,515	1,514	37,510
^{233}U	46.3	46.0	45.6	45.3	44.9	44.6	39.3	42.8	43.5	43.4	43.4	43.3	42.6	43.1	43.2	1,060
^{234}U	26.1	25.8	25.6	25.4	25.2	24.9	7.14	13.4	17.2	17.3	19.7	21.5	12.5	15.7	17.2	430
^{235}U	10.9	10.7	10.6	10.5	10.4	10.2	14.8	10.7	10.6	10.2	10.1	13.4	8.34	9.39	10.0	450
^{236}U	15.3	15.3	15.3	15.3	15.3	15.3	15.6	11.3	12.8	12.5	14.4	20.7	6.68	9.73	8.2	256
^{238}U							5.37	2.51	1.72	1.60	1.02	2.04	1.97	1.71	1.70	46
^{239}Pu							0.14	0.064	0.044	0.040	0.026	0.051	0.049	0.044	0.041	0.96
^{240}Pu							0.068	0.032	0.022	0.020	0.013	0.026	0.025	0.021	0.021	0.42
^{241}Pu							0.071	0.033	0.023	0.021	0.013	0.026	0.025	0.022	0.021	0.35
^{242}Pu							0.071	0.033	0.022	0.020	0.013	0.027	0.026	0.022	0.022	0.22

[a] For each of five fuelings hereafter.

[b] At full-power operation, time shown to the start of a period.

[c] 99% of fertile particle discharge delayed 0.8 full-power years.

Table C.6. Mass Balance for HTGR Reference Design with ^{233}U in Makeup

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12	13	14
Time,[b] years	0	1.200	1.333	1.467	1.600	1.733	1.867	2.000	2.133	2.267	2.400	2.533	2.667	2.800
Fresh makeup feed, kg														
^{232}Th	38,780	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{233}U		29.2	68.7	69.7	70.2	70.5	70.5	43.3	41.7	40.1	38.6	37.1	34.4	33.2
^{234}U		10.6	25.0	25.4	25.5	25.6	25.6	15.7	15.1	14.6	14.0	13.5	12.5	12.1
^{235}U	1,764	6.10	14.3	14.6	14.7	14.7	14.7	9.05	8.71	8.38	8.06	7.75	7.19	6.94
^{236}U		4.60	10.8	11.0	11.0	11.1	11.1	6.82	6.57	6.32	6.08	5.84	5.42	
^{238}U	114													
Recycle feed,[c] kg														
^{233}U								26.0	27.6	29.0	30.3	31.4	33.3	34.1
^{234}U								2.10	2.45	2.80	3.16	3.51	4.21	4.55
^{235}U								0.200	0.255	0.316	0.381	0.453	0.607	0.690
^{236}U								0.010	0.014	0.019	0.026	0.033	0.052	0.063
Discharge, kg														
^{232}Th		1,576	1,572	1,568	1,563	1,559	1,554	1,550	1,546	1,541	1,537	1,533	1,529	1,525
^{233}U		26.2	27.9	29.3	30.6	31.7	32.8	33.7	34.5	35.2	35.9	36.4	36.9	37.4
^{234}U		2.55	2.88	3.21	3.57	3.91	4.25	4.59	4.93	5.26	5.59	5.90	6.23	6.54
^{235}U		35.8	29.7	26.9	24.4	22.3	20.3	18.5	16.9	15.5	14.2	13.1	12.0	11.2
^{236}U		8.47	8.98	9.40	9.79	10.1	10.4	10.6	10.8	11.0	11.1	11.3	11.4	11.5
^{238}U		4.55	4.50	4.46	4.41	4.37	4.32	4.28	4.24	4.20	4.16	4.11	4.07	4.03
^{239}Pu		0.102	0.100	0.100	0.099	0.098	0.098	0.098	0.097	0.097	0.096	0.096	0.095	0.095
^{240}Pu		0.046	0.048	0.049	0.050	0.050	0.050	0.050	0.050	0.050	0.050	0.050	0.049	0.049
^{241}Pu		0.034	0.038	0.040	0.042	0.044	0.045	0.046	0.047	0.047	0.047	0.047	0.047	0.047
^{242}Pu		0.011	0.014	0.017	0.020	0.024	0.027	0.030	0.033	0.036	0.039	0.042	0.045	0.048

[a] For fueling events 15 through final loading, see Table C.6 (continued).

[b] At full-power operation.

[c] 99% of fertile particle discharge delayed 0.8 full-power years.

Table C.6 (continued)

Fueling event	15	16	17	18	19	20	21	22	23	24	25–30[a]	31–35	36–40	41–45
Time,[b] years	2.933	3.066	3.200	3.333	3.467	3.600	3.733	3.867	4.000	4.133	4.267	5.067	5.733	6.400
Fresh makeup feed, kg														
^{232}Th	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{233}U	31.9	31.7	29.6	28.5	27.5	26.4	25.4	30.5	27.0	24.2	21.2	16.1	23.9	25.0
^{234}U	11.6	11.5	10.7	10.3	9.99	9.59	9.22	11.0	9.80	8.79	7.66	5.85	8.68	9.08
^{235}U	6.66	6.62	6.18	5.95	5.74	5.52	5.31	6.37	5.64	5.06	4.43	3.36	4.99	5.22
^{236}U	5.23	5.03	4.99	4.66	4.49	4.33	4.16	4.00	4.80	4.25	3.81	3.34	3.77	3.94
^{238}U														
Recycle feed,[c] kg														
^{233}U	34.9	35.5	36.1	36.6	37.0	37.4	37.8	38.1	38.3	38.5	39.1	43.3	43.8	43.9
^{234}U	4.89	5.23	5.56	5.88	6.20	6.51	6.82	7.11	7.40	7.69	8.49	20.8	19.2	18.7
^{235}U	0.776	0.864	0.955	1.05	1.14	1.24	1.33	1.43	1.53	1.62	1.92	7.93	6.84	6.54
^{236}U	0.076	0.090	0.011	0.12	0.14	0.16	0.18	0.21	0.23	0.26	20.34	11.0	7.77	6.70
Discharge, kg														
^{232}Th	1,520	1,516	1,512	1,508	1,504	1,499	1,495	1,491	1,490	1,483	1,512	1,512	1,512	1,513
^{233}U	37.8	38.2	38.5	38.7	39.0	39.2	39.4	39.5	39.7	39.7	42.4	44.3	44.3	44.4
^{234}U	6.85	7.15	7.43	7.72	8.00	8.27	8.53	8.79	8.85	9.28	18.5	19.5	18.9	18.5
^{235}U	10.4	9.65	9.00	8.42	7.90	7.44	7.03	6.67	6.30	6.06	8.80	7.79	6.61	6.43
^{236}U	11.5	11.6	11.6	11.7	11.7	11.7	11.7	11.7	11.6	11.6	12.3	9.80	6.76	5.94
^{238}U	3.99	3.95	3.91	3.88	3.84	3.80	3.76	3.73	3.69	3.65				
^{239}Pu	0.094	0.093	0.093	0.092	0.091	0.091	0.090	0.089	0.089	0.088				
^{240}Pu	0.049	0.048	0.048	0.048	0.047	0.047	0.047	0.046	0.046	0.046				
^{241}Pu	0.047	0.047	0.047	0.047	0.047	0.046	0.046	0.046	0.046	0.045				
^{242}Pu	0.050	0.053	0.055	0.057	0.059	0.061	0.063	0.065	0.067	0.068				

[a]For each of five fuelings hereafter.

[b]At full-power operation.

[c]99% of fertile particle discharge delayed 0.8 full-power years.

Table C.6 (continued)

Fueling event	46–50	51–55	56–60	61–65	66–70	71–75	76–80	81–85	86–90	91–95	Thereafter	Final loading
Time,[a] years	7.067	7.733	8.400	9.067	9.733	10.400	11.067	11.733	12.400	13.066	13.733	
Fresh makeup feed, kg												
^{232}Th	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	
^{233}U	26.1	26.0	17.9	22.0	24.1	23.4	21.9	26.2	21.3	22.6	23.0	
^{234}U	9.48	9.44	6.50	7.99	8.71	8.50	7.95	9.51	7.73	8.21	8.35	
^{235}U	5.45	5.43	3.74	4.60	5.03	4.88	4.58	5.47	4.45	4.72	4.80	
^{236}U	4.11	4.10	2.81	3.47	3.80	3.69	3.45	4.13	3.36	3.56	3.62	
^{238}U												
Recycle feed,[b] kg												
^{233}U	44.0	44.1	44.5	45.8	46.1	46.4	46.8	47.2	46.6	47.8	48.0	
^{234}U	18.3	18.1	18.1	24.7	24.0	24.1	24.2	24.3	22.2	27.3	28.0	
^{235}U	6.36	6.27	6.32	9.90	9.47	9.52	9.62	9.73	8.70	11.5	12.0	
^{236}U	5.88	5.30	5.02	14.7	12.4	11.7	11.2	10.9	9.09	19.7	20.0	
Discharge, kg												
^{232}Th	1,513	1,514	1,515	1,516	1,517	1,517	1,518	1,518	1,519	1,519	1,519	37,560
^{233}U	44.6	44.9	46.3	46.5	46.9	47.3	47.7	47.1	48.3	48.5	48.5	1,350
^{234}U	18.2	18.2	24.8	24.2	24.3	24.4	24.6	22.4	26.9	27.9	28.3	720
^{235}U	6.34	6.39	10.0	9.86	9.62	9.71	9.83	8.79	11.5	11.9	12.1	320
^{236}U	5.36	5.07	15.4	13.3	11.8	11.3	11.0	9.18	18.9	20.2	20.2	380
^{238}U												
^{239}Pu												
^{240}Pu												
^{241}Pu												
^{242}Pu												

[a] At full-power operation.

[b] 99% of fertile particle discharge delayed 0.8 full-power years.

Table C.7. Mass Balance for HTGR Reference Design with Bred ^{233}U in Initial Loading and Makeup, Full Recycle

Fueling event[a]	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Time,[b] years	0	1.200	1.333	1.466	1.600	1.733	1.866	2.000	2.133	2.266	2.400	2.533	2.666	2.800	2.933	3.067	3.200	3.333	3.467
Fresh makeup feed, kg																			
^{232}Th	38,777	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616
^{233}U	1,378	87.0	74.3	75.7	76.8	77.7	78.5	39.7	38.9	38.1	37.2	36.4	35.7	27.3	26.7	26.1	25.5	24.9	24.2
^{234}U	552	34.8	29.7	30.3	30.7	31.1	31.4	15.9	15.6	15.2	14.9	14.5	14.3	10.9	10.7	10.4	10.2	9.89	9.63
^{235}U	193	12.2	10.4	10.6	10.8	10.9	11.0	5.56	5.45	5.33	5.21	5.09	4.99	3.84	3.75	3.66	3.57	3.47	3.39
^{236}U	192	12.1	10.3	10.5	10.7	10.8	10.9	5.52	5.41	5.29	5.17	5.06	4.95	3.80	3.71	3.63	3.54	3.45	3.37
Recycle feed,[c] kg																			
^{233}U								35.4	36.1	36.8	37.5	38.1	38.6	44.3	44.3	44.3	44.3	44.3	44.3
^{234}U								10.0	10.2	10.4	10.5	10.7	10.8	18.0	18.0	18.0	18.0	18.0	18.0
^{235}U								3.34	3.36	3.39	3.42	3.50	6.52	6.52	6.53	6.55	6.56	6.58	6.59
^{236}U								3.33	3.35	3.38	3.40	3.42	3.44	6.87	6.91	6.93	6.96	6.99	7.02
Discharge, kg																			
^{232}Th		1,577	1,573	1,569	1,565	1,561	1,556	1,552	1,548	1,544	1,540	1,536	1,533	1,529	1,525	1,521	1,517	1,513	1,509
^{233}U		35.7	36.5	37.2	37.9	38.5	39.0	44.8	44.8	44.8	44.8	44.8	44.8	47.8	47.5	47.3	47.0	46.8	46.6
^{234}U		10.2	10.3	10.5	10.6	10.8	10.9	18.2	18.2	18.2	18.2	18.1	18.1	24.3	24.1	24.0	23.8	23.6	23.5
^{235}U		3.38	3.39	3.43	3.45	3.49	3.53	6.59	6.60	6.61	6.63	6.64	6.65	9.49	9.46	9.43	9.40	9.36	9.32
^{236}U		3.36	3.39	3.41	3.43	3.45	3.47	6.94	6.97	7.00	7.03	7.06	7.09	10.6	10.7	10.7	10.7	10.8	10.8

[a]For fueling events 20 through final loading, see Table C.7 (continued).

[b]At full-power operation; time shown to the start of a period.

[c]99% of entire uranium discharge delayed 0.8 full-power years.

Table C.7 (continued)

Fueling event	20	21	22	23	24	25	26	27	28	29	30–34[a]	35–39	40–49[b]	50–59	60–69	Thereafter	Final loading
Time,[c] years	3.600	3.733	3.867	4.000	4.133	4.267	4.400	4.533	4.667	4.800	4.933	5.600	6.266	7.600	8.933	10.266	
Fresh makeup feed, kg																	
^{232}Th	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	1,616	
^{233}U	16.9	16.3	15.7	15.1	14.4	13.8	21.8	24.9	24.9	25.1	25.0	26.1	24.4	22.7	25.0	23.5	
^{234}U	6.71	6.53	6.27	6.03	5.77	5.52	8.74	9.96	9.97	10.0	9.99	10.4	9.75	9.07	9.99	9.39	
^{235}U	2.36	2.28	2.20	2.12	2.01	1.93	3.07	3.50	3.50	3.52	3.51	3.66	3.43	3.19	3.51	3.30	
^{236}U	2.35	2.27	2.18	2.10	2.00	1.92	3.04	3.47	3.47	3.49	3.48	3.63	3.40	3.16	3.48	3.27	
Recycle feed,[d] kg																	
^{233}U	47.3	47.0	46.8	46.5	46.3	46.1	47.6	47.2	46.9	46.5	47.3	48.0	47.9	47.7	49.0	50.6	
^{234}U	24.1	23.9	23.7	23.5	23.4	23.2	28.3	28.0	27.7	27.4	26.5	25.0	23.8	25.7	28.6	30.6	
^{235}U	9.40	9.37	9.34	9.30	9.27	9.23	11.7	11.6	11.5	11.4	11.5	9.91	9.32	10.9	12.1	14.0	
^{236}U	10.5	10.5	10.6	10.6	10.6	10.7	14.2	14.2	14.3	14.3	13.0	11.1	10.5	15.2	16.3	21.0	
Discharge, kg																	
^{232}Th	1,505	1,502	1,497	1,494	1,490	1,486	1,518	1,518	1,518	1,518	1,518	1,519	1,519	1,520	1,520	1,520	37,600
^{233}U	48.1	47.7	47.3	47.0	46.7	46.3	49.5	48.2	48.4	48.5	48.5	48.4	48.1	49.2	49.6	51.1	1,414
^{234}U	28.5	28.2	27.9	27.7	27.1	28.5	25.4	25.4	25.8	26.0	24.8	24.0	24.8	29.1	27.6	30.9	830
^{235}U	11.8	11.7	11.6	11.5	11.4	11.3	11.7	10.2	10.3	10.5	9.76	9.30	9.90	12.3	11.6	14.1	354
^{236}U	14.3	14.4	14.4	14.4	14.5	14.5	13.5	11.5	11.7	11.9	10.7	10.4	12.7	17.4	14.7	21.2	525

[a]Five of these.

[b]Ten of these.

[c]At full-power operation; time shown to the start of a period.

[d]99% of entire uranium discharge delayed 0.8 full-power years.

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