

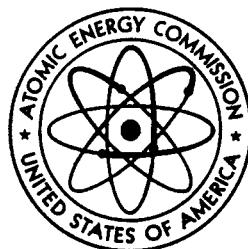
EVALUATION

AEC -- 95

# GAS COOLED FAST REACTOR

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U.S. ATOMIC ENERGY COMMISSION

AN EVALUATION  
OF  
THE GAS COOLED FAST REACTOR

JUNE 1972

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## AN EVALUATION OF THE GAS COOLED FAST REACTOR

### I

#### INTRODUCTION

Although most of the development work on fast breeder reactors has been devoted to the use of liquid metal cooling, interest has been expressed for a number of years in alternative breeder concepts using other coolants. One of a number of concepts in which interest has been retained is the Gas-Cooled Fast Reactor (GCFR). As presently envisioned, it would operate on the uranium-plutonium mixed oxide fuel cycle, similar to that used in the Liquid Metal Fast Breeder Reactor (LMFBR), and would use helium gas as the coolant.

The long-term objective of any new reactor concept and the incentive for the government to support its development is to help provide a self-sustaining, competitive industrial capability for producing economical power in a reliable and safe manner. Successful achievement of this objective is required to permit the utilities, and others, to consider the concept as a viable option to existing power production systems and to gain public acceptance of a new form of power production. It is only after this is achieved that the utilities and industry could make the heavy, long-term commitments of resources in funds, facilities and personnel to provide the transition from the early experimental facilities and demonstration plants to full scale commercial reactor power plant systems. Consistent with the policy established for all power reactor

development programs, the GCFR would require the successful accomplishment of three basic phases:

- . An initial research and development phase in which the basic technical aspects of the GCFR concept are confirmed, involving exploratory development, laboratory experiment, and conceptual engineering.
- . A second phase in which the engineering and manufacturing capabilities are developed. This includes the conduct of in-depth engineering and proof testing of first-of-a-kind components, equipment and systems. These would then be incorporated into experimental installations and supporting test facilities to assure adequate understanding of design and performance characteristics, as well as to gain overall experience associated with major operational, economic and environmental parameters. As these research and development efforts progress, the technological uncertainties would need to be resolved and decision points reached that would permit development to proceed with necessary confidence. When the technology is sufficiently developed and confidence in the system is attained, the next stage would be the construction of large demonstration plants.

- . A third phase in which the utilities make large scale commitments to electric generating plants by developing the capability to manage the design, construction, test and operation of these power plants in a safe, reliable, economic, and environmentally acceptable manner.

Significant experience with the Light Water Reactor (LWR), the High Temperature Gas-Cooled Reactor (HTGR) and the Liquid Metal Fast Breeder Reactor (LMFBR) has been gained over the past two decades pertaining to the efforts that are required to develop and advance nuclear reactors to the point of public and commercial acceptance. This experience has clearly demonstrated that a logical progression through each of the three phases is an extremely difficult, time consuming and costly undertaking, requiring the highest level of technical management, professional competence and organization skills. This has again been demonstrated by the recent experience in the expanding LWR design, construction and licensing activities which emphasizes clearly the need for even stronger base technology and engineering efforts than were initially provided, although these were satisfactory in many cases for the first experiments and demonstration plants. The LMFBR program, which is relatively well advanced in its development, tracks closely this LWR experience and has further reinforced this need as its applies to the technology, development and engineering application areas.

It should also be kept in mind that the large backlog of commitments and the shortage of qualified engineering and technical management personnel and prooftest facilities in the government, in industry and in the utilities makes it even more necessary that all the reactor systems be thoroughly designed and tested before additional significant commitment to, and construction of, commercial power plants is initiated.

The large scale commitments to the uranium-plutonium fuel cycle through purchases of the LWRs and the substantial investment of the Nation's resources engendered by these commitments led the U. S. Atomic Energy Commission (AEC), Division of Reactor Development and Technology to initiate reviews in 1966 of the technical status and the possible benefits implicit in the development of the various reactor concepts being considered in the civilian nuclear power program for meeting future power needs. These reviews were needed to help provide guidance on making effective use of our national resources and help determine the requirements for, and allocations of, these resources. With regard to the GCFR, the AEC evaluated this concept in 1969 and issued "An Evaluation of Gas-Cooled Fast Reactors," WASH-1089, April 1969, along with a companion report "An Evaluation of Alternate Coolant Fast Breeder Reactors," WASH-1090, April 1969.

The GCFR designs evaluated in WASH-1089 were based, in large measure, on information provided by the single industrial developer of the GCFR, and

therefore, the report generally reflected this viewpoint and enthusiasm. The information in WASH-1089 was used as input to the subsequent systems analysis (WASH-1098) and cost-benefit studies (WASH-1126, USAEC, "Cost-Benefit Analysis of the U.S. Breeder Reactor Program," 1969 and WASH-1184, USAEC, "Updated (1970) Cost-Benefit Analysis of the U.S. Breeder Reactor Program," January 1972).

WASH-1098, "Potential Nuclear Growth Patterns," December 1970, was prepared by the Systems Analysis Task Force (SATF) which was concerned with the development and application of a model of the U.S. electrical power economy. The model input data was provided by the individual task forces charged with the evaluation of the reactor concepts under consideration, including GCFR. The results of this evaluation, as it applied to GCFR, indicated that largely because of the uncertainties in the cost estimates for both the LMFBR and the GCFR, it was not possible to draw a definitive conclusion concerning the soundness and desirability of conducting a parallel breeder program on the GCFR in addition to the LMFBR. Accordingly, WASH-1098 concluded that "the current state of knowledge is not sufficient to support a definitive evaluation of the potential of the GCFR against that of the LMFBR."

The consistent conclusion reached in the cost-benefit studies (WASH-1126 and -1184), viz., sufficient information is available to indicate that

the projected benefits from the LMFBR program can support a parallel breeder program, is highly sensitive to the assumptions on plant capital costs. With the recognition that even for ongoing concepts on which ample experience exists, capital costs and especially small estimated differences in costs are highly speculative for plants to be built 15 or 20 years from now, it is questionable whether analyses based upon such costs should constitute a major basis for decision making relative to the desirability of a parallel breeder effort.

In compliance with a request from the Office of Science and Technology for further review of the GCFR at this time, the AEC has undertaken this internal assessment which examines the technical developments that have taken place in the continuing research and development and design efforts on the GCFR system. This request recognized that the GCFR has been in the initial research and development phase with emphasis on the development of basic GCFR technology. The program has been carried out primarily at Gulf General Atomic (GGA) and has been supported by government-sponsored research at a level of about \$1 million per year over the past several years. The utility industry is supporting research and development on this concept at a level of about \$1.4 million per year. Total expenditures from 1963 to date on GCFR technology have amounted to approximately \$17 million.

The GCFR is still in the early phases of an overall R&D effort notwithstanding the benefits expected to accrue to it from the HTGR and LMFBR programs. An adequate investigation of the problems associated with the GCFR concept to move forward toward their satisfactory resolution would require a substantial research and development program along the lines outlined herein with sufficient proof-testing of major components and systems to provide assurances commensurate with the large commitment and investment in such plants for large-scale power generation. Some prerequisites for such a program are provided by portions of major ongoing programs, e.g., some of the fuel development, physics and safety work in the Liquid Metal Fast Breeder Reactor (LMFBR) program and component development in the High Temperature Gas Reactor (HTGR) program. Also, component and plant operating and maintenance experience at the 330 MWe Fort St. Vrain reactor will provide important basic information. However, most of the critical and unique characteristics of the GCFR are not adequately represented in research and development programs on other concepts. In addition, the flexibility for resolution of such development and engineering problems normally required by the designers, constructors, and operators of even the more proven concepts is restricted due to the compact arrangement proposed for the GCFR. Consequently, if feasibility were to be confirmed through appropriate experimental reactor and other facility operation, the additional overall technical effort needed for such a full-scale program on the

GCFR could be comparable in magnitude to the efforts on other major reactor development programs such as the LWR and LMFBR programs.

Experience in reactor development programs in this country and abroad has demonstrated that different organizations in evaluating the projected costs of introducing a reactor development program and carrying it forward to the point of large-scale commercial utilization, would arrive at different estimates of the methods, scope of development and engineering efforts, and the costs and time required to bring that program to a stage of successful large scale application and public acceptance.

Based upon the extensive program required to bring a new concept to commercial utilization, and taking into account the benefits expected to accrue from progress in the LMFBR and HTGR programs over the next 5 years, the incremental cost to the government of a parallel breeder program of this type has been estimated by the AEC to range up to about \$2 billion in undiscounted direct costs, (WASH-1184). The GCFR would require magnitude of funding up to this level in order to establish the necessary technology and engineering bases; obtain the required industrial capability; and advance through a series of test facilities, reactor experiments, and demonstration plants to a commercial GCFR, safe and suitable to serve as a major energy option for central station power generation in the utility environment.

## II

SUMMARY

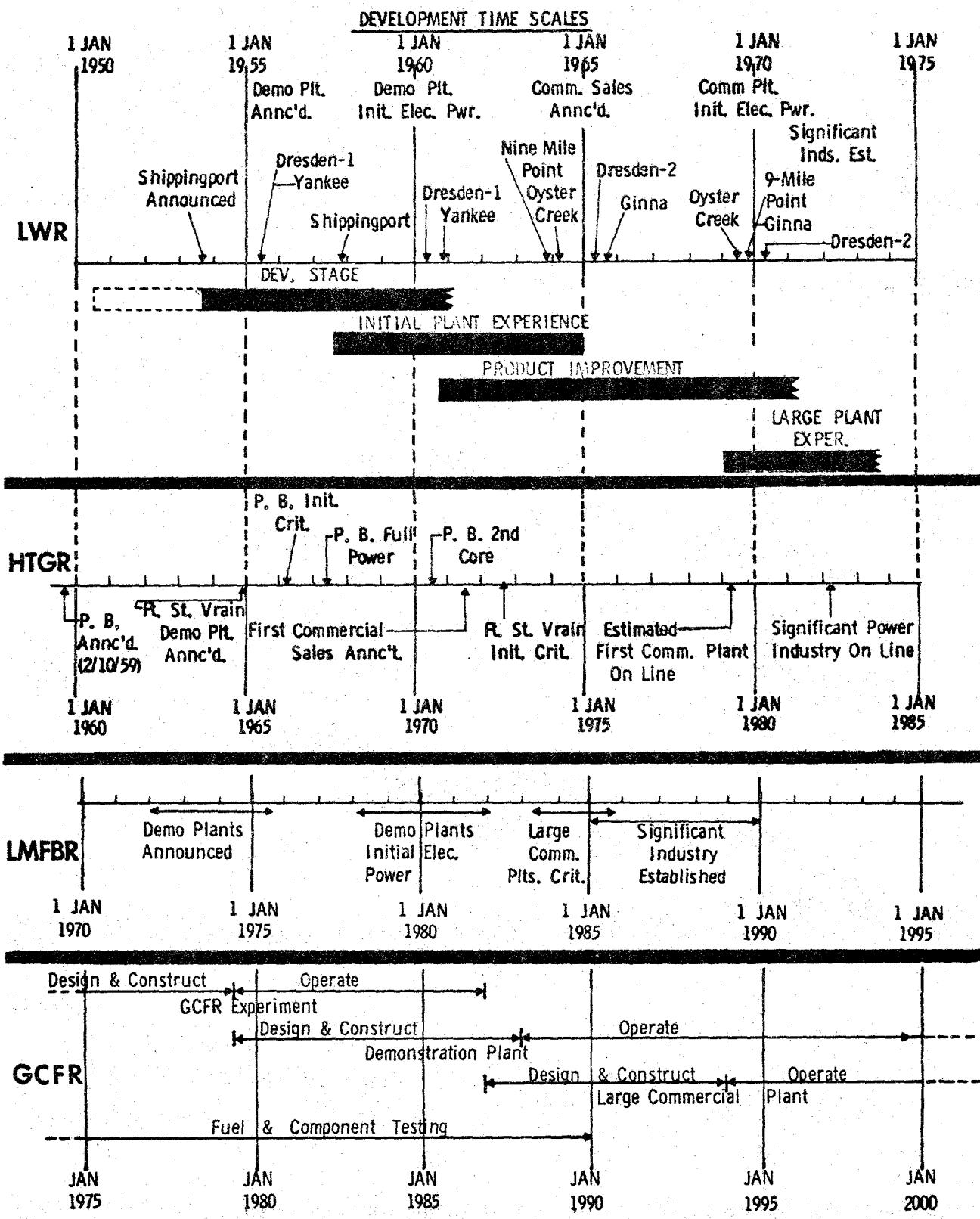
The GCFR concept uses helium as the coolant gas, which leads to several potentially favorable attributes of the GCFR. Helium is both optically and neutronically transparent and does not become radioactive. The GCFR has a potentially high breeding ratio resulting largely from the coolant properties. Since the use of gas cooling requires a high coolant pressure to secure adequate heat transfer, the GCFR is subject to the possibility of depressurization accidents as well as loss-of-flow accidents. The potential for occurrence of these types of accidents requires resolution of significant reactor damage and safety questions which are unique to the GCFR concept. In addition, there are other outstanding engineering problems associated with safety, reliability and maintainability which differ significantly from similar considerations for the HTGR and LMFBR.

The additional technical work done since the publication of WASH-1089 and WASH-1098, both by the AEC and private industry, and further definition of problem areas aided by discussions with the Regulatory staff and the Advisory Committee on Reactor Safeguards (ACRS) have served to reinforce the earlier conclusion of WASH-1089 that a substantial program would be needed to support the commercial introduction of the GCFR. This is particularly so since only limited effort on the GCFR is underway outside

the United States and few benefits can accrue from significant parallel development work and related large scale commitments in other countries such as is the case for the LMFBR concept. Based upon the information currently available, as discussed in this report, there have been no significant changes in the GCFR development status, relative to the total effort required for its commercial application, that would appreciably alter the general conclusions of WASH-1089.

Although the proponents of the GCFR concept would intend to depend heavily upon the technology developed in the High Temperature Gas Reactor (HTGR) program and, to the extent possible, the technology of the LMFBR program, the GCFR can be characterized as being in only the early stages of an overall development program. This should not be surprising, since the government and industrial funds expended on this concept through FY 1972 total less than \$20 million. With a significantly increased commitment to an R&D program along with personnel and other resources made available on a high priority basis, the development time scale which could optimistically be expected for development of the GCFR would have to be in the context of the approach and experience with the LWR, HTGR and LMFBR programs, Fig. 1. With the size of the major government and industrial investment, recent experience with LWR's on problems arising in multiple commitments to full size power plants provides ample evidence of the necessity to provide a broad industrial base and to conduct extensive, in-depth development and testing efforts addressed to all critical components

Figure 1



and systems throughout all project phases. An overall full-scale program on the GCFR would be expected to cover those phases that are currently under consideration for the LMFBR Program. (See Figure 1 and Figure 4, p. 41). Such a program would involve costs comparable to other reactor development programs (in the range of two billion dollars from the Government) in addition to large outlays and commitments by industry and the utilities. Such a GCFR program would have to take into account any overlapping or concurrent commitments and availability of resources with the HTGR commercialization efforts. This, of course, is a problem faced by other reactor programs as well.

## III

BASIS FOR WASH-1089 AND ASSOCIATED EVALUATIONSBackground

GCFR development was initiated in November 1963 by the AEC under a contract with Gulf General Atomic (GGA)\* to investigate the concept which had evolved from earlier privately supported GCFR studies. The AEC-sponsored work outlined a development program that started with the objective of a gas-cooled fast reactor experiment of 50 MWT which was to lead to a demonstration power plant as a step towards a full-scale plant. An outcome of the next year of AEC sponsored R&D (1964) was a conceptual design for a reactor experiment which would serve as a test bed for fuel development, and would provide experience in designing and constructing a special prestressed concrete reactor vessel (PCRV).

In the period 1965-68, the AEC and GGA continued studies of the GCFR. A conceptual design for a 1000 MWe GCFR power plant was evolved which featured a horizontal PCRV instead of the original vertical arrangement and also differed in other important respects from the original concept. This effort incorporated ideas of the utility companies, particularly as to the layout and design of the nuclear steam supply components from the viewpoint of operation, maintenance and safety. Also, a new design for a reactor experiment was developed to reflect engineering aspects of the new large plant design.

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\*Gulf General Atomic formerly was the General Atomic Division of General Dynamics.

The 1000 MWe GCFR reference conceptual design prepared by GGA for study by the AEC Fast Breeder Reactor Alternate Coolant Task Force<sup>2,3,4</sup> was an extension of the above conceptual design. The task force analyses were reported in WASH-1089 dated April 1969. A brief description of the GCFR plant designs considered in this study is included in the next section. A more detailed description may be found in WASH-1089. During 1967, to satisfy the needs of the Alternate Coolant Task Force as well as to meet AEC contractual requirements, a Preliminary Development Plan for the GCFR was also prepared.<sup>5</sup>

WASH-1089, WASH-1126, and WASH-1184

The conclusions of WASH-1089 were based upon the 1000 MWe plant conceptual design provided to the Alternate Coolant Task Force. The development plan and costs projected in WASH-1089 were associated with bringing the concept to the stage of commercial introduction including construction and operation of a reactor experiment to help demonstrate the adequacy of the fuel performance.

As noted at the time WASH-1089 was published, the designs evaluated were based on information provided by developers of GCFR and generally reflected their viewpoints and opinions on achievable technology.

Based upon the information in WASH-1089, the cost of development of a parallel breeder was estimated and examined in the context of the overall LMFBR development program. As noted in WASH-1126, "Cost-Benefit Analysis

of the U.S. Breeder Reactor Program," April 1969, if the LMFBR were to be introduced in 1984 or earlier, it was concluded that sufficient benefits would be generated to support the cost of a parallel breeder program and still maintain a benefit-cost ratio in excess of one.

Although the LMFBR introduction date was deferred to 1986 in WASH-1184, "Updated (1970) Cost-Benefit Analysis of the U.S. Breeder Reactor Program," January 1972, the conclusions remain essentially unchanged, viz., a tentative case can be made to expand the industrial breeder base by establishing a parallel breeder program.

It should be stressed that the cost-benefit ratios derived in both WASH-1126 and WASH-1184 depend heavily upon the assumed capital costs for the various power plants, especially the breeders. The sensitivity of the results to a slight increase in the capital cost of the GCFR was illustrated in WASH-1098 (page 6-50), where an increase of 7.7% in the assumed cost of a GCFR from 130 to 140 \$/KWe permitted the LMFBR to compete at an assumed cost of 150 \$/KWe. The capital cost studies performed over the past two years have confirmed the tenuousness of utilizing assumed small capital cost differences such as these. Variations in design, plant location, labor productivity and cost, escalation, interest rates and periods of construction can readily produce capital cost differences of 10 to 15%. Capital costs for all of the breeder designs considered in past studies were only best estimates at the time, normalized to some extent for unit costs. At present, while a reasonable capability to predict costs of

LWR designs has been developed over the past few years, the calculated capital costs for plants in various locations in the U.S. based on relatively well established LWR technology, will vary 15% or more. Other factors such as safety, seismological, and environmental features can materially increase estimated costs. On this basis, the use of capital cost comparisons between concepts yet to be developed and then to be built 15 or 20 years from now as a major basis for programmatic decisions and choices is highly questionable.

## IV

GCFR DESIGNS PRESENTED IN WASH-1089ORNL Working Group

The gas-cooled fast reactor designs reviewed in WASH-1089 were developed by GGA.<sup>2,3,4</sup> Two oxide fueled designs were submitted to the Oak Ridge National Laboratory (ORNL) working group for evaluation. A carbide fueled design was also prepared by GGA, but it was submitted too late in the study to allow evaluation.

Oxide Fueled GCFR Designs

The two oxide fueled designs were designated "reference" and "derated" designs. Some principal characteristics of the two oxide fueled designs are given in Table 1 which has been modified from Table 2.1 of WASH-1089 to show the parameters for the latest available 1000 MWe GCFR plant design. Design parameters are also shown for a 300 MWe demonstration plant.

The entire primary cooling system for the oxide designs was housed within a horizontal PCRV which contained the centrally located vertical reactor core, four steam generators and their associated circulators located in internally isolated compartments at the ends of the vessel. The PCRV had double containment of all penetrations, individual standpipes above each core and blanket element, and a watercooled, but uninsulated, steel liner. The whole primary circuit flow path was confined within insulated ducting and shells surrounding the reactor and the steam generators; thus the main internal compartments were at substantially room temperature. These compartments were continuously purged with clean helium.

Table 1  
Summary of Oxide-Fueled GCFR Design Characteristics<sup>a</sup>

Urgent Info Feb. 1972

	GCFR-4, Reference Design	GCFR-4 Derated Design	300 Mw(e) Plant <sup>c</sup>	300 Mw(e) Plant <sup>c</sup>	1000 Mw(e) Plant <sup>d</sup>
Power					
Reactor nuclear power, Mw	2530	2681	826	311	1050
Net electrical power, Mw	1000	1000	37.6	37.6	38.0
Net thermodynamic efficiency, %	39.5	37.3			
Coolant					
Composition					
Core inlet pressure, psia					
Reactor pressure drop, psi					
Flow rate, lb/hr					
Temperatures, °F (°C)					
Reactor inlet	629 (332)	589 (338)	593	593	602 (317)
Core inlet	635 (334)	595 (313)	593	593	602 (317)
Core outlet	1183 (639)	1088 (587)	1007	1007	1100 (594)
Reactor outlet	1190 (643)	1094 (590)			
Steam plant conditions					
Leaving superheater					
Pressure, psia					
Temperature, °F	2515	1850	2900	2900	2900
Condenser pressure, in. Hg	1000	900	875	875	925
Core thermal performance (at 100% power)	1.5	1.5	1.75	1.75	1.75
Peak linear heat rating, kw/ft	17.8	16.3	12.5	12.5	13
Maximum cladding surface temperature, °F (°C)	1315 (713)	1205 (652)	1205 (652)	1205 (652)	1205 (652)
Nominal					
Local hot spot	1425 (779)	1315 (713)	1290	1290	1290
Maximum interior fuel temperature, °F (°C)	4935 (2724)	4765 (2629)	4765 (2629)	4765 (2629)	4765 (2629)
Nominal					
Local hot spot	5925 (3234)	5039 (2787)	5039 (2787)	5039 (2787)	5039 (2787)
Peak-to-average core power ratio	1.47	1.47	1.47	1.47	1.47
Mean core power density, kw/in <sup>3</sup> /liter					
Mean effective fuel rating, Mw/in <sup>3</sup> /kg	280	260	260	260	260
Core and blanket description					
Active core volume, liters	0.86	0.82	0.82	0.82	0.82
Active core length, cm (in.)	8510	9712	9712	9712	9712
Active core diameter, cm (in.)	148.2 (58.4)	152.8 (60.9)	152.8 (60.9)	152.8 (60.9)	152.8 (60.9)
Axial blanket thickness, cm (in.)	268.5 (106.0)	281.5 (110.8)	281.5 (110.8)	281.5 (110.8)	281.5 (110.8)
Radial blanket thickness, cm (in.)	60 (23.6)	60 (23.6)	60 (23.6)	60 (23.6)	60 (23.6)
Core length-to-diameter ratio	51 (20.1)	51 (20.1)	51 (20.1)	51 (20.1)	51 (20.1)
Fuel material (core)	0.55	0.55	0.55	0.55	0.55
Fuel material (blanket)	Mixed Pu and depleted UO <sub>2</sub> (90% T.D.)	Depleted UO <sub>2</sub> (90% T.D.)	Mixed Pu & depleted UO <sub>2</sub>	Mixed Pu & depleted UO <sub>2</sub>	Mixed Pu & depleted UO <sub>2</sub>
Fuel distribution	Uniform axially; zoned radially to give 1.2 radial maximum-to-mean ratio	Uniform axially; zoned radially to give 1.2 radial maximum-to-mean ratio			

<sup>a</sup> Calculated values are results of ORNL review.<sup>b</sup> Calculated fuel temperatures above melting point of the oxide (4950°F) indicate that, according to ORNL calculations, local melting will probably occur.<sup>c</sup> Taken from ORNL-10067, dated May 15, 1970.<sup>d</sup> Taken from GA-10678 dated June 8, 1971.

The reactor core was assembled inside a vertical cylindrical steel core barrel, which spanned the PCRV from top to bottom. The core barrel served the multiple purpose of supporting the core grid plate, containing and directing the flow of coolant, and acting as a thermal shield. The core consisted of bundles of small diameter fuel rods contained in thinwalled square metal boxes. These fuel boxes were cantilevered downward from the deep section upper grid plate.

Both of the oxide fueled designs had free standing stainless steel-clad pins about 0.3 in. in diameter. The cladding surface was intentionally roughened over approximately 60 to 70% of the active core length to enhance the heat transfer coefficient in the downstream portion of the core and thereby prevent excessive cladding surface temperatures.

The designs also included an alternate fuel element design that had a pressure equalizing collection system to vent the fission product gases to a receptacle that was isolated from the coolant system. The objective of this manifolding system was to eliminate the large gas pressure differential across the cladding while preventing the fission product gases from escaping into the coolant.

#### Carbide Fueled GCFR Design

An advanced GCFR design to exploit the high density and excellent thermal conductivity of carbide fuel was also prepared by GGA. The carbide fuel design was submitted too late in the study period to be subjected to a technical evaluation. However, the design data were reviewed. (See Table 2). More details on this design are contained in WASH 1089.

Table 2 Comparison of Oxide and Carbide Fueled GCFR Designs\*

	Reference Oxide Fueled Reactor	Advanced Carbide Fueled Reactor
<b>Common conditions</b>		
Coolant	Helium	Helium
Total pumping power fraction, % of thermal	4	4
Design maximum cladding temperature, °C	700	700
Cladding (stainless steel) thickness-to-diameter ratio	0.04	0.04
Control-rod and structure blockage, %	10	10
Net electric output, Mw	1,000	1,000
Average enrichment of fresh core fuel, $^{239}\text{Pu} + ^{241}\text{Pu}/\text{U} + \text{Pu}$	0.127	0.127
<b>Appropriate core proportions</b>		
Core length-to-diameter ratio	0.55	0.40
Core volume, liters	8,510	4,030
Active length, cm	148	93
Active diameter, cm	270	234
Fuel volume fraction	0.29	0.30
Coolant volume fraction	0.55	0.53
Rod diameter, cm	0.805	0.736
Number of rods	40,000	37,000
<b>Operating conditions</b>		
Core pressure, psi	1,250	1,750
Gas inlet temperature, °F	644	567
Gas outlet temperature, °F	1,175	1,057
<b>Steam conditions</b>		
Outlet temperature, °F	1000	900
Reheat temperature, °F	1000	900
Feed temperature, °F	375	275
Pressure, psi	2,400	1,800
Net plant efficiency	0.395	0.373
<b>Performance</b>		
Maximum rod heat load, kw/ft	18	30
Specific power, total Mw(th)/kg core fissile at startup	0.90	1.50
Power density, kw/liter	277	615
Conversion ratio, average	1.51	1.60
Assumed maximum burnup, Mwd/MT	$10^5$	$1.4 \times 10^5$
Core life, years	2.29	2.49
Out-of-pile time, years	1.0	0.5
Fractional increase in fissile plutonium per cycle	0.27	0.62
Geometric doubling time, years	8.8	4.3

\*Unevaluated characteristics presented by General Atomic.

Because of the potential advantages of the carbides and nitrides as fuel for the LMFBR or the GCFR, research has continued. The current advanced fuel technology program at Los Alamos Scientific Laboratory involves an effort of approximately \$750,000 per year and includes the irradiation of carbide and nitride fuel pins in the EBR-II reactor and transient tests in the TREAT facility. The current status of technology indicates that low density carbide fuels have promise for high burnup and high power density reactor applications. The major apparent problem involves swelling due to retention of fission gases. The retention of fission gas followed by a sudden release of this gas in the event of a temperature transient presents a potential safety problem that could result in clad distortion or failure.

The multi-cavity PCRV which was used in this design consisted of a vertical cylindrical block of prestressed and reinforced concrete containing multiple cavities in which the reactor core and the steam generators were individually housed. The cavities were interconnected by passages in the concrete through which the helium coolant flowed. The helium circulators, both main and auxiliary, were mounted in double containment penetrations in the vessel. The general arrangement of the components in the PCRV is shown in Fig. 2 of Section VI.

CHANGES SUBSEQUENT TO EARLIER EVALUATIONFuel Element Design Changes

Subsequent to the issuance of WASH-1089, certain changes in the reference GCFR fuel element design evolved. One change was a modified design for fuel venting as a means for equalizing the gas pressure inside and outside of the fuel cladding. GCFR fuel venting as first conceived<sup>2</sup> used tight seals between the fuel elements and the grid plate and required the recovery and storage of the released fission-gas effluent in gas cylinders. Further design iteration led to a proposed system in which fission gases would be removed from the element by a small differential pressure and diluted by a small stream of reactor coolant helium at the element-to-grid-plate connection. This approach would eliminate the requirement for a gas-tight joint where the fuel element joins the grid plate. The mixture of fission gases and helium diluent would be passed through a system of traps similar to those used in an HTGR helium purification system. The residual purified helium would be returned to the suction of the main helium circulator. This change in the venting design led to the adoption of fuel venting as a reference conceptual GCFR design feature by GGA. It should be noted, however, that the use of vented fuel eliminates the noble gas fission product containment feature regarded as an important advantage of non-vented fuel, with its attendant implications on other plant characteristics such as maintenance and fuel handling. Another major change was a lowering of the cladding hot-spot temperatures

by modified core coolant orificing design and physics design including partial reloading.

#### Possible Alternate Development Approach

With the decreased cladding temperature and the use of fuel venting in mind, an alternate development approach was chosen by the designer, namely, that the first GCFR facility could be a power-producing prototype plant based as closely as possible on LMFBR fuel and HTGR plant technologies rather than a reactor experiment. In the latter half of 1968 a conceptual design study of a 330 MWe GCFR Demonstration Plant was prepared.<sup>6</sup> This design was based on the proposed use of fuel venting and a cladding hot-spot temperature of 1382°F, which was lower than for earlier designs. This revised new plant design also incorporated the use of a PCRV of the cylindrical vertical multi-cavity type being developed for large HTGR plants, which is quite different from the type used in the WASH-1089 reference and derated designs. The primary virtue of the multi-cavity PCRV is the easier accessibility of primary system components.

An associated draft development program plan was also prepared, based initially on this conceptual design. Much of the overall GCFR effort subsequent to WASH-1089 has been addressed to the demonstration plant design rather than to a 1000 MWe plant design.

### Further Plant Design Modification

Additional design work has been performed which is addressed to a demonstration plant design. A reference demonstration plant design<sup>7</sup> was adopted by GGA in March 1970 based on a further 90°F reduction in cladding hot-spot temperature (1292°F) and improved predicted plant performance, resulting from more sophisticated reactor physics design and from resuperheating the steam after expansion through the circulator drive turbines. (See Table 1 of Section IV).

Subsequent to completion of the 300 MWe demonstration plant conceptual design, effort has been conducted on a more detailed safety analysis which was developed for this concept in collaboration with utility operating engineers and nuclear safety specialists representing the utility sponsors. This work has resulted in a Preliminary Safety Information Document,<sup>8</sup> which is now being reviewed by the AEC Regulatory Staff including the Advisory Committee for Reactor Safeguards (ACRS). A number of meetings on this document have been held and are continuing.

The latest GGA draft development program plan (Draft GA-A10788) is based on a systematic review of the 300 MWe GCFR demonstration plant conceptual design to define the problems that would require development work for a demonstration plant project. The scope of the plan is limited to the development needs of one GCFR demonstration plant by a single industrial participant, and thus encompasses a lesser program than the work described in WASH-1089 which was intended to cover an entire program with a competitive industrial capability through full commercial introduction of the GCFR.

## VI

DESCRIPTION OF DEMONSTRATION PLANT CONCEPTUAL DESIGN

The demonstration plant design is described because it is a more recent design than the latest available 1000 MWe GCFR plant design. The description emphasizes the nuclear steam supply system. The remainder of the plant is typical of modern high-temperature steam-turbine practice. The design is based on the utilization of LMFBR physics and fuel technology to the extent possible and on the continuing development of the component technology that forms the basis of the 40 MWe prototype HTGR at Peach Bottom, Pennsylvania, and the 330 MWe Fort St. Vrain HTGR Generating Station in Colorado and follow-on commercial HTGR's. Table 1 of Section IV gives the principal parameters.

Plant Configuration

The plant is housed in a reactor building, a fuel service building, and a turbine building. The reactor building contains the PCRV, which in turn contains the core, the helium primary coolant system and the steam generators. In addition, the reactor building functions as a secondary containment structure and also includes the fuel handling area and some reactor plant process and service systems. The fuel storage pool is in a fuel service building adjacent to the reactor building.

The configuration of the reactor and its associated primary circuit components, all of which are housed within the PCRV, is shown in Fig. 2 of

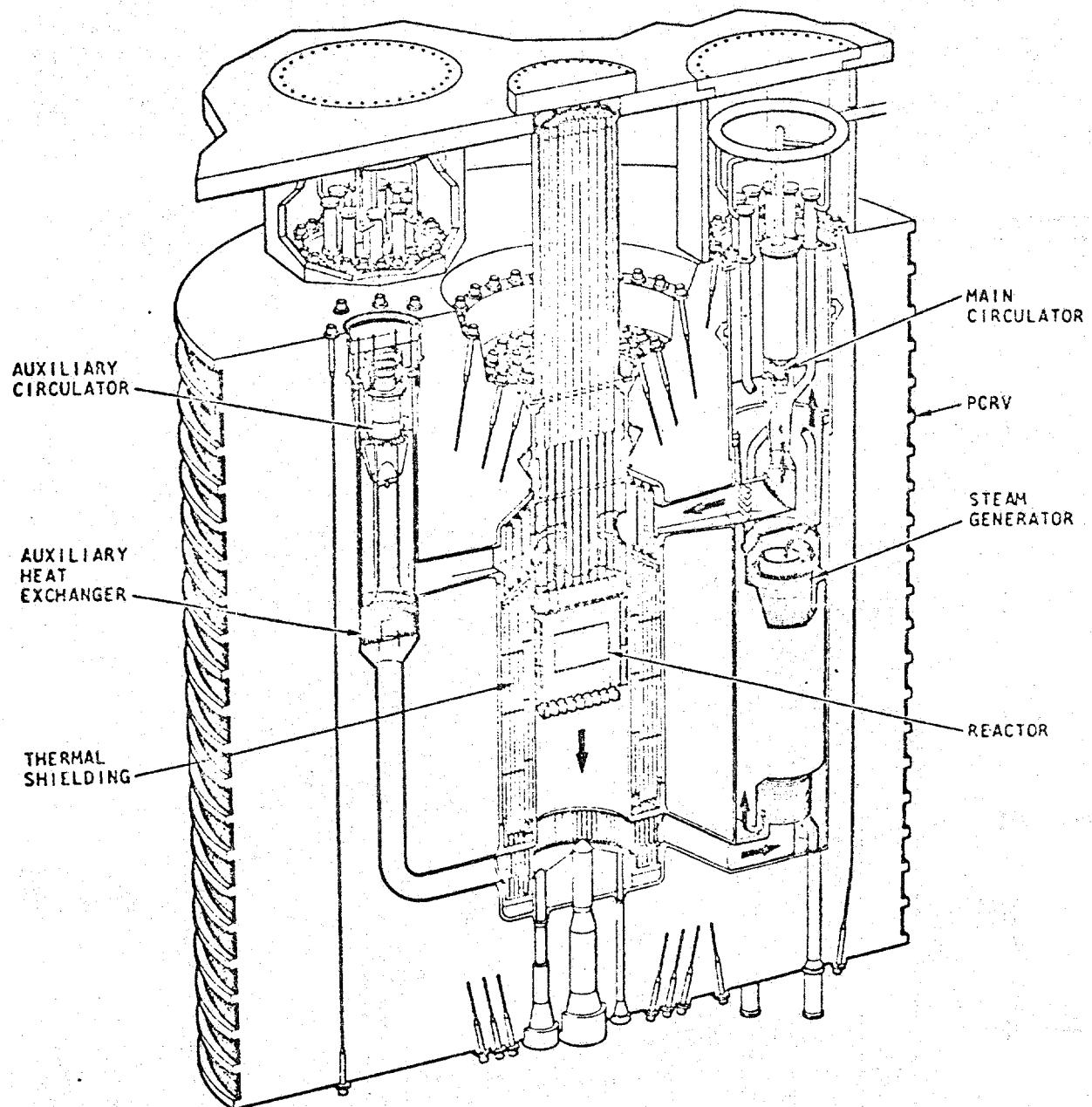


Figure 2 Cutaway view of PCRV showing the principal components of the nuclear steam supply system

this section. The PCRV is similar in principle to the PCRV's that have already been built in Europe and Great Britain and is also similar to GGA's large commercial HTGR design. An inner steel liner makes the PCRV leak-tight. The penetrations are provided with steel closures and structurally independent flow-limiting features where necessary to limit the depressurization rate in the event of primary barrier failure to a value that would not jeopardize core cooling. The large penetrations are closed by concrete plugs designed so as to always be in compression and held in place by two structurally independent and redundant means. These concrete plugs would not normally be removed and thus are joined to the vessel liner by seal welds. The liner is insulated by a thermal barrier, and cooling of the liner and penetrations is provided by cooling tubes on the concrete side of the steel liner. This PCRV is designed with redundancy of tension members that are inspectable and replaceable to preclude a gross failure of the pressure vessel.

#### Secondary Containment

To limit the consequences of a possible penetration closure failure, separate secondary containment is provided, similar to that proposed for the large commercial HTGR's. It performs two functions: it ensures a minimum coolant pressure (~2 atm) for core cooling following an accidental primary system depressurization and it confines fission products that potentially could be released from the fuel.

#### Primary Reactor System

The reactor coolant system contains three main loops, each with independent steam generators and circulators, and three auxiliary loops, each

with its own circulator and heat-removal system which are used as a backup for the main loops for shutdown cooling. The steam generators and their associated circulators are housed in vertical cavities in the walls of the PCRV surrounding the reactor core. The helium coolant, at a pressure of about 1250 psia, flows downward through the core, where it is heated to a temperature of about 1010°F at rated capacity. The flow is also downward across the tube banks of the helically coiled, once-through steam generators to accommodate the use of upflow boiling. This necessitates appropriate reversals in the gas flow path. These flow reversals are obtained by directing the core exit gas up through a central hole in the boiler and down through the tube bundles, up again around the boiler shells and then to top-mounted circulators from which the gas is discharged to the reactor top plenum at a temperature of about 595°F.

Main helium is circulated by three steam turbine-driven circulators of the type developed for the HTGR, which employs a single axial compression stage driven by a single impulse steam turbine stage that is in series with the steam generator. Auxiliary circulation is provided by electrically driven centrifugal circulators powered by separate and individually driven alternators.

#### Reactor Core

The reactor core is composed of 211 hexagonal fuel and radial blanket elements containing the fuel and blanket rods. The elements, which are

10 ft. in total length and about 6 1/2 in. across the flats, are supported from a top-mounted single grid plate to which each element is rigidly attached; they are clamped to the grid plate solely at their "cold" ends. The grid plate, which consists of an 11 ft. diameter, 2 ft. thick disc with closely spaced 6-in. diameter holes to accommodate the circular extensions of the fuel elements, is itself top supported by a surrounding cylinder connected to the liner of the top penetration in the PCRV. The clamping of the fuel elements to the grid plate, as well as the means for operating a variable orifice within each fuel element, is facilitated by providing individual small penetrations above each element in the top access plug.

In the reactor core there are 27 control fuel elements, each of which contains a movable control rod. Twenty-one of these rods are operated as conventional control rods for normal control, burnup, and shutdown requirements; each has a reactivity worth of approximately \$0.85. Some of these rods will normally be fully or partially inserted in the reactor core during power generation. The other 6 control rods provide additional shutdown capability and are a backup emergency shutdown system. These 6 rods have a reactivity worth of \$1.60 each. They will always be fully withdrawn from the core during power generation; therefore, they are not subject to burnup or to irradiation heating as are the conventional control rods.

All control rods are physically identical except that surface roughening is used to enhance the heat transfer from the 21 standard control rods.

The 6 secondary shutdown rods have smooth surfaces. The neutron absorber material is  $B_4C$ . The worth of the rods is established by adjustments in  $B^{10}$  isotope enrichment. The mechanical design of all the absorbers is the same.

The absorber section of the control rod is vented to the surrounding coolant to provide pressure equalization. A vent hole and a metallic filter at the upper end of the absorber section allow the gas to escape to the fuel element inlet coolant stream but prevent particulate matter from leaving the absorber section.

The coolant flow in each fuel element is orificed so that the same hot spot cladding temperature is reached in each element, and on-line adjustment of all orifices is provided.

Pressure in the fuel rods and blanket rods is equalized to that of the reactor coolant by collective venting, and the fission gases pass through the vent manifold to the helium purification system. (See Figure 3). This system provides means for detecting and locating a fuel element with failed cladding. This is done by using radioactivity monitors on the vent lines.

## SCHEMATIC OF FUEL ELEMENT PRESSURE EQUALIZATION SYSTEM

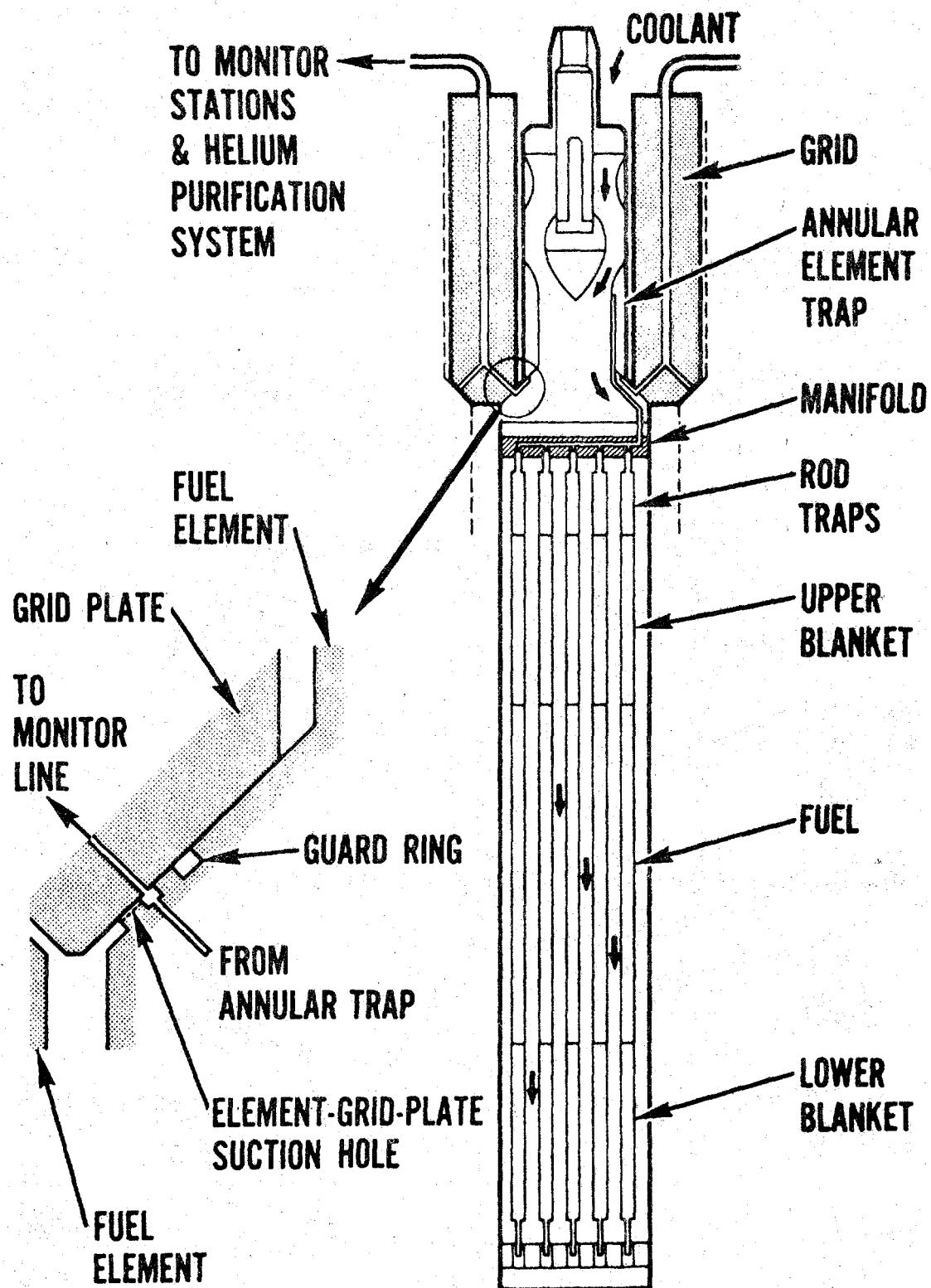


Figure 3

## VII

IMPACT OF MORE RECENT GCFR DEVELOPMENT  
ON WASH-1089 EVALUATION

Major Technical Changes Subsequent to WASH-1089

The previous discussion of GCFR development effort since the issuance of WASH-1089 is summarized as follows:

- (a) The modification of certain system parameters notably in fuel cladding temperature.
- (b) Further effort on fuel venting (pressure equalization).
- (c) A closer alignment of the GCFR design with the HTGR design to obtain as much benefit as possible from the Fort St. Vrain project and the development program for commercial HTGR plants.

This work has served to further identify the basic problems associated with the development of the GCFR. These are detailed in the draft development program plan (Draft GA-A10788) and are outlined below.

Component Development Needs

While the development of the GCFR depends to a significant degree on the successful operation, maintainability, and reliability of Fort St. Vrain (FSV) and large HTGR's, there are many major problem areas unique to the GCFR design that require extensive developmental effort

and proof testing, over and above HTGR needs. The fact that such efforts are proposed to be conducted concurrently with increased engineering scale-up efforts on the HTGR introduces special considerations. In addition, the use of a compact arrangement in the PCRV will impose design constraints heretofore extremely difficult and costly to handle in such first-of-a-kind engineering development programs. Such a complex development effort must be addressed in a detailed and disciplined manner based upon a carefully conceived plan and adequate test facilities.

1. Containment of the entire primary coolant system within the PCRV is fundamental to the GCFR concept. Additional model testing will be needed to validate the design at the higher GCFR pressure (1250 psi vs 700 psi for HTGR), although some of the PCRV development needed for GCFR might be accomplished as part of the HTGR development program, depending on exact timing and nature of needs. Such efforts must also be integrated closely with all of the major component development and first-of-a-kind engineering efforts.
2. There are a number of unique first-of-a-kind components that represent a significant engineering extrapolation from other first-of-a-kind components, some of which have yet to be designed and built and others of which have yet to be prooftested under actual operating conditions or operated in a reactor plant. These components and their associated maintenance equipment must be developed, fabricated and tested along with related development and proof-test facilities.

- a. The circulator horsepower rating for a 300 MWe GCFR is 4 times that for FSV, and 1 1/2 times that of the 1100 MWe HTGR's. In addition, the steam drive is based on 2900 psi steam versus 840 psi for FSV.
- b. The steam generators for the GCFR are 10 times larger than the FSV units in heat transfer surface area, and about the same as proposed for the large HTGR units. Boiling stability for the GCFR at normal load range and at flows as low as 2% of normal for the shutdown cooling phase has to be provided.
3. Reactor mechanisms such as control drives, orifice drives, and core clamping devices are unique to the GCFR and have to be operated in hot helium with the attendant problems of lubrication, prevention of self-welding, vibration, metallurgical creep and radiation damage in a fast neutron flux environment, although some of the experience with similar HTGR components may be applicable.
4. A difficult requirement exists to provide spent fuel cooling during removal from the core and transport to a water storage pool. Such requirements will be influenced by the proposed vented fuel concept.

5. The successful development and proof test of the GCFR refueling system and components also require first-of-a-kind engineering plus testing in the operating environment to demonstrate the adequacy of the design.
6. Special reactor instrumentation--some continuously operable, other for initial test purposes--is required to monitor the gas exit temperature from each fuel element, to measure vibrations and other parameters in the first-of-a-kind plant, and to assure design performance of all components under normal and abnormal operating conditions. Performance and reliability of such instrumentation in a high fast flux and high temperature environment pose very difficult development problems.
7. The GCFR core, because of the use of gas cooling with its heat transfer limitations, has a large core void fraction which leads to neutron streaming and leakage problems and may introduce problems relating to the internal shielding of components.

#### Fuel and Core Development Needs

In order to maximize the benefits to be gained from other ongoing activities, the GCFR effort in this category should utilize to the extent possible the spin-off technology from the large scale efforts

being carried out under the top priority LMFBR program. However, there are major areas unique to the GCFR concept that will require extensive and costly development.

1. While fast reactor physics methods and fundamental data developed by the expertise and from facilities at ANL for the LMFBR program will benefit the GCFR program, considerable additional effort will be required to meet GCFR needs. Further work is required on reactivity effects due to the presence of steam introduced into the core as a consequence of a steam generator tube failure.
2. The fuel rods proposed for the GCFR are different from those planned for LMFBR designs in having a larger diameter, roughened outer surface, and venting of the fuel rods and assembly. Surface roughening may affect the strength of the cladding and irradiation testing will be required to evaluate such effects.
3. Fuel venting introduces a number of questions that would require substantial development effort, including the rate of release of fission products from the fuel pellets, their diffusion rate through the length of the rod to the charcoal traps, the effects of breathing at the juncture of the fuel assembly vent and the grid plate that occurs with changes in plant load, fission product plateout throughout the vent system, charcoal behavior

under fast flux irradiation, the maintenance of alignment of seals under bowing and vibration stresses, and lastly the operation of the venting system as a whole under pressure transients with and without cladding leaks. The vented-fuel concept, which is attractive in principle for the LMFBR as well, has not been adopted in any current reactor system. It presents major design difficulties during plant operation and shutdown and during fuel handling, and it weakens the defense-in-depth safety concept by removing one barrier to the release of noble gas fission products. Little test information on this concept is available as yet and much more remains to be done before proof-of-feasibility can be established, including a variety of integral in-pile proof tests of prototypical fuel subassemblies and assemblies and safety tests relating to the vented concept, under a range of operating, transient and shutdown conditions. Fuel handling tests would also be needed.

4. There are heat transfer questions to be resolved as to the effects of rod spacers, fuel-element box walls, and possible rod bowing. Knowledge of the heat transfer over the whole range of flow from full power conditions on down is required for transient and safety analysis purposes.

5. Flow induced vibration of the fuel rods and of the whole fuel element is a potential problem to be overcome including the possibility of excessive seismically induced loads.
6. There are questions respecting the behavior of the interfaces at the spacer/fuel rod and the fuel element/grid in the hot helium environment.
7. Difficult problems, common to any fast reactor, are those related to obtaining the desired fuel burnup of 100,000 Mwd/T and coping with irradiation-induced swelling and creep for all metal parts in the core. Much reliable data are needed in these areas.

#### Safety Needs

In addition to many of the problems that have been raised on other reactors, there are a number of critical safety questions which have been identified for GCFR's beyond those noted above.

1. More detailed assurance including test data is needed to assure that adequate reliability of core cooling in potential emergency and faulted conditions can be provided. The plant design would need to have several main loops and auxiliary loops. If all main loops fail, shutdown cooling of the core would have to be maintained by the auxiliary loops, each of which would require its own electric-motor-driven circulator and water-cooled heat exchanger. Startup requirements and adequacy of reliability of the auxiliary cooling loops must be further analyzed and demonstrated.

2. Depressurization of the primary coolant system has been considered as the design basis for engineered safeguards in the GCFR. The maximum allowable depressurization rate would have to depend on some type of flow limiting devices in the large penetrations. This subject is being reviewed by the Regulatory staff to assess whether the system design can accommodate the proposed depressurization accident. Assurance must be given that the design provides adequate margin for a more rapid depressurization and for other concurrent failures.
3. Although analysis indicates that the reactivity change of the core is small and negative for all conceivable steam concentrations resulting from steam generator tube failures, additional analyses and critical assembly experiments must be carried out to confirm this.
4. The use of a core support system in which the fuel elements are tightly clamped at one end into a thick grid plate with no additional radial restraint, along with the potential deleterious effects of such materials phenomena as radiation damage and stainless steel creep, has led to detailed questioning by Regulatory and ACRS on the integrity of the system and the reactivity effects under transients and earthquakes. These problems will have to be resolved.
5. The adequacy of the proposed protection provided by control system actions backed up by two independent shutdown rod systems against anticipated transients must be proven. The consequences of failure of protective action in anticipated transients must be analyzed to establish their acceptability.

### Development Approach

The current GCFR draft development plan proposed by GGA does not call for many of those aspects of the reactor development program identified in Figure 4. For example, past experience in other reactor programs has demonstrated the vital need for a relatively small reactor experiment in addition to other test facilities to provide an essential contribution to the evidence of technical feasibility, to facilitate meaningful evaluations prior to a commitment for a large scale development program and a demonstration plant program. Considerable weight must be given to the many problem areas which historically have been discovered and solved as an outgrowth of specific operating experience on such experimental reactors. Cases in point are the many problems on small plants such as STR, EBWR, HRE, SIR, SRE, Saxton, EBR-I, EBR-II, Hallam, Fermi, MSRE, EBOR, UHTREX, EGCR, Piqua and BONUS, as well as the British experience with Dounreay.

WASH-1089 pointed out the need for a GCFR reactor experiment (1) to provide more information concerning the physics and safety of GCFR's, (2) to assist in the development and testing of GCFR components and (3) to provide a facility for fuel element development testing. Experience since publication of WASH-1089 appears to confirm the fact that a sound engineering approach requires a GCFR reactor experiment along with other related efforts outlined in Figure 4 to help establish the technical feasibility and requisite technical basis for this concept prior to full scale commitment to a demonstration plant program. Moreover, such accomplishments appear to be essential to permit meaningful evaluation or acceptance by the utility industry and others. (See Figure 5)

Figure 4

## PHASES IN ACHIEVING LMFBR PROGRAM OBJECTIVES

- OVERALL OBJECTIVE OF LMFBR PROGRAM - ACHIEVE EARLY ESTABLISHMENT OF SELF-SUSTAINING, COMPETITIVE LMFBR INDUSTRIAL ECONOMY
- ACHIEVEMENT OF OBJECTIVE REQUIRES SUCCESSFUL ACCOMPLISHMENT OF 3 PHASES:
  - 1. R & D PHASE - TO CONFIRM TECHNICAL ASPECTS OF CONCEPT
    - DEVELOP FUELS AND MATERIALS TECHNOLOGY
    - DESIGN CONTROL INSTRUMENTATION
    - DETERMINE PHYSICS AND HEAT TRANSFER DATA
    - PROVIDE BASIC ENGINEERING DATA FOR COMPONENT DESIGN
    - CONSTRUCT AND OPERATE FACILITIES TO PROVIDE DESIGN INFORMATION
  - 2. ENGINEERING & MANUFACTURING PHASE - TO PROVIDE BROAD INDUSTRIAL BASE
    - MANUFACTURE AND TEST FUEL, CLAD, CORE HARDWARE
    - DESIGN, BUILD AND TEST COMPONENTS AND SYSTEMS
    - BUILD AND TEST RELIABLE INSTRUMENTATION
    - CONDUCT OVERALL PROOFTESTING AND QUALITY ASSURANCE PROGRAMS
    - DEVELOP AND APPLY CODES AND STANDARDS
  - 3. UTILITY COMMITMENT PHASE - TO PURCHASE, BUILD & OPERATE COMMERCIAL LMFBR
    - TAKE FINANCIAL RISKS ASSOCIATED WITH DEMONSTRATION PLANTS AND FIRST-OF-A-KIND POWER PLANTS
      - UNCERTAIN COSTS
      - UNCERTAIN SCHEDULES
      - UNCERTAIN PLANT FACTORS
    - PURCHASE, MANAGE CONSTRUCTION, OPERATION, AND MAINTENANCE OF COMMERCIAL LMFBRS.

Figure 5

LMFBR PROGRAM

UTILITY ACCEPTANCE REQUIREMENTS FOR LMFBR PLANTS

- INCENTIVES MUST BE ATTRACTIVE
  - CHEAPER ELECTRICITY
  - MARKET FOR PLUTONIUM FROM WATER PLANTS
  - MINIMUM IMPACT ON ENVIRONMENT
- NATIONALLY SUPPORTED LMFBR PROGRAM
- TECHNOLOGY AND INDUSTRIAL RESOURCES MUST BE DEVELOPED IN ALL TECHNICAL AREAS
- PLANT DESIGN
  - COMPONENTS
  - INSTRUMENTATION & CONTROL
  - SODIUM TECHNOLOGY
  - CORE DESIGN
- FUELS & MATERIALS
  - FUEL HANDLING
  - FUEL RECYCLE
  - PHYSICS
  - SAFETY
- SOLUTIONS MUST BE AVAILABLE FOR ENGINEERING AND APPLICATION PROBLEMS AND FOR EXPEDIENT LICENSING APPROVALS
- SUPPORTING INDUSTRY SHOULD BE KNOWLEDGEABLE, COMPETITIVE AND SELF-SUSTAINING
- INVESTMENT OF RISK CAPITAL BY UTILITIES SHOULD BE AS SMALL AS PRACTICABLE
- LARGE PLANT CAPACITY WARRANTS HIGH PLANT AVAILABILITY
- CONTINUED PROGRESS AND SUCCESS IN FOREIGN LMFBR PROGRAMS

## VIII

CONCLUSIONS

Progress has been made in further defining the problems of the GCFR concept since the analysis reported in WASH-1089. While much of the privately-sponsored effort has been aimed at a 300 MWe demonstration plant, the overall full-scale plant system design has remained largely unchanged and the major developmental and safety issues specific to the GCFR as outlined herein require resolution. The further definition of these problem areas, aided by experience to date in the LWR, HTGR and LMFBR programs and discussions with the Regulatory staff and the ACRS, has served to reinforce the earlier conclusions that a substantial program of development and proof testing would be required to resolve adequately all of the major outstanding technical issues on this concept.

Based upon the information currently available, it is concluded that on a relative basis there have been no significant changes in the GCFR development status to justify appreciably altering the conclusion of WASH-1089 that considerable research and development effort would be needed to provide the required assurance of safe, reliable and economic operation at the design performance levels. A measured, orderly program similar to that outlined for the LMFBR including fuels and materials testing, component test facilities, critical experiments, a reactor experiment affording extensive in-pile fuel testing capability, safety tests, and a demonstration plant program would be required to bring the GCFR forward as an option in the commercial stage.

The GCFR concept is at an early stage of development with respect to its engineering base, reactor operating experience, industrial participation and commitment, and funding level. It does not have the benefit of considerable existing fast reactor experience both in the United States and abroad, such as enjoyed by the LMFBR. Assuming that solutions to the basic and major technological problems could be developed, achievement of the potential of the concept as a commercial power plant would require a greatly expanded Government effort with significant industrial participation and commitment.

If additional resources beyond those needed for the development and commercial introduction of the LMFBR were to be made available, they could be employed in a worthwhile manner to move toward a GCFR experiment and other key test facilities to help establish the technical basis prior to committing to conduct a parallel breeder development program on the GCFR. The timing and funding for such a program described in earlier reports would need to be reevaluated on a current basis, taking into account status and commitments on the HTGR, LWR and LMFBR programs.

In the absence of any additional resources, the GCFR technology will continue to benefit from the substantial efforts on the LMFBR and HTGR technologies, as well as the ongoing development work on the GCFR sponsored by the AEC and private industry. The cumulative direct expenditure on GCFR development by the AEC through FY 1972 has been approximately \$9.0 million, while private support for the concept from utilities over the four year period through

FY 1972 has been \$5.5 million. The option remains open over the next few years for increasing the degree of effort on the GCFR should national energy priorities and the degree of progress on the LWR, LMFBR and HTGR so dictate.

## IX

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