GENERAL DYNAMICS

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**MARITIME GAS-COOLED
REACTOR PROGRAM**

A REVIEW OF THE MARITIME GAS-COOLED
REACTOR PROGRAM

by

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THE MGCR PROGRAM AND ITS OBJECTIVES

The Need for an MGCR-type Power Plant

The MGCR Program, which is under way at General Atomic under the sponsorship of the U. S. Atomic Energy Commission and the Maritime Administration, has included studies of a nuclear power plant of 22, 000 to 32, 000 shaft horsepower for commercial maritime use. This shaft-horsepower requirement corresponds to a reactor power of 53 to 77 Mw(t) at a net cycle efficiency of approximately 32%, which is obtainable at a reactor outlet temperature of 1300°F. The plant is not limited to the above application but has broad capability for both mobile and stationary commercial nuclear power plants in the 15- to 100-Mw(e) range. Examples would be load-topping plants and basic power plants for use in remote areas where the power demand is small and fluctuating and where transportation costs are high.

The Objectives of the Power-plant Studies

In general, the MGCR Program staff has studied a power plant with

1. Low to intermediate power output,
2. High thermodynamic cycle temperature,
3. High thermal efficiency over the power range,
4. Rapid plant response to demands for power changes,
5. Minimum staff for operation and maintenance,
6. Low capital cost,
7. Competitive mills/kw-hr costs with plants of similar application,
8. Safe and reliable operation, and
9. Development potential.

Selection of a Closed-cycle, Gas-cooled System

The closed-cycle gas turbine directly coupled to a gas-cooled

reactor was selected primarily for the following reasons:

1. It has a high basic cycle efficiency. This will increase as higher-temperature gas becomes available from the reactor, since efficiency is proportional to the temperature difference across the system. The increase in cycle efficiency is predictable, since for gas-cooled reactors the outlet gas temperature and the temperature rise across the reactor have tended to increase over the years of their development.

2. It maintains its high basic cycle efficiency over a wide range of fractional powers. This is because the pressure ratio across the turbine is fairly constant, and the mass flow is reduced by reduction of gas density. To reduce power costs, the maintenance of high cycle efficiencies over wide ranges of fractional powers is important for a marine plant or for a stationary plant with a fluctuating power demand.

3. Evidence exists that the gas turbine, when applied to marine and stationary plants, will be a reliable piece of equipment which will respond well to rapid changes in load and long, unattended periods of operation.

BASIC MGCR PLANT DESCRIPTION

The basic flow path of the helium gas in the MGCR may be seen in Fig. 1, which is a simplified version of the heat balance for the 1500°F system. Helium enters the reactor at 883°F and 1127 psia, where its temperature is raised to 1500°F and its pressure reduced to 1087 psia. The gas enters the high-pressure power turbine and then exhausts into the low-pressure turbine, which drives the compressors. The gas enters the regenerator at 968°F and 451 psia, and after rejecting its heat to the reactor inlet gas, leaves the regenerator at 342°F and 446 psia. The gas passes through an auxiliary service heater and a precooler and enters the low-pressure compressor at 85°F and 442 psia. After intercooling, the gas passes through the high-pressure compressor and enters the other side of the regenerator at 236°F and 1144 psia. Leaving the regenerator at 883°F and 1127 psia, it flows into the inlet side of the reactor.

Fig. 1--Flow diagram for the MGCR

Power output from the reactor is controlled by varying the system pressure and, therefore, the mass flow through the core at a constant outlet gas temperature. Basically, the operator exercises control over the reactor power, reactor outlet temperature, turbine power output, and compressor speed.

MAJOR MGCR COMPONENTS OF INTEREST

The major components of interest in the MGCR primary system are the turbomachinery and the reactor. In the MGCR system now proposed, the high-pressure turbine is the propulsion turbine, and the low-pressure turbine drives the compressors. This permits a better balance of blade stresses by exposing the smaller high-pressure turbine to the higher temperature and the larger low-pressure turbine to the lower temperature.

For the helium turbomachinery^{(1)*} the optimum pressure ratio is low (2.6), and this, together with the low compressibility of helium, results in a small volume change as the fluid flows through the turbomachinery. The operating range is broader than would be the case with air or carbon dioxide, and efficiency does not fall off as rapidly during off-design-point operation. Mach numbers are low--approximately 0.4 in the compressor. The machine is within the state-of-the-art aerodynamically. A special problem of the helium turbine is the large number of stages required--approximately 40 in the compressors and 16 in the turbines. This requirement results in long, slender rotors with a high ratio of running speed to critical speed. Helium thus simplifies aerodynamic problems, but makes mechanical design more difficult. Figure 2 shows a section through the turbomachinery.

Gas enters the high-pressure turbine from the reactor and passes through the low-pressure turbine and to the regenerator. The low-pressure compressor receives gas from the precooler and, after intercooling, the gas passes through the high-pressure compressor and discharges to the regenerator. (The high-pressure turbine is mounted coaxially with the other components, but is not mechanically connected to them.)

*References are listed at the end of this report.

Fig. 2--Turbomachinery for the MGCR

The MGCR is shown in Fig. 3. The helium gas of the primary system enters through the outer annulus of the concentric duct and regeneratively cools the pressure vessel, thermal shields, and internally insulated core barrel, before passing through the beryllia-moderated core and leaving through the inner pipe of the concentric duct. The nozzles shown on the head are for the support and location of the control-rod-drive mechanisms.

PRESENT STATUS OF THE PROGRAM

In June, 1961, the Electric Boat Division of General Dynamics Corporation completed its research studies of the plant arrangement for the MGCR and other propulsion-system activities, including helium heat-transfer measurements and development of turbomachinery seals and plant-control studies.⁽²⁾

The helium turbomachinery for the MGCR has been under development by the Westinghouse Gas Turbine Division, and its conceptual design has been completed.⁽²⁾

The Experimental Beryllium Oxide Reactor (EBOR) has been designed by General Atomic and the preparation of specifications for the procurement of components is in progress. Kaiser Engineers is designing the test facility. Administrative direction of the program is under the San Francisco Operations Office of the Atomic Energy Commission.

THE EBOR PROGRAM

Since at this time the EBOR forms a part of the MGCR program, a review of its principal design features and development areas will be given below.

Principal Design Features

The choice of helium as a working fluid⁽¹⁾ was influenced by the desire to build development potential into the MGCR system. Carbon dioxide would be a good reactor coolant at 1300°F, but at higher temperatures

Fig. 3--The MGCR prototype reactor

it becomes sufficiently corrosive⁽³⁾ to rule out the possibility of using some of the refractory metals which hold promise as future cladding materials. In addition, helium will not dissociate or become chemically active⁽¹⁾ as the temperature is increased beyond the initial design turbine temperature of 1300°F, and the gas-turbine power cycle will permit advances in temperature to be used to obtain increased efficiency, while retaining its basic simplicity.

The choice of beryllium oxide as the moderator for MGCR systems has been influenced by the following considerations:

1. Since BeO has approximately twice the volumetric heat capacity of graphite,⁽³⁾ it provides a better heat sink to which uncooled, metallic-clad fuel elements may reject their heat--for instance, during a loss-of-coolant accident.
2. It permits the use of air as an emergency coolant, because of its resistance to oxidation at elevated temperatures.⁽⁴⁾
3. Beryllium oxide has attractive nuclear properties as a moderator.⁽³⁾
4. Its present high initial cost will be reduced during the term of the MGCR program.⁽³⁾
5. The effects of the reactor environment on the physical properties of BeO are now reasonably well understood, although significant further investigation into this field is still merited. A recent irradiation test of BeO samples by General Atomic demonstrated that adequate resistance to irradiation damage can be obtained through the proper choice of composition and fabrication technique.

The primary system of the EBOR facility comprises, in essence, the reactor, the regenerator, the main helium blower, and the heat dump. The relationship of these components can be seen in Fig. 4.

The helium gas passes from the main helium blower, through the regenerator, and enters the reactor through the outer annulus of the

Fig. 4--Flow diagram for the EBOR

concentric duct. The gas leaves the reactor through the inner pipe of the concentric duct and passes through the other side of the regenerator on its way to the helium-to-air heat exchanger. From this point, the helium gas returns to the main blower. Figure 4 also shows the decay heat removal and emergency air-cooling systems.

The reactor is shown in more detail in Fig. 5. However, the external bypass pipe shown in this diagram has been replaced by a configuration in which the gas flow splits on entry to the vessel flow annulus. One stream flows upward between the pressure vessel and the thermal shield and downward between the internally insulated core barrel and the thermal shield. The other inlet stream flows downward between the pressure vessel and the thermal shield. The streams join below the core and flow upward through the core support risers and the core, and into the upper plenum before leaving the reactor.

The construction materials for the pressure vessel and internal structure have design temperatures of 850°F and 1000°F , respectively, which correspond to inlet gas temperatures of 750°F and 883°F . These inlet gas temperatures correspond to outlet gas temperatures of 1300°F and 1500°F , respectively. The system pressure for the 1300°F outlet gas temperature is 1120 psia and the design pressure is 1250 psia.

The material selected is SA-387B, which is a low-alloy steel of 1 Cr-1/2 Mo composition. SA-302B, which is a Mn-1/2 Mo composition steel, and SA-212B, which is a C-Si composition steel, are more commonly used in reactor pressure vessels up to a design temperature of 650°F . The SA-302B steel was rejected because Section I of the ASME Code does not recommend its use over 850°F owing to possible transformation of the carbon phase into graphite. SA-212B was rejected because of its low strength at high temperature.

Data on radiation damage are available on SA-336, which is a Cr-Mo composition forging steel. An extrapolation of the data⁽⁵⁾ suggests that the increase in nil-ductility transition strength and the loss of impact strength

Fig. 5--Vessel and internals of the EBOR

are small at the EBOR operating temperature conditions for the integrated neutron flux of interest, 5×10^{18} fast (>1.0 Mev). Nevertheless, it is proposed to insert surveillance samples of construction materials having representative fabrication history into the EBOR system. Samples will be taken of the weld, the heat-affected zone, and the parent metal. Charpy samples will be cut with different axes of orientation and from the 1/4-point location in the parent metal. It is proposed to plot the impact strength versus the test temperature with integrated neutron flux (fast) as the parameter. Initial values with the integrated neutron flux (fast) equal to zero will be taken as part of the vessel quality-control program. In addition, the relationship of the operating pressure to the temperature of the materials of construction will be established to guide the plant operators. The operating cycles for the plant will be reviewed, and a comparison will be made of the operating spectrum with the cumulative damage factor of the fatigue analysis. This is essential for a reactor experiment in which it is not possible to define precisely the future sequence of experimental conditions.

Seals are classified as "seal welded" or "seal weldable." An instance of the former would be the primary flange closure. A seal weld is provided here, since confidence in the leaktightness of a shipboard plant must be extremely high, owing to the small exclusion area and to the uncertainty of the performance of mechanical seals of this diameter in a helium environment. Mechanical seals have been added at the primary closure to develop data for this case. An instance of "seal weldable" would be the refuelling-port closures. Here mechanical seals are provided to permit easier access to the core for the land-based application, and provision is made to weld the seal if gross leakage should occur through the mechanical seals to the atmosphere.

The concentric duct was selected, rather than a single duct,

1. To permit the high-pressure pipe to be kept at a low enough temperature that low-alloy steels could be used without a

prohibitive amount of internal insulation being required.

2. To provide for a compact machinery layout for the MGCR by combining the duct with a movable reactor, and
3. To reduce heat losses to the compartment by providing external insulation on the outer duct.

The control-rod drives are mounted on the top of the pressure vessel to provide for control-rod insertion from the hot end of the core. Control-rod insertion from the hot end causes the maximum power generation to occur at the lower-temperature end of the core. This results in a cladding temperature approximately 200°F lower than insertion from the cold end.

The control-rod-drive mechanism shown in Fig. 6 comprises, essentially, a normal positioning system and a scram system. The normal positioning system consists of an electric motor that drives through a gear box and a clutch to rotate a vertical lead screw with axial constraint. The lead screw drives through a nut into a drive shaft with rotational constraint and an axial degree of freedom. The drive shaft is connected to the control element through a mechanism which provides for the remote decoupling of the control element. In the scram system, the electric motor and gear box are replaced with a pneumatic motor that is controlled by a scram valve and a modulating valve.

The core comprises 36 core elements, 52 reflector elements, and 4 control elements. The core for the EBOR takes its shape from the fact that this core is really the center 36 fuel channels of the 308-fuel-channel MGCR core. This feature provides for representative heat fluxes, temperatures, and mechanical loadings in the core structure.

A core element is shown in Fig. 7. The essential parts of the core element are the fuel-pin array, the square-annular BeO blocks, and the shroud tube, which defines the flow passage and provides axial structural continuity for the core element and transverse support for the beryllia blocks. The fuel pins are Hastelloy-X tubes that contain UO_2 -BeO fuel compacts.

Fig. 6--Control-rod drive for the MGCR

Fig. 7--The 18-fuel-pin core element

The reflector element is dimensionally similar to a core element; its essential difference is the replacement of the fuel configuration with annular BeO rings.

The cruciform control elements are of dysprosium oxide distributed in alumina tiles that are contained in a Hastelloy-X structure.

Development Areas of Interest

Predicting the thermal performance of the reactor requires a knowledge of the geometry of the flow passages (defined by the heat-transfer and friction surfaces), the distribution and velocities of the coolant, and the properties of the coolant. Experimental programs are in progress to determine the local and average heat-transfer coefficients and the friction factors as a function of Reynolds number for fuel configurations of interest.

The fuel cladding will experience thermal and mechanical loading over a complex spectrum of load directions and magnitudes, loading rates, and number of loading cycles. The properties of the cladding will be affected by the environment, especially by the irradiation. Experimental programs are proceeding to investigate the properties of the materials, such as creep and stress rupture, and to expose the complete fuel pins to typical cycles of thermal and mechanical loading out-of-pile.

As part of the fabrication process, the fuel cladding is creep-shrunk onto the fuel compact by a combination of gas pressure and temperature. This process introduces residual tensile stresses in the cladding which, by relaxation over the reactor cold-to-hot cycle, reduce the separation between the cladding and the fuel compact, and thus reduce the operating stress in the cladding during steady-state operation. Further experimental work is being done to develop a residual tensile stress in the cladding as near to the yield point as possible for the as-installed condition.

In addition, a complete assembly of fuel pins is functioning in an in-pile experiment at Hanford. The fuel assembly has operated 2841 hr

at 1500°F cladding temperature, 70 hr at 1550°F, 136 hr at 1600°F, 90 hr at 1650°F, and 285 hr at 1700°F. Over the test period the element has been thermally cycled 30 times. At present the fuel assembly is still functioning quite satisfactorily in that the loop activity remains at normal background level.

The fuel compacts being developed are a mixture of UO_2 in a BeO matrix. This design results in significantly lower fuel-body temperatures, because of the higher thermal conductivity associated with BeO. The lower temperature results in a significant reduction of the release of fission products from the fuel compact. The UO_2 -BeO fuel compacts exhibit good dimensional stability and retain about two thirds of the original crushing strength after irradiation to 55,000 Mwd/T of uranium. The experimental programs in progress are intended to develop fuel compacts with improved fission-product retention, dimensional stability, and strength.

The moderator and reflector BeO experiences internal heat generation, and in rejecting the heat to the fuel-channel coolant, temperature gradients are developed that cause thermal stresses. There are a number of factors to be considered in predicting the maximum power to which the BeO should be exposed for reactor operation; these factors may be summarized as follows:

1. The effect of grain size, density, and composition on the physical and mechanical properties.
2. The effect of the reactor environment, with special reference to temperature and irradiation, on the original physical and mechanical properties of the materials. Of particular interest is the effect on the BeO of the superimposed mechanical loading resulting from the generation of helium gas in the $\text{Be}(n, \alpha)$ and $\text{Be}(n, 2n)$ reactions in the BeO.
3. The effect of the size and shape of the BeO on the stress distribution within the blocks for given conditions of power and surface cooling.

4. The effect of the rate of application, magnitude, and number of cycles of thermal loading on the BeO blocks.
5. The definition of some criteria that relate the distribution of stresses in the ceramic body under load to the probability of rupture, and the correction to be applied to this relationship to factor in the loading rate and the number of cycles.
6. A review of the available methods of quality control, in order to select procedures by which the characteristic differences in a production run of BeO may be identified.
7. A review of processes and process parameters in the fabrication of BeO to select the criteria which result in the highest resistance to thermal stress, the highest density consistent with the gas generation problem, the minimum dimensional change due to irradiation damage, and the highest purity consistent with reasonable cost.

The control element will experience mechanical loads induced by the drive system and thermal loads that occur as a result of the temperature distribution within the cruciform control element. In addition, since the space provided to ensure free insertion and removal of the control element from the core is void volume, the incentive is high to reduce the required working clearances between the control element and the core. This fact makes the geometric and dimensional stability of the cruciform-section control element of interest. Experimental work is in progress to study these problems.

The cost of a reactor core may be significantly reduced if performance requirements and fabrication complexity are balanced against one another. The practical difficulties of fabricating and inspecting numerous detailed parts and the assembly of these parts into a component need to be understood. This consideration has led to emphasis on full-scale mockups of core components, using the reactor construction materials and fabrication procedures. Where tolerances become asymptotic with cost, the effect of

tolerance relaxation on performance is reviewed. Performance of the core is reviewed using a statistical accumulation of tolerances established by mockup experience.

For the complex core elements the effects of flow-induced vibrations are not suitable for analysis. In particular, the stiffness of the fuel pin is strongly temperature dependent, owing to the residual tensile stress in the cladding. To investigate the effect of vibrations, the mockups previously referred to will be instrumented and an experimental investigation of the vibration effects will be made.

Although the major areas of development are associated with the core components, a significant investigation of the control-rod-drive mechanism has been made. Extensive performance tests of component parts have been made, including the compatibility of materials of construction under load in a helium environment. At this time, tests of endurance and performance characteristics are being conducted in a test loop. The loop is designed to expose the mechanism to the reactor system helium at design temperatures, pressures, helium purity, and mechanical loadings. To ensure safe and reliable operation, a mockup of a control-element channel is provided in the autoclave, and a mockup control element with a design model disconnect is used in the test program. All production control-drive mechanisms will be performance-tested in the loop before installation in the reactor. This testing is essential, since the control-rod-drive mechanism cannot be considered a developmental item in an operating reactor system.

In summary, to review a program as broad and technically rewarding in professional interest as the MGCR is difficult. The discussion presented here, however, has had the essential objective of explaining, in broad terms, what General Atomic is doing on the MGCR program in various areas of engineering.

Many other interesting areas of development exist in disciplines not

mentioned here, such as the physics of beryllium oxide reactors and fundamental ceramics and metallurgy. Although these programs are not considered in this paper, they are undoubtedly of great interest and value to the successful completion of the project.

ACKNOWLEDGMENTS

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Appendix

EBOR PRELIMINARY DATA

REACTOR CORE AND REFLECTOR

Power rating	10 Mw(t)
Average neutron flux:	
Fast	3.1×10^{13} (>0.2 Mev)
Thermal	4.8×10^{12} (<0.68 ev)
Power peaking factors:	
Axial	2.10
Radial	1.120
Excess reactivity (δk) to compensate for	
Temperature changes	4%
Xe and Sm at rated power	2%
Burnup	3%
Maximum excess reactivity with dry core	9%
Fuel:	
UO ₂ in BeO	22 vol-%
Uranium enrichment	70%
Pellet diameter	0.33 in.
Pellet length	0.43 in.
Pellets per fuel pin	178
Cladding	0.020 in. Hastelloy-X tubing, 0.335 in. I. D. by 80 in. long
Core dimensions (active core)	23.43 in. square by 76 in. long
Reflector	7 in. of BeO
Number of fuel elements	36
Lattice	Square, 3.64 in. pitch
Core loading	100 kg U ²³⁵
Average power density in core	14.4 kw/liter = 1.40×10^6 Btu/(hr)(ft ³)
Total BeO in active core	2,350 lb

Total BeO in reflector	5,279 lb
Heat flux at fuel-element surface:	
Average	5.8 cal/cm ² -sec = 77,100 Btu/(hr)(ft ²)
Maximum	15.0 cal/cm ² -sec = 200,000 Btu/(hr)(ft ²)
Film-temperature drop, design	43°C = 109°F
Fuel-element temperatures, design	
maximum:	
Fuel, nominal value without hot- channel factor	910°C = 1670°F
Cladding, nominal value without hot- channel factor	730°C = 1350°F
Coolant flow area:	
Each fuel element	7.5 cm ² = 1.165 in. ²
Total	270 cm ² = 41.94 in. ²
Channel velocity of coolant:	
Minimum	50 m/sec = 166 ft/sec
Maximum	75 m/sec = 246 ft/sec
Heat-transfer coefficient	0.134 cal/(cm ²)(sec)(°C) = 989 Btu/(hr)(ft ²)(°F)
Coolant mass flow rate	6.3 kg/sec = 13.88 lb/sec
Coolant temperature:	
Inlet	400°C = 750°F
Outlet	700°C = 1300°F
Hot-channel factors:	
*	
Temperature rise of coolant	1.152
Temperature drop of film	1.20
Over-all temperature drop for cladding and fuel	1.594

CONTROL

Shim-safety rods (cruciform):

Number	4
Active length	76 in.
Worth of shim-safety rods	25% δk
Shim-safety rod speed:	
Insertion or withdrawal	0.15 cm/sec
Insertion only	0.30 cm/sec

*Factors are for the lowest-powered channel, which gives the maximum cladding temperature.

Regulating rods (shim-safety rods used
for regulation):

Worth of regulating rods 25% δk

Regulating-rod speed:

Insertion or withdrawal 0.15 cm/sec

Insertion only 0.30 cm/sec

Maximum reactivity insertion rate $1.5 \times 10^{-4} \delta k/k/sec$

Scram time and type of mechanism:

50% insertion 0.7 sec

100% insertion 2.0 sec

High-speed, pneumatic, motor-driven
lead screw, positive scram blade
insertion

Temperature coefficients, $\delta k/k/^{\circ}C$:

Cold:

Prompt -0.54×10^{-4}

Delayed $+0.14 \times 10^{-4}$

Total -0.40×10^{-4}

Hot:

Prompt -0.25×10^{-4}

Delayed -0.02×10^{-4}

Total -0.27×10^{-4}

Hot equilibrium poison:

Prompt -0.25×10^{-4}

Delayed $+0.10 \times 10^{-4}$

Total -0.15×10^{-4}

REACTOR VESSEL

Configuration Cylindrical shell, hemispherical
heads

Material SA-387B

Shell wall thickness 5-1/4 in.

Vessel pressure:

Working pressure 1120 psia

Design pressure 1250 psia

Test pressure 1875 psia

Over-all reactor dimensions:

Over-all height of reactor, including
control-rod drives 34 ft

Outside diameter of pressure vessel
over primary flange 13 ft

Containment type and material No containment vessel; reactor
is located in a vault that has
concrete walls and bottom
and that is covered with
removable concrete slabs.

REFERENCES

1. de Hoffmann, F., and W. T. Furgerson, MGCR As a Small Central-station Power Plant, General Atomic, Report GA-1516, July, 1960.
2. Moore, W. C., Experimental Bases for the Design of the EBOR, General Atomic, Report GA-2477, August, 1961.
3. Background Information for Selection of Beryllia as Moderator for the Maritime Gas-cooled Reactor, General Atomic, Report GA-1280, April 1, 1960.
4. Moore, W. C., The Experimental Beryllium Oxide Reactor, General Atomic, Report GA-2385, July, 1961.
5. Hawthorne, J. R., and L. E. Steele, Preliminary Observations on the Effectiveness of Heat Treatment for the Recovery of Properties of Irradiated Steels, U. S. Naval Research Laboratory, Report NRL-5582, February, 1961.

ERRATUM

GA-2603, A REVIEW OF THE MARITIME GAS-COOLED
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The following erratum should be noted in General Atomic Report
GA-2603:

Page 17, line 12 --

Change 10^{20} nvt fast (>1 Mev) to read 55,000 Mw-days per ton of
uranium.

corrected 1/27/62

JCR