JUL 23 1962

MASTER

GENERAL ATOMIC

DIVISION OF **GENERAL DYNAMICS**

GA-2944

THERMAL-DESIGN ASPECTS OF GAS-COOLED REACTORS

by

Robert Katz and Marius Troost

U. S. Atomic Energy Commission
Contract AT(04-3)-187
and
Contract AT(04-3)-314

June 13, 1962

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

GENERAL ATOMIC
DIVISION OF
GENERAL DYNAMICS

JOHN JAY HOPKINS LABORATORY FOR PURE AND APPLIED SCIENCE

P.O. BOX 608, SAN DIEGO 12, CALIFORNIA

GA-2944

THERMAL-DESIGN ASPECTS OF GAS-COOLED REACTORS

by

Robert Katz and Marius Troost

This paper was presented at the Spring Meeting of the Engineering Joint Council Nuclear Congress in New York and will be published in the Transactions of the Meeting.

U.S. Atomic Energy Commission
Contract AT(04-3)-187
and
Contract AT(04-3)-314

June 13, 1962

ABSTRACT

A discussion is presented of the thermal design of two gas-cooled reactors--the High-temperature Gas-cooled Reactor (HTGR) and the Experimental Beryllium Oxide Reactor (EBOR). Since the thermal-design aspects of these two systems are different, each system is treated individually. The basic temperature criteria of each system are identified, and the influence of heat-transfer coefficients, hot spots, and transient analysis on the temperature criteria is indicated.

The HTGR program is focused on the development of large (100- to 500-Mw(e)) stationary power plants. The core construction is homogeneous--graphite is used exclusively. The helium coolant outlet temperature is about 1380°F, and a conventional steam cycle is used to extract the power.

The EBOR program is focused on the development of compact power plants in the 20- to 100-Mw(e) class. The core construction is heterogeneous--beryllium oxide is used as the moderator. The reactor is coupled directly to a closed-cycle gas turbine, with the helium coolant in the reactor also serving as the working fluid in the system. The initial design coolant outlet temperature is 1300°F.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

CONTENTS

ABSTRACT	i
INTRODUCTION	1
THE HTGR SYSTEM	3
Description of Reactor Core and Fuel Elements	3
Steady-state Thermal Analysis	5
Heat-transfer and Pressure-drop Correlations	5
Analytical Methods	6
Hot-spot Allowances	6
Transient Response	8
DESCRIPTION OF THE EBOR PROGRAM	11
Basis for Core-element Design	12
Over-all Thermal Design of the Core	15
Detailed Thermal Aspects of the Fuel Element	18
FUTURE DEVELOPMENTS IN HEAT TRANSFER FOR GAS-COOLED REACTORS	23
REFERENCES	25

INTRODUCTION

Gas-cooled reactors are characterized by high coolant outlet temperatures and correspondingly high thermodynamic efficiencies without high system pressures. The selection of inert gas as coolant eliminates corrosion problems and, therefore, simplifies the selection of reactor materials. These features assure the great potential of gas-cooled reactors. At the General Atomic Division of General Dynamics Corporation, work is presently progressing on two different gas-cooled-reactor concepts: the High-temperature Gas-cooled Reactor (HTGR) and the Experimental Beryllium Oxide Reactor (EBOR). In this report, the thermal-design aspects of the two systems are discussed. Since the thermal-design problems are basically different for these two reactors, each system is treated separately.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

THE HTGR SYSTEM

A prototype of the HTGR type of nuclear reactor has been designed by General Atomic and is currently under construction at Peach Bottom, Pennsylvania. The discussion which follows pertains to this prototype.

The prototype HTGR system⁽¹⁾⁽²⁾ consists of a nuclear reactor containing graphite fuel elements, two blowers to circulate the helium coolant, and two steam generators. The reactor system is designed to deliver 116.6 Mw(t) or 40 Mw(e), resulting in a net efficiency of 34.8%. The coolant has a flow rate of 458,250 lb/hr at 350 psia, an inlet temperature of 600°F, and an average outlet temperature of 1287°F. The steam conditions are 1005°F at 1551 psia at the turbine throttle.

One of the primary design aims of the HTGR effort has been the achievement of low cost. The following features of the HTGR contribute toward this goal:

1. High coolant outlet temperatures permit the use of compact boilers and modern steam conditions and result in a good thermal efficiency.
2. The use of an inert coolant permits extensive use of carbon steel for the reactor vessel and internals.
3. The all-graphite core construction permits high coolant temperatures and good neutron economy, which lower fuel costs.

DESCRIPTION OF REACTOR CORE AND FUEL ELEMENTS

The reactor core consists of about eight hundred cylindrical fuel elements, each 3.50 in. in diameter and 12 ft long (see Fig. 1). The elements form a closely packed hexagonal pattern, with the 3.50-in. elements on a 3.55-in. pitch. The elements are kept apart by four ring-shaped spacers around each element. Each element is surrounded by six tricuspid cooling passages. The bottom 2.5 ft of each fuel element forms part of the bottom reflector and contains a fission-product trap, the center 7.5 ft contains the fuel, and the top 2 ft forms part of the top reflector. The fuel is present as fuel rings, consisting of graphite mixed with small (200- μ -diameter) particles of uranium and thorium carbide. The size of the particles was chosen such that the graphite matrix suffers little damage from the recoil of fission fragments but still retains the favorable nuclear characteristics of a mixed fuel-moderator. Under steady-state conditions,

Fig. 1--HTGR fuel element

these small particles are at essentially the same temperature as the moderator.

A fuel ring was chosen in preference to a solid central fuel cylinder of the same diameter in order to limit peak temperatures of the fuel. The thickness of the ring is convenient for fabrication reasons. The sleeve is provided to keep most fission products out of the coolant stream and to give structural rigidity to the fuel element. The purge grooves in the fuel body and the gap between the fuel body and the sleeve carry a helium flow designed to sweep fission products toward fission-product traps in the bottom reflector and outside the reactor vessel.

The fuel-element diameter of 3.50 in. was determined from the maximum allowable fuel temperature, which was not determined by a sharp temperature limitation (e. g., melting point or phase change), but by fission-product release from the fuel grains, which increases appreciably between 2700°F and 3000°F. It is important to point out that a small local hot spot is thus not very serious, as a somewhat larger than normal fission-product release can be accepted in small areas of the core. Therefore, it is more important to know what percentage of the fuel is above certain temperatures than to prevent any fuel at all from reaching a certain temperature.

STEADY-STATE THERMAL ANALYSIS

The main aim of thermal analysis is to obtain desired information as accurately and cheaply as possible. Although it is in principle possible to obtain almost all the desired results by experimental means, it is in general much cheaper to obtain answers by calculation from basic data. In the HTGR program, only those basic data which were in doubt were obtained experimentally--namely, the circumferential variation of the heat-transfer coefficient around a fuel element⁽³⁾ and the conductivity of graphite. Some other uncertainties, such as the flow distribution in the reactor core and the effects of bending of fuel elements, were investigated analytically.

Heat-transfer and Pressure-drop Correlations

Experimental work⁽³⁾ has shown that the "standard" (smooth pipe) heat-transfer correlation employing the hydraulic diameter of the tricuspid-shaped channels around HTGR fuel elements yields answers about 5% higher. To obtain conservative design results and to account for uncertainties in the helium properties, the heat-transfer correlation used was the "standard" correlation with a 15% reduction factor.

The same experimental work has shown that the HTGR friction factor also lies slightly below the "standard" correlation. To obtain conservative results, the "standard" correlation was used.

Analytical Methods

Temperature and pressure-drop calculations for the HTGR are complicated because the HTGR employs an open core, i.e., all cooling channels are interconnected. The local and total power inputs into the coolant vary from channel to channel; therefore, the resistance to flow varies also, owing to the differences in acceleration and viscosity of the coolant. The resulting pressure differences between channels are equalized by flow from one channel to another, i.e., cross-flow. On first consideration it is not obvious whether cross-flow is beneficial or disadvantageous as compared with closed-channel flow. Cross-flow is disadvantageous in that the hottest channel loses gas owing to the higher accelerational and frictional pressure drops. On the other hand, cross-flow is advantageous in that it allows more gas to enter at the beginning of the hottest channel than would be possible with a closed channel.

In the case of the HTGR, it has been shown that these two effects approximately cancel each other, with the result that there is very little difference between the maximum temperature calculated for an assumed open core without any resistance to cross-flow and that calculated for a closed core. It is therefore concluded that cross-flow is not likely to influence the hot-spot temperature, and consequently no uncertainty factor for flow distribution has been applied in the hot-spot calculations. However, in all calculations cross-flow has been assumed because it describes the real situation best and because it requires less computer time.

A typical temperature distribution, showing the fraction of the fuel above a given temperature, is plotted in Fig. 2. Similar curves were obtained for other components of the core.

Hot-spot Allowances

No hot-spot allowance was made for uncertainties in the heat-transfer correlation because a conservative correlation was used. Some other factors which may cause local hot spots are bending of the fuel elements, changes in the gap between the fuel and the sleeve, and local deviations from the predicted power. These factors and their influence on the highest fuel temperature are discussed below.

Bowing. A fuel element in the core can bow because of heat-generation gradients across its diameter or because of some built-in initial bow. The convex side of the element will tend to be hotter than the concave side, which will make the bow worse (because of additional differential thermal expansion). This will continue until the bowing fuel element comes to rest against adjacent elements. Because of spacer pads, a bowing fuel element can only deflect a maximum of about 0.015 in.

Fig. 2--HTGR fuel temperatures

For the hot-spot analysis, the conservative assumption was made that three adjacent fuel elements bow inward toward the channel center, producing a maximum channel restriction at the elevation where the maximum fuel-element temperature occurs. To simplify the calculations, it was assumed that the channel area decreases linearly between the ends of the rod and the hot spot. On this basis, a temperature rise due to bowing of 104°F has been calculated

Gaps and Eccentricity. All calculations were based on an "effective thermal gap" between the compact and the sleeve. An attempt was made to compensate with this effective gap for the existence of purge grooves and for the change in gap width that occurs with changes in fuel-element temperatures.

True gap width in a particular fuel element depends on diametral tolerances. A gap variation of ± 0.001 in. from the nominal value will cause a fuel-compact temperature variation of about $\pm 45^{\circ}\text{F}$ at the point of maximum power. The effect is smaller than one might expect because of the large amount of radiant heat transfer, which does not depend on gap width. Fuel-element tolerances are such that the gap could be 0.0025 in. above nominal. The corresponding increase in hot-spot temperature is 75°F .

Possible eccentricity between compact and sleeve can cause an additional temperature rise in the compact. However, the effect is small--less than 20°F .

Power Deviations. It was assumed that the power generation at the hot spot was 5% higher than the value predicted from the flux distribution. This margin was included to allow for variation in fuel loading (expected to be on the order of $\pm 3\%$) and other difficult-to-assess factors, such as variations from compact to compact in ability to retain fission products.

Transient Response

The change that occurs in coolant outlet temperature in the HTGR owing to sudden changes in power, coolant inlet temperature, or coolant flow rate is quite slow in comparison with the corresponding coolant-temperature change in other reactor systems. In the case of a sudden increase in power, the fuel ring, the spine, and, finally, the sleeve must be heated before any appreciable extra amount of heat can be transferred to the coolant. In the case of a sudden change in coolant inlet temperature, the surface temperature and, therefore, the sleeve temperature must be altered considerably before an effective change in outlet temperature becomes obvious. In the case of a sudden change in flow rate, the outlet temperature changes slowly because, again, the surface and bulk sleeve temperatures

must change considerably. The relatively high heat capacity of graphite (about 0.45 Btu/lb at the temperatures involved) thus serves as an effective brake on any transient behavior.

Some typical response curves are presented in Fig. 3. In each case, the ultimate change in outlet temperature is the same. All changes were step changes at time zero from steady-state operation. It is clear that the HTGR responds quite slowly to changes in operating conditions, which permits relatively simple controls to be used with the reactor.

Fig. 3--Transient response of HTGR coolant outlet temperature

DESCRIPTION OF THE EBOR PROGRAM

The original goal of the Experimental Beryllium Oxide Reactor program (formerly the Maritime Gas-cooled Reactor program) was the development of a compact power plant for merchant-ship application which directly couples a gas-cooled reactor into a closed-cycle, gas-turbine system. However, this objective has recently been expanded to include the development of small central-station power plants employing the same design features. Emphasis is currently being placed on the design and construction of the Experimental Beryllium Oxide Reactor (EBOR) to test the reactor concept. The EBOR has been approved for construction at the National Reactor Testing Station.

The objectives of the EBOR program are the attainment of high thermodynamic efficiency, simplicity of design with the attendant ease of operation, low maintenance costs, and maximum efficiency of operation over a wide range of power settings. One power-plant application studied⁽⁴⁾⁽⁵⁾ was designed to produce 32,000 shp, or 23 Mw(e), with the beryllium oxide moderated core designed for 74.4 Mw(t). The helium coolant outlet temperature was 1300°F. The thermodynamic efficiency of the gas-turbine cycle employed was 31.8%, which remained essentially constant over a range of settings from about 20% to 100% of maximum power. This was accomplished by varying the helium inventory linearly with the reactor power. As a result, the coolant mass flow changed with the power but the coolant volume flow rate, the cycle temperatures, and the operating point of the turbomachinery remained constant.

The design features of the gas-turbine power plant give this system particular advantages for the production of electrical power in the 20- to 100-Mw(e) range. Among the applications proposed⁽⁶⁾ are load-topping of large central power stations and a remote, self-sufficient power plant.

The EBOR is based on the concept of testing the fuel-element and beryllia-moderator components for the power plants mentioned above.⁽⁷⁾⁽⁸⁾ The EBOR core, essentially the center 36 elements of the full-size core, produces 10-Mw(t) power and will be installed in a pressure vessel designed for possible future upgrading. Upgrading will consist of building the core up to full size and coupling turbomachinery into the system--subject, of course, to authorization of this part of the program.

The reactor temperatures and pressures for the EBOR are the same as for the full-size (74.4-Mw(t)) reactor, and the power distribution

and maximum heat flux of $210,000 \text{ Btu}/(\text{ft}^2) (\text{hr})$ are representative of conditions to be met in the full-size power plant. The EBOR flow diagram is shown in Fig. 4, where it can be seen that power produced by the reactor will be dissipated to the atmosphere. The design coolant outlet temperature of 1300°F indicated in the figure is also the design value for the full-size reactor. Since it is expected that there is a future potential for coolant outlet temperatures greater than 1300°F , the EBOR test program will demonstrate the capabilities of the fuel element up to 1500°F .

BASIS FOR CORE-ELEMENT DESIGN

A complete discussion of the thermal design of a complete power plant would encompass a number of important components. However, the reactor fuel element has the highest temperature of any component in the complete system and thus is the limiting component from the standpoint of thermal design. The discussion of the thermal design is therefore limited to the fuel element.

In the EBOR, the fuel and moderator are designed as an integral unit which is termed a core element. The core-element design for the EBOR is shown in Fig. 5. Eighteen fueled pins are arranged in a ring around a center BeO spine. The core is based on a square-lattice arrangement, and consequently the outer BeO blocks have a square cross section. The fuel pins are $3/8$ -in.-diameter Hastelloy tubes containing fuel pellets composed of UO_2 -BeO compacts. The fueled length is 76 in.

The coolant channel is an annular passage formed by the inner BeO spine and a metallic shroud around the fuel pins. The misalignment of the ring of fuel pins within the annular passage is limited by means of helical spacers attached to each pin. The ratio of spacer pitch to fuel-pin diameter is 20:1. A parameter of importance in fuel elements composed of clusters of rods is the ratio of the minimum clearance (c) to the fuel-rod diameter (d). For the ring-type element, this parameter is $c/d \cong 0.14$ between adjacent fuel pins and $c/d \cong 0.07$ between a fuel pin and either the spine or the shroud.

The selection of a fuel-element design is based on balancing the requirements of the thermal and nuclear performance against mechanical design and economic considerations. A basic requirement for the EBOR was the retention of fission products, which has led to the usual design of metallic tubes loaded with cylindrical fuel pellets. The nuclear performance for low enrichment usually leads to fuel lumping, which consequently suggests fuel elements composed of rod clusters. The geometrical arrangement of the rods, the diameter of the rods, the clearances within the geometrical assembly, and the total number of fueled assemblies still remain to be determined. After a particular geometry is selected, parameter studies are conducted to determine the dimensions of the element and number of

Fig. 4--EBOR flow diagram

Fig. 5--EBOR 18-fuel-pin core element

elements required. Some of the considerations in selecting the final EBOR design were (1) the simplicity of the element, (2) the number of fuel elements and the economic factors of fabrication, as well as the economic factors of refueling time, (3) the prediction of the thermal performance to a high degree of confidence, and (4) the temperature, and consequently the structural behavior, of the element in the event of a loss-of-coolant accident.

The final selection of the ring-type element was based, in part, on the inherent simplicity of this element, the judgement that its thermal performance was more readily predictable than that of other elements considered, and its superior behavior in the event of a loss-of-coolant accident. This last factor results from the fact that each fuel rod directly "sees" the inner spine and outer BeO block, both of which act as temperature sinks for the decay heat, whereas in other designs such as rod clusters an inner ring of fuel rods is shielded from the temperature sink by an outer ring of rods. The temperature rise of the fuel cladding is less rapid in the ring-type element than in rod-cluster elements, and a longer time is available for initiating emergency cooling procedures.

OVER-ALL THERMAL DESIGN OF THE CORE

Although it is not possible to completely divorce the reactor thermal design from the over-all power plant, this can be done to a certain extent. The fundamental problem of the reactor thermal design can be considered to be attaining the thermodynamic cycle requirements while limiting the maximum fuel-cladding temperature to a prescribed value. The maximum fuel-cladding temperature is thus the critical temperature of the reactor and is fixed by material strength properties, with thermal and mechanical loads being taken into account. In addition to the critical temperature, the core pressure drop is of major importance owing to the large power absorbed in circulating the coolant. In a closed-cycle system the core pressure drop is doubly important because of the necessity of matching the reactor performance with that of the turbomachinery. In effect, the maximum cladding temperature and core pressure drop determine an envelope within which the fuel element must be designed.

After selection of the 18-pin, ring-type element design, the number of elements and the fuel-pin diameter were chosen. The only remaining quantity to determine in order to completely specify the design was the clearance between the fuel pins. The ratio of the minimum clearance to the fuel-pin diameter, c/d , is used as the basic parameter, and a small change of this parameter has a significant effect on both the core pressure drop and the maximum fuel-cladding temperature. The design of the EBOR core is proceeding on the basis of a reference value for the c/d parameter,

which will be adjusted, if necessary, after completion of heat-transfer experiments now in progress.

Orificing the coolant flow to the individual fuel elements is necessary to ensure that each element operates near its maximum potential as dictated by the maximum-temperature criterion for the fuel cladding. Orificing for equal coolant outlet temperatures from each element is presently specified for the EBOR. Orifice schedules based on obtaining equal coolant outlet temperatures and equal maximum cladding temperatures have been investigated. With the power distributions calculated for this reactor, orificing for constant coolant outlet temperature gives maximum cladding temperatures within $\pm 15^{\circ}\text{F}$ of the value obtained by orificing for a constant maximum cladding temperature. The difference in the core pressure drops with the two orificing schedules is negligible.

With each fuel element orificed for a specified coolant flow rate, the temperatures within any element may be determined on an individual basis, except insofar as variations in fuel-element and orifice geometry (due to manufacturing tolerances) affect the flow distribution. Variations in orificing and fuel-element geometry, as well as uncertainties in the flux distribution, heat-transfer coefficient correlation, etc., usually determine so-called hot-channel or hot-spot factors. The procedure used for determining the fuel-element hot spot was to determine first the nominal or ideal performance and then to consider the effect of each uncertainty individually and in combination with other uncertainties. This procedure is applied to the reactor as a whole in such a way that the various uncertainties will combine to give the lowest coolant mass flow rate and the highest value of the fuel-cladding temperature in one of the elements ("hot" element) under consideration.

The results of the flow and power uncertainty analysis for the EBOR are summarized in Table 1.

The combined uncertainties give a maximum fuel-cladding temperature of 1515°F , as compared with the nominal value of 1350°F for a coolant outlet temperature of 1300°F . For the 1500°F coolant outlet condition, the maximum cladding temperature will be limited to approximately 1700°F .

The distribution of flow to the elements must be accurately controlled, since the magnitude of the flow determines the coolant-temperature rise and the film-temperature difference through the heat-transfer coefficient, both of which influence the cladding temperature. For gas-cooled reactors the coolant-temperature rise has a more significant influence on the cladding temperature than does the film-temperature difference. For example, at the EBOR hot spot the coolant-temperature rise is 510°F and the film-temperature difference is 255°F . A 5% decrease in coolant flow will cause the cladding temperature to increase about 25°F owing to the coolant-

Table 1
EBOR HOT-SPOT ANALYSIS

Condition	Flow Change (%)	Maximum Cladding Temp. (°F)
Nominal	0	1350
Increased heat generation due to uncertainties of 3% for the fuel loading and 10% for axial power distribution	0	1370
Increased heat generation due to 5% uncertainty in radial power distribution	-0.3	1380
Heat-transfer-coefficient uncertainty of 20%	0	1385
Fuel-element areas increased 5% (except for the hot element)	-4.5	1375
Fuel-element area of hot element decreased 5%	-3.0	1360
Orifice coefficients decreased 5% such that flow in the hot element is reduced	-1.9	1360
Combined effect of all uncertainties	-9.5	1515

temperature increase and to increase only about 10°F owing to the film-temperature difference. Accurate flow distribution will be ensured in the EBOR by close tolerances on the orifices and tests to determine the orifice pressure-loss coefficient.

Of all the various factors considered (see Table 1), the heat-transfer correlation caused the largest single increase in the cladding hot-spot temperature. This was owing to the necessity of assuming a large uncertainty in applying a general heat-transfer correlation since experimental values were not available. For this reason, a rather extensive out-of-pile experimental program on heat transfer was initiated in order to obtain the necessary data for the ring-type element.

DETAILED THERMAL ASPECTS OF THE FUEL ELEMENT

Because of the variation in the coolant cross-sectional flow area in the element, it is known that there will be a nonuniform distribution of flow. A greater amount of the coolant will flow in the relatively larger passages, and a lesser amount of flow will be in the smaller passages between adjacent fuel pins. As a result of the flow distribution within the element, there will be a coolant-temperature structure which is dependent on the particular flow pattern, and this in turn will cause a variation in the cladding temperature about the circumference of the fuel rods. It has been indicated that the critical temperature is the maximum fuel-cladding temperature; thus, knowing the variation of the fuel surface temperature about the circumference is basic to the thermal design.

Calculations of the detailed temperature structure in the coolant and heat-producing bodies have been made for infinite arrays of rods on square and triangular patterns.⁽⁹⁾ Rather than perform calculations in this manner, for the EBOR the approach taken was to define an average heat-transfer coefficient based on the bulk coolant temperature associated with a given fuel rod and also to define another heat-transfer coefficient which would give the circumferential variation about the mean value. These two quantities are determined from experiments. The axial power production in a reactor is given from the standpoint of the thermal design. The bulk coolant temperature along the axial direction is readily calculated from an energy balance. With the average heat-transfer coefficient known from experiment, the surface temperature can be computed. By applying the circumferential variation of the heat-transfer coefficient, obtained from tests and correlated as the ratio of the local value to the average about the circumference, the complete fuel-pin cladding temperature and the hot spot can be found.

Implied in the definition of the average heat-transfer coefficient as stated above is the fact that the local bulk temperature must be calculated from an energy balance for the flow associated with each fuel rod. Since the power produced by each fuel rod is different, each fuel rod must be treated individually. It was assumed a priori that there would be no coolant mixing. In fact, it was assumed that imaginary barriers to coolant cross-flow and mixing existed between each rod and the shroud and between each rod and the spine. The assumption of no coolant mixing has been verified by experiment, both qualitatively by means of visual flow observations and quantitatively by means of a coolant-temperature survey downstream of a single heated rod in the test element.

The helical spacers attached to each rod were found not to alter the normal axial flow through the element except, of course, in the immediate vicinity of the spacer. Pressure-drop measurements also verified that the flow is essentially undisturbed by the helical spacers.

Friction-coefficient results for both smooth rods and rods with helical spacers attached are shown in Fig. 6. It can be seen that the two sets of data are correlated by the relation $f = 0.046 \text{ Re}^{-0.2}$, where f is the Fanning friction factor and Re is the Reynolds number based on the equivalent hydraulic diameter. This correlation is commonly used for correlating friction data for smooth tubes in the turbulent-flow regime.

Helical spacers have been used extensively in reactor fuel elements--not only for maintaining the geometrical alignment of fuel rods but also for promoting coolant mixing. In those rod clusters where the flux depression may be 20% or greater, coolant mixing may be warranted even at the expense of pressure drop in order to prevent some fuel rods from running hotter than others. In the EBOR ring-type element the variation in power among the fuel rods is only about 3%, and hence coolant mixing is not an important factor.

An observation of importance was made during the experimental work. This was the occurrence of flow filaments at the base of the helical spacer. These flow filaments were observed to spiral along the length of the fuel rod and ensure good cooling of the cladding in the region near the base of the spacer. If the spacers were set normal to the flow, the streamlines in deviating over the obstruction would give rise to separation of the flow from the surface and to "dead-water" regions at the base of the spacer. Since the spacers are set at an angle to the flow, there is a pressure gradient along the spacer and also a velocity component tangent to the spacer. It is postulated that these two factors cause a small quantity of flow to fill the dead-water region and set up a continuous flow filament along the length of the spacer. The occurrence of these flow filaments at the base of the spacer has also been found to result in good heat-transfer coefficients in this area, and would explain the observation of low temperatures at the base of spiral-wire-wrap rods reported in a recent paper.⁽¹⁰⁾

The circumferential variation of the local-to-average heat-transfer coefficient is found experimentally by using mass-transfer techniques. This method has been employed for heat-transfer investigation of fuel elements for other reactors.⁽¹¹⁾ The heat-transfer coefficients thus obtained are valid for the condition of constant surface temperature. The condition in a reactor probably approaches that of a constant heat flux at the surface, and checks are in progress to determine the difference between the heat-transfer coefficient for the two boundary conditions. Once the circumferential variation of the heat-transfer coefficient is known, the variation in the cladding surface temperature can be found. In calculating the surface temperature, account is taken of the radial and circumferential conduction in the fuel body and the cladding.

In the experimental investigation it was of interest to determine the effect of manufacturing and assembly tolerances on the cladding hot spot.

Fig. 6--Friction coefficient for the ring-type element ($c/d = 0.10$)

Of particular concern was the effect of alignment of the fuel rods in the element. Tests were performed, using the mass-transfer technique, with rods intentionally offset from their nominal position. Smooth rods were used, and the offset of rods was greater than would occur in the actual element, where the spacers would restrict the displacement. Typical results are shown in Fig. 7, where the ratio of the local-to-average heat-transfer coefficient, $h/\bar{h}$, is plotted against angular position, θ , around the test rod. It is seen from this figure that the minimum heat-transfer coefficient which occurs at the minimum spacing between the test rod and the spine does not change when the adjacent rods are offset. This result has important implications for fuel-element design tolerances and, therefore, for the cost of the components comprising the element.

Fig. 7--Variation of the local heat-transfer coefficient with offset rods

FUTURE DEVELOPMENTS IN HEAT TRANSFER FOR GAS-COOLED REACTORS

The primary aim in improving heat transfer is the reduction of capital costs. The two main areas where appreciable improvements are possible are (1) increasing the effective heat transfer to the coolant, thus permitting higher power densities, and (2) reduction of pumping costs, resulting in higher net plant efficiencies.

Research is in progress toward the improvement of heat transfer by means of fins and local turbulence promoters. Improvements in this area appear quite promising.

Pumping costs may be reduced by increasing the specific heat per unit volume of the coolant. Three possible methods are (1) increased system pressures, (2) use of organic vapors, e.g., benzene, and (3) use of appreciable amounts of dust in the coolant. Each of these possibilities has considerable difficulties or unknowns associated with it, making adoption of any of them unlikely in the near future.

REFERENCES

1. Rose, D., The High-Temperature, Gas-Cooled, Graphite-Moderated Reactor (HTGR), General Atomic, Report GA-1158, January, 1960.
2. Rickard, C. L., The HTGR, An Advanced, High-Temperature, Gas-Cooled, Graphite-Moderator Reactor, General Atomic, Report GA-1187, January, 1960.
3. Palmer, L. D., and L. L. Swanson, "Measurements of Heat Transfer Coefficients, Friction Factors, and Velocity Profiles for Air Flowing Parallel to Closely Spaced Rods," Proceedings of 1961 International Heat Transfer Conference, August, 1961. Part III, p. 535.
4. Thompson, W. I., The Maritime Gas-Cooled Reactor Program, General Atomic, Report GA-1089, November, 1959.
5. Trickett, K. A., A Review of the Maritime Gas-Cooled Reactor Program, General Atomic, Report GA-2603, December, 1961.
6. deHoffmann, F., and W. T. Furgerson, MGCR As a Small Control-Station Power Plant, General Atomic, Report GA-1516, July, 1960.
7. Moore, W. C., The Experimental Beryllium Oxide Reactor, General Atomic, Report GA-2385, July, 1961.
8. Moore, W. C., Experimental Bases for the Design of the EBOR, General Atomic, Report GA-2477, August, 1961.
9. Deissler, R. G., and M. F. Taylor, Analysis of Axial Turbulent Flow and Heat Transfer Through Banks of Rods or Tubes, TID-7529 (Pt. 1), Book 2, 1956.
10. Knudson, J. G., and J. M. Batch, "Heat Transfer Coefficients for a Wire Wrapped 19-rod Bundle Fuel Element," Transactions of the ANS, Vol. 4, No. 2, November, 1961; Paper 37-6.
11. Hoffman, H. W., J. L. Wantland, and W. J. Stelzman, "Heat Transfer with Axial Flow in Rod Clusters," Proceedings of 1961 International Heat Transfer Conference, August, 1961, Part III, p. 533.