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THERMAL DESIGN ASPECTS OF
GAS-COOLED REACTOR CORES*

by

Robert Katz

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**THERMAL DESIGN ASPECTS OF
GAS-COOLED REACTOR CORES**

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Robert Katz

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INTRODUCTION

The purpose of this paper is to review the thermal design aspects of gas-cooled reactor cores. This is sometimes thought to be the private realm of the heat transfer "expert." However, the thermal design of a reactor is inextricably interlaced with coordinated efforts on the structural design, neutronics, metallurgy, etc. The problems which are purely heat transfer ones can be discussed separately only with the understanding that a final reactor design involves many interactions.

It should be noted from the title that a restriction has been imposed on the scope of this paper since the reactor core has been specifically identified for discussion. The main emphasis will also be on fuel elements in relation to reactors for electrical power application. The justification for these qualifications is that much of the total design effort is devoted to fuel elements, since this is the predominate source of heat in a reactor, and the heat transfer problems in a reactor core for an electrical power application are similar in other reactor applications and in other heat exchanger (e.g. steam generator) components.

Within certain bounds on design parameters, such as, size, void fraction, fraction of parasitic neutron absorption materials, the design of a fuel element may be varied to a reasonable extent. The main limits on the thermal design are prescribed temperature values for the coolant and allowable temperatures for the fuel and structural materials. In effect, a temperature envelope can be defined, as well as a design temperature criterion. For example, in a core design where a metallic cladding is required around the fuel, the structural or corrosion properties of the cladding material usually dictate the maximum temperature limit. In other core designs the maximum fuel temperature may be the limiting quantity. Either the maximum allowable fuel cladding or fuel temperature becomes the design temperature criterion.

If the only restriction on core thermal design was to limit certain temperatures, the heat transfer problems would be greatly simplified.

However, the temperature limits must be met with minimum pumping power. The pumping power requirements and the design temperature criterion are coupled problems which occupy a great deal of the efforts by heat transfer workers. Therefore, this paper will mainly explore these two aspects of reactor core design. To this end, (1) a few selected reactors will be reviewed and the design temperature criterion identified, (2) the main design parameters available for reducing pumping will be discussed, (3) the heat transfer characteristics needed for design evaluation will be illustrated, (4) a few special heat transfer problems will be briefly presented, and finally (5) the current trend in heat transfer work will be discussed.

HISTORICAL SKETCH

If one were to stretch a point, the history of gas-cooling for reactors would have to start with the first sustained nuclear reaction - the CP-1 atomic pile at Chicago in 1942 - where natural convection of air was used for cooling. However, it is not the purpose of this paper to review all gas-cooled reactors but a brief sketch of the history of gas-cooled reactors, and particularly fuel element designs, will be useful to gain some perspective and to tie in with the main emphasis on core heat transfer.

An extended coverage of gas-cooled reactor technology is given in Reference 1 and further information is available by consulting the references cited in that paper. Reviews of gas-cooled reactors in the United States⁽²⁾ and Britain⁽³⁾ have also been published. The importance of gas-cooled reactors as a source of electrical power is illustrated by a statistic given in Reference 1 that on a world-wide basis the total installed electrical capacity of gas-cooled reactors was about 4000 Mw(e) at the end of 1965 as compared to 1500 Mw(e) for all other types of reactors combined.

Engineering efforts on gas-cooled reactors started about 1945 in the United States, the U.K. and France. The U.K. efforts culminated in the first Calder Hall plant achieving full power in 1956. The first French power reactor was in service a short time later. The Daniel's design study project in the U.S. was never carried to fruition but some of the design features are similar to those found in current advanced designs. Large engineering programs in the U.S. were started about 1956 at the Oak Ridge National Laboratory and at the General Atomic Division of General Dynamics Corporation. The Oak Ridge approach may be roughly compared to the then current British trend except helium was used as the coolant instead of carbon dioxide. The efforts at General Atomic began with the MGCR (Marine Gas-Cooled Reactor)⁽⁴⁾, which later evolved into the EBOR (Experimental Beryllium Oxide Reactor)⁽⁵⁾, and the HTGR (High Temperature Gas-Cooled Reactor)⁽⁶⁾ systems. The HTGR system utilizes all-ceramic cores and thus further advances the technology of gas-cooled reactors.

Table I gives some data on a few selected gas-cooled reactors. The first three in the table are U.K. designs and are representative of the first generation plants, the advanced versions, and the second generation plants, respectively. The Bugey 1 reactor represents the latest efforts by the French to increase performance. The last two reactors are General Atomic HTGR (High Temperature Gas-cooled Reactor) designs. The Peach Bottom reactor, built for the Philadelphia Electric Company, was completed in 1965 and is now in operation. Performance characteristics are also given for the 330 Mw(e) HTGR to be built for the Public Service Company of Colorado.

It may be seen from Table I that the performance of these gas-cooled reactors - power, power density, thermodynamic efficiency, steam temperatures and pressures - has steadily increased. The increased performance is a result of advances in all phases of reactor technology. For example, the increased performance of the Wylfa plant over the Hinkley Point is in large part due to doubling the system pressure because of the use of a prestressed concrete pressure containment instead of a steel vessel.

The first fuel elements developed for gas-cooled power reactors were those used in the British Calder Hall power stations. These elements used uranium metal fuel and were clad with magnox, a magnesium alloy. The choice of this fuel element, as well as the choice of gas-cooling and graphite moderation, was dictated by the lack of enriched fuel. The French were also without means of producing their own enriched fuel and came to basically the same decisions for designing their early reactors; that is, they used uranium metal, a magnesium alloy cladding, graphite moderation, and gas-cooling.

The magnox fuel elements have rather severe temperature limitations. A phase transition of the uranium metal occurs at about 1220°F so the fuel temperature must be limited to below that value to avoid gross dimensional changes. The magnox cladding is limited to below about 850°F due to creep properties. Note in Table I that the gas outlet temperature for the reactors using magnox clad elements is limited to a value below about 780°F and, consequently, the steam conditions are also lower than accepted modern practice. Nevertheless, the magnox elements have proven to be quite successful since the containment of fission products and corrosion resistance in

the CO₂ environment have been satisfactory.

The use of extended surfaces, that is, fins, were essential for the magnox fuel elements in order to provide sufficient heat transfer surface. The first fin designs took what seemed the most obvious form, being longitudinal fins solely for the purpose of providing secondary heat transfer surface. Unhappily these designs had a serious flaw since flow in the small space near the root of the fins did not mix with the remainder of the flow. This resulted in a starved and, therefore, hot flow stream near the root. A partial sketch of a longitudinally finned element with approximate flow contours is shown in Figure 1.

Transverse fins were adopted for the Calder Hall reactors. The transverse fins not only serve as secondary heat transfer surface but also act to significantly increase heat transfer performance without an undue increase in pumping power⁽⁷⁾. Gas mixing and, therefore, heat transfer is excellent since intense vortices are formed between the fins. The vortices are unstable and are periodically expelled into the main stream and reform again. Figure 2 is a sketch of the flow conditions over transverse fins. The characteristics of transverse fin elements are apparently sensitive to flow conditions and the fin height-to-pitch ratio. If the fins are too close together fluid may become trapped at the root leading to a local hot spot.

Because of the limits with transverse fins, more efficient designs of extended surface fuel elements are used in the Hinkley Point and Wylfa reactors in the U.K. and in the latest French reactors. Considerable development work has been carried out in Europe on newer, polyzonal and herringbone, types of finned elements.^(8,9,10,11,12) Both the polyzonal and herringbone finned elements obtain good gas mixing, and consequently good heat transfer, by creating a spiral motion to the flow as well as forming flow streams or vortices between the fins. In the polyzonal type, small helical fins are provided on the cladding and several longitudinal splitters extend from the fuel element surface to the walls of the fuel channel. The flow streams that are formed between the fins are disrupted by the splitters causing a gross helical motion to the main flow pattern. In the herringbone type, the fin helices are fabricated in opposite directions so a longitudinal

splitter is not necessary to disrupt the flow streams between fins to create the gross helical flow pattern. The interaction of the fluid streams between fins in the herringbone fuel element is sketched in Figure 3. It may be easily understood that the flow and heat transfer properties for these fuel elements are very complex. Variations in heat transfer have been observed both circumferentially and longitudinally.

In order to increase performance, the French are going to use an annular fuel element cooled on both the inner and outer surfaces for the Bugey 1 reactor. Figure 4 is a picture of one proposed fuel element showing transverse fins and splitters on the outer surface and a form of roughness on the inner surface. It is reported that herringbone fins will be used on both surfaces in the final fuel element design. The proper distribution of coolant through the inner and outer channels requires accurate experimental data on flow characteristics.

The British AGR (Advanced Gas-Cooled Reactor) program, was a step away from the limitations imposed by natural uranium. In the AGR's, stainless steel cladding and enriched UO_2 fuel will be used. The maximum surface temperature of the stainless steel cladding is limited to about $1480^{\circ}F$. The maximum fuel temperature is limited to below about $2900^{\circ}F$ since fission product release begins to increase significantly above this temperature. The poor conductivity of UO_2 and the need to limit fuel and cladding temperatures mean the fuel must be finely divided to increase heat transfer surface. The rod-cluster type fuel element is being designed for the AGR; see Figure 5.

Increasing heat transfer surface by means of extended surfaces for stainless-steel clad fuel elements is not attractive since stainless steel is a poorer conductor than magnox by a factor of about six, leading to poor fin efficiency. Parasitic absorption of neutrons in stainless steel also makes it desirable to limit the amount in the core. Instead of fins, surface roughening is being used to increase heat transfer characteristics. A discussion of roughening as a means of improving heat transfer will be deferred until later.

The limitation of the maximum fuel cladding temperature may be drastically changed by substituting a ceramic material for the metallic

cladding. This is what is done in the General Atomic Peach Bottom reactor, the Dragon reactor in England and the German AVR Pebble Bed Reactor. These reactors all incorporate graphite as the fuel element structural material allowing surface temperatures of about 1850°F. All of these reactors use fuel particles, the order of 100 microns in diameter, coated with pyrolytic carbon to serve as a barrier to the release of fission products. The fission products which are released are purged from the Peach Bottom and Dragon fuel elements and taken to external trapping systems. This is not possible for the AVR Pebble Bed Reactor and a clean-up system is used to remove fission products from the primary system.

The Peach Bottom fuel element is illustrated in Figure 6. The outer graphite sleeve is 3-1/2 inches in diameter and the total length of the element is 9-1/2 feet. The low permeability graphite sleeve houses annular fuel compacts and an unfueled graphite spine. Fission products are purged by a small flow of coolant to an internal fission product trap at the bottom of the element and what is not held up is carried to an external trapping system.

Although experiments on pyrolytic-carbon coated particles have demonstrated good fission product retention⁽¹³⁾ development work has been continuing on multiple coated fuel particles. Triplex coated particles use a soft buffer layer to absorb fission recoil and thus protect the harder outer layers. In-pile tests on triplex coated particles have been carried to 200,000 Mwd/tonne at 2500°F with no significant damage⁽¹⁴⁾.

These new fuel particle developments have resulted in a trend to unpurged fuel elements as that used for the Fort St. Vrain HTGR. Figure 7 shows the fuel element for this reactor. The hexagonal graphite block is about 14 inches across the flats, and parallel fuel and coolant holes (about 1/2 inch in diameter) are arranged in a triangular pattern.

Besides the fuel and cladding materials discussed thus far, many others have been used or proposed. For example, investigations have been conducted on UO₂ fuel dispersed in a BeO matrix. Hastelloy has been used in place of stainless steel and SiC and BeO have been proposed as alternates for graphite.

Interest in gas-cooled fast reactors has increased in the last few years. Design studies at General Atomic⁽¹⁵⁾ have shown that conversion ratios of 1.55 and 1.65 can be obtained with oxide and carbide fuel, respectively. In these studies, stainless steel was selected as the reference cladding material. The power density in the gas-cooled fast reactor reference design is about 250 kw/liter which may be compared to the values of about 1-10 kw/liter for thermal reactors.

In the brief review of gas-cooled reactors, certain temperature limitations have been indicated which determine the principal thermal design criterion. In the metallic-clad elements, the maximum metal temperature is usually the limiting quantity due to structural properties while the maximum fuel temperature is the limit imposed on ceramic clad elements. Roughened surfaces on metallic-clad elements can improve heat transfer sufficiently that the limit on cladding temperature may be removed with the design temperature criterion becoming the maximum allowable fuel temperature from considerations of fission product release.⁽¹⁶⁾

GENERAL CORE THERMAL DESIGN ASPECTS

One of the first questions that arise in the design of gas-cooled reactors deals with the choice of gas. This is a good starting point since the equations used in comparing gases naturally lead into other important aspects. The choice of gas for reactors has been dealt with by a number of authors. (17,18,19,20) A very simple analysis will be developed here in order to bring out the salient points.

Consider the simplest of reactor cores - a cylinder of length L and diameter D_c which contains a single, constant area coolant hole of diameter D . This may also be thought of as a unit cell segregated from a large core. The thermal and hydraulic characteristics are found from the following equations.

The conservation of mass, energy, and momentum are:

$$\text{Mass:} \quad m = \rho v A \quad (1)$$

$$\text{Energy:} \quad Q = m c_p \Delta T \quad (2)$$

$$\text{Momentum:} \quad \Delta p = \frac{1}{2} \rho v^2 \left(\frac{4 f L}{D} \right) \quad (3)$$

The momentum equation considers only wall shear stresses which are calculated using the friction factor, f , obtained experimentally.

In keeping with the simplicity of the development, the gas will be assumed to obey the perfect gas equation

$$p = \frac{R}{M} \rho T \quad (4)$$

The pumping power, W , necessary to circulate the coolant is approximately given by

$$W = m \frac{\Delta p}{\rho} \quad (5)$$

Two geometrical parameters will be introduced to define the core. These are the void fraction, ν , where

$$\nu = \frac{A}{A_c} = \frac{\pi D^2}{\pi D_c^2} \quad (6)$$

and the total heat transfer surface, S_H , given by

$$S_H = \pi D L \quad (7)$$

The heat transfer coefficient, h , is defined by the equation

$$h = \frac{Q}{S_H \Delta t} \quad (8)$$

The heat transfer coefficient cannot be calculated from first principles for turbulent fluid flow and is empirically correlated by the dimensionless Stanton number, St , defined by

$$St = \frac{h}{\rho \nu c_p} \quad (9)$$

The equations (1) to (9) may be combined in a straightforward manner to give

$$\frac{Q}{A_c} = \nu \frac{p}{R T} \sqrt{2 \frac{W}{Q} \frac{St}{f} \Delta t} M c_p^{3/2} \quad (10)$$

This equation, or variations of it, have been used to compare gases. Although it was derived for a very simplified core design, it is actually independent of the configuration.

To make a direct comparison of gas coolants, assume that the performance parameters - void fraction, pressure, coolant temperature, surface temperatures, and ratio of pumping power to core power - are all prescribed. The quantity $\frac{Q}{A_c}$ is the ratio of the core power to frontal area and is one measure of merit. The influence of the gas only enters via the terms $\frac{St}{f}$

and $(M c_p^{3/2})$. However, for turbulent flow of all gases, the quantity $\frac{St}{f}$ is found to have nearly a constant value for all gases. Therefore, the main influence of the gas may be expressed by

$$\frac{Q}{A_c} \propto M c_p^{3/2} \quad (11)$$

or, introducing the molar specific heat capacity, $C_p = M c_p$, by

$$\frac{Q}{A_c} \propto \sqrt{\frac{C_p^3}{M}} \quad (12)$$

If one were to now seek the ideal gas for this application, the complex hydrocarbons would be a reasonable starting point since the kinetic theory of gases indicates that the molar specific heat increases with the complexity or, more precisely, with the number of degrees of freedom - translational, rotational, vibrational. Considering monatomic and diatomic gases, a low molecular weight is desirable indicating hydrogen, which is, indeed, a good choice for certain applications.

A comparison of a few of the gases proposed for reactors, extracted from Reference 20 which deals with this subject in considerable detail, is presented in Table II. It is seen that, relative to helium, the thermal power is greater for hydrogen, carbon dioxide and steam and less for air. However, keeping the same quantities fixed (pumping power, pressure, temperatures and flow areas), all gases in the table require more heat transfer area than helium. This latter point leads to complexities in the mechanical design such as close spacing of fuel rods in a bundle and aggravates thermal problems. The quantity $\frac{Q}{A_c}$ is not the only parameter which may be used as a basis for comparing coolants. For example, the ratio of the core pressure drop to the system pressure, $\frac{\Delta p}{p}$, which relates to loads imposed on grid plates, etc. tends to favor helium over carbon dioxide. The comparisons also depend upon the average temperature and pressure conditions used in evaluating properties.

Referring back to Table I, it is seen that CO₂ and He are the preferred choices for power reactor application, with the former being used extensively in the U.K. and France, and the latter in the U.S. Obviously, the choice of gas for reactor cooling is not made solely on the basis of thermal performance. Other factors influence the choice such as: Compatibility with structural materials, chemical stability, neutron absorption, availability, and, of course, economics.

The pumping power is usually a significant fraction, 5% - 10%, of the core power in a gas-cooled reactor and a great deal of effort has been devoted to this aspect of the design. Assuming that the core power, coolant temperatures, surface temperature and frontal area are fixed, equation (10) may be rearranged to give an expression for the pumping power. It is found that

$$\frac{W}{Q} \propto \frac{1}{p^2 A^2} \left(\frac{f}{St} \right) \quad (13)$$

Since the pressure is decoupled from the temperature for gas-cooled reactors, the pressure may be increased to reduce pumping power. Equation (13) shows that the pumping power is inversely proportional to the pressure squared so doubling the pressure reduces pumping power by one quarter. Increasing system pressure to reduce pumping power may lead to problems in the containment design. However, as pointed out before, adopting concrete containments has allowed design pressures to increase greatly in the U.K. power reactors.

The investigations on roughened surfaces applied to reactor fuel elements are the result of the heat transfer workers' efforts to tackle the problem of pumping power in gas-cooled reactors. Surface roughening alters both the friction factor, f , and the Stanton number, St . Referring to equation (13), it is seen that reducing the term $\frac{f}{St}$ will reduce pumping power, holding the flow area and pressure constant. One can now conceive of a program of tests to find the roughened surface which gives the greatest reduction in $\frac{f}{St}$ as compared to smooth surfaces. Unfortunately, in no known case of artificial roughening does $\frac{f}{St}$ reduce. In fact, a rough rule for the better roughening schemes is that the friction factor triples while the

Stanton number only doubles, compared to smooth surfaces. This would suggest that all the efforts on surface roughening are misguided. Actually, the key point is that the flow area need not be held constant.

Combining equations (1), (2), (8), and (9) gives

$$\frac{A}{S_H} = St \frac{\Delta t}{\Delta T} \quad (14)$$

This expression shows that the ratio of flow area to heated surface is directly proportional to the Stanton number; that is, it is only a function of heat transfer and independent of friction. Again assuming the core temperatures are fixed and substituting equation (14) into equation (13) gives

$$\frac{W}{Q} \propto \frac{1}{p^2 S_H^2} \frac{f}{St^3} \quad (15)$$

Now it is seen that to reduce the pumping power, the pressure may be increased, the heat transfer surface may be increased, or a roughening scheme applied to the heated surfaces such that the friction factor does not increase faster than the Stanton number cubed. For the simple model discussed here, the better roughening designs would reduce the pumping power ratio by a factor of 3/8 compared to smooth surfaces.

Extended surfaces by means of fins may also have a decided influence on pumping power. Fins are only attractive when the fuel can be a good conductor which is the case with the magnox fuel cans used by the British and French and explain why they have gone to the complex machining necessary for their herringbone fuel elements.

Roughened surfaces for fuel elements yields another benefit. If the heat transfer surface is held constant, equation (15) shows that roughening will reduce pumping power while equation (14) shows that the flow area increases. In a rod-cluster fuel element, it is then possible to increase the spacing between rods which further alleviates heat transfer problems.

The ideas expressed above on the relation of the pumping power ratio to the parameter f/St^3 have been presented a number of times before, most recently in Reference 21.

The story on reactor core design with regard to the pumping power requirements is not complete without indicating that the various benefits from extended surfaces, roughening, and system pressure variations must be related to economics. For example, in the simplified model used to obtain equation (15) it was tacitly assumed that roughening would extend over the entire length of the fuel element. This is neither necessary nor desirable. Roughening need only be applied over that portion of the fuel element required to limit reactor structural temperatures to within prescribed limits.⁽²²⁾ Therefore, it can be appreciated that several avenues are available for optimizing costs of a reactor plant by just considering the fuel element thermal design and the containment design.

FUEL ELEMENT HEAT TRANSFER CHARACTERISTICS

In the general discussion of core thermal performance, it was possible to obtain a great deal of information on basic principles without specific reference to an actual design. Of course, it is only possible to proceed with generalities for a limited time after which it becomes necessary to more fully explore the details of the fuel element design.

Fuel element design has taken many forms and there is still a place for further innovation. The various rod-cluster fuel elements have been employed extensively in the past and are again of interest in fast reactor applications. Therefore, this is a reasonable design to look at in detail in order to illustrate some heat transfer problems.

The rod-cluster fuel elements were introduced in early core designs since it was necessary to provide a large amount of heat transfer surface in order to limit the maximum temperatures of the metallic cladding and fuel. Consequently, accurate information on the heat transfer characteristics were important. Considerable information on heat transfer in smooth, round tubes has been available for some time.⁽²³⁾ Unfortunately, no data existed for clusters of rods and efforts were made to determine the heat transfer characteristics analytically^(24,25) and experimentally.^(26,27)

As a first approximation the heat transfer and frictional characteristics of rod-cluster fuel elements may be estimated by introducing the equivalent hydraulic diameter concept. The equivalent hydraulic diameter is defined as

$$De = \frac{4A}{P_f} \quad (16)$$

where A is the cross-sectional flow area and P_f is the wetted or frictional perimeter. Note that for round tubes the equivalent hydraulic diameter is identically equal to the tube diameter. The equivalent hydraulic diameter enters into the heat transfer and frictional properties through the Reynolds number. For example, commonly used correlations for turbulent flow in smooth

tubes are the following:

$$f = 0.046 \text{ Re}^{-0.2} \quad (17)$$

$$\text{St} = 0.023 \text{ Re}^{-0.2} \text{ Pr}^{-0.6} \quad (18)$$

The Reynolds number is defined as

$$\text{Re} = \frac{\rho v D}{\mu} \quad (19)$$

Therefore, the first approximation in obtaining the heat transfer and frictional correlations for rod bundles is to simply replace D in the Reynolds number by D_e and use equations (17) and (18). It is sometimes surprising how well this concept works for rod-clusters. However, the correlations obtained are only useful for average conditions and no information can be extracted on the local effects about closely packed rods. Also, caution must be observed in using this concept. For example, a dilemma arises in whether to use just the heated perimeter in the equivalent hydraulic diameter for the Stanton number of the total perimeter including parasitic frictional surface. Both parasitic and heated perimeters add to frictional losses so it seems appropriate to sum both when determining the friction coefficient.

The experimental work on rod-clusters has provided much needed data for the thermal design of reactors. One point has become clear from the experimental efforts; namely, the analytical approaches to determining flow behavior in rod-clusters requires further work. The careful experiments by Palmer and Swanson⁽²⁸⁾ at General Atomic have shown that the variation of heat transfer about closely packed rods is not as bad as predicted analytically by Deissler.⁽²⁴⁾ The inadequacy of analytical procedures was also found to be the case by Hoffman, *et al.*⁽²⁹⁾ for another geometrical array of rods. Until better analytical methods are forthcoming, a test program is the only alternative.

A number of experimental techniques have been used for determining the heat transfer characteristics about the periphery of rods in close-packed

arrays. Palmer and Swanson used a single heat flux meter on one test rod in an array and rotated the test rod to obtain the variation of heat flux about the circumference. Others have strategically spotted thermocouples about the circumference to obtain similar data. A mass transfer method has been used by Hoffman⁽²⁹⁾, Katz⁽³⁰⁾ and also by the British and French for investigating complicated roughened surfaces.

In the mass transfer method, a test rod is coated with naphthalene and measurements of the amount of material sublimated from the surface before and after operation in a flow stream yield data which may be related to heat transfer through the analogy of the mechanisms of heat and mass transport. The heat transfer coefficient, h , is defined by

$$h = \frac{\dot{q}}{\Delta t} \quad (20)$$

where \dot{q} is the heat flux from the surface and ΔT is the difference in temperature between the surface and the bulk coolant. The mass transfer coefficient is defined by

$$h_m = \frac{\dot{g}}{\Delta c} \quad (21)$$

where \dot{g} is the mass transfer rate and Δc is the difference in concentration between the surface and fluid.

The heat transfer coefficient is correlated by the Stanton number, as for example by equation (19). A mass transfer Stanton number, St_m , may also be defined. The extended Reynolds analogy, or the Colburn analogy, states that

$$\frac{f}{2} = St Pr^{0.6} = St_m Sc^{0.6} \quad (22)$$

In other words, the Colburn analogy describes the relation between the mechanisms of momentum, energy, and mass transport. This analogy shows how an experimental determination of performance characteristics by one mechanism gives results for another.

Therefore, a detailed mass transfer experiment on a rod-cluster fuel element can yield the needed heat transfer information. Carefully controlled experiments are required in mass transfer work. However, information on the variation of the heat transfer coefficient about the periphery of rods in a fuel element may be found rather simply. The ratio of the local-to-average mass transfer coefficient about a rod is

$$\frac{h_m(\theta)}{\bar{h}_m} = \frac{\dot{g}(\theta)}{\bar{\dot{g}}} \quad (23)$$

Applying the Colburn analogy gives

$$\frac{h(\theta)}{\bar{h}} = \frac{h_m(\theta)}{\bar{h}_m} = \frac{\dot{g}(\theta)}{\bar{\dot{g}}} \quad (24)$$

In other words, the circumferential variation of heat transfer about a rod may be found by measuring the amount of naphthalene sublimated from a coated rod after operation in a flow loop. This procedure has been found to yield repeatable and reliable data and has the distinct advantage that the properties of the sublimating material need not be known.

A typical result obtained from mass transfer tests is shown in Figure 8. These data were obtained for a 12 rod, telephone-dial type of fuel element which was under development by General Atomic for use in the MGCR (Marine Gas-Cooled Reactor) and the EBOR (Experimental Beryllium Oxide Reactor) systems. This figure shows how the heat transfer coefficient varies about the circumference of the fuel rods for the normal configuration and an off-design configuration where the rods were purposely displaced. It is seen from this figure that the ratio of the minimum clearance to the rod diameter, $\frac{c}{d}$, is a basic parameter for this element and this is true for all rod-cluster fuel elements. The maximum variation of the heat transfer coefficient is seen to be about $\pm 20\%$ from the average. The temperature of the fuel cladding does not vary as much since conduction in the cladding and fuel tend to smooth out this condition. Analyses⁽³¹⁾ have shown that for this element the temperature variation is reduced to $\pm 13\%$ by heat

conduction in the rods.

Other conventional tests conducted on the telephone-dial fuel element yielded data on mixing, friction and the average heat transfer. Visualization tests confirmed the prediction that mixing of the coolant among the rods was poor. Helical wire spacers were wrapped around the rods in order to maintain alignment in the element and these spacers only caused a very local perturbation in the flow. Pressure drop measurements showed that the friction factor could be correlated by the smooth tube equation, equation (17), using the equivalent hydraulic diameter. This result was found by experiment to be true for both smooth rods and rods with the helical spacers installed. The final friction correlation used was

$$f = 0.075 \text{ Re}^{-1/4} \quad (25)$$

which is the Blasius equation for smooth tubes and gives results equal to equation (17) within the experimental error. The heat transfer results were correlated by

$$\text{St} = 0.029 \text{ Re}^{-1/4} \text{ St}^{-0.6} \quad (26)$$

The Colburn analogy should give a coefficient of 0.037 in the Stanton number correlation if the equivalent hydraulic diameter concept were used. Clearly, the equivalent hydraulic diameter concept is not valid here which may be guessed since this design has a relatively large parasitic friction surface.

SPECIAL HEAT TRANSFER CONSIDERATIONS

Predicting the thermal performance of a nuclear reactor requires quite elaborate information on heat transfer and fluid behavior for the fuel element. The above discussion on the heat transfer characteristics of rod-cluster type fuel elements is meant to show the extent to which investigations have gone to obtain accurate data. There are some further aspects of heat transfer in reactors which must be taken into account. These are rather special considerations and will only be touched upon briefly since each could become a subject in itself.

The heat transfer and friction correlations given by equations (17) and (18) are valid for small differences between the wall temperature and bulk coolant temperature. When the temperature difference becomes large it has been found that the above correlations underestimate the heat transfer mainly because the fluid properties, primarily the viscosity in the Reynolds number, are temperature dependent and a question arises as to what temperature to use for evaluating the fluid properties. One method for removing this difficulty is to apply a correction factor to the Stanton number and friction factor correlations. The correction factor has taken many forms. For gases, one device is to include a multiplication factor

$$\left(\frac{t_w}{t} \right)^{-\alpha}$$

where t_w is the wall temperature, t is the bulk coolant temperature, and α is an experimentally determined coefficient. The α coefficient typically has a value of 0.1 for the friction factor and 0.5 for the Stanton number correlations, respectively. Another device is to define a Reynolds number, $[Re]_f$, where the properties in this parameter are evaluated at the film temperature, t_f , defined by

$$t_f = \frac{1}{2} (t_w + t)$$

Since the heat transfer correlations, including large differences in wall and bulk fluid temperatures, are obtained experimentally, caution must prevail in their use since the empirically determined correlations are based on specific fluids and geometrical arrangements. In a recent paper⁽³²⁾ sixteen different heat transfer correlations were compared and most were found to give fair agreement, although differences could be traced to the choice of physical properties used as well as differences in geometries tested.

The correlation for heat transfer, equation (17) and (18), are not only valid for small temperature differences but are also only valid for fully developed flow and temperature conditions. Another correction factor must be taken into account for the entry region. The heat transfer coefficient in turbulent duct flow is mainly dependent upon the boundary layer conditions adjacent to the surface. Near the entrance of the channel the boundary layer is very small and builds up as the flow attains fully established conditions. Therefore, heat transfer is very good at the entrance region and reduces to an asymptotic value downstream. The entry length depends very much upon the particular geometry but is usually considered to be restricted to lengths 10 to 50 times the equivalent diameter. For fuel elements extending the full length of a reactor core, the correction for the entry region is usually unimportant since the maximum fuel temperatures occur far downstream from any entry affect. However, in the British and French reactors the fuel elements are relatively short slugs and the entry length influence on heat transfer is significant. In effect, when short fuel element slugs are used or grid spacers are placed along the length of the fuel channel, fully developed flow may never occur.

In those reactors which stack short sections of fuel elements along a fuel channel, the inevitable misalignment of elements may deteriorate the heat transfer performance. The effects of a misalignment of two adjacent elements has been observed⁽³³⁾ to influence heat transfer performance of several elements in the downstream direction.

The subject of hot-spot or hot-channel factors deserves some mention. It is sometimes stated that hot-spot factors are the heat transfer engineers' connection with reality or by others as being engineering safety

or "ignorance" factors. In early design work each uncertainty - heat transfer correlation, power distribution, flow measurement, coolant temperature measurement, manufacturing tolerances, etc. - were assessed individually and multiplied together to obtain an overall hot-spot factor which could then be used in establishing performance limits. This procedure is suggested in older textbooks⁽³⁴⁾ and even new ones.⁽³⁵⁾ Multiplying all the uncertainties together is known to be highly conservative and unrealistic. Statistical methods for combining uncertainties has been proposed.^(36,37,38,39) This approach has merit, although it is still necessary to find better ways to identify and evaluate each uncertainty as to magnitude and distribution in order to properly combine them by statistical methods.

The above thermal design aspects of reactor cores are not restricted to only gas-cooled systems but are features to be evaluated for any coolant medium. However, gases do have one further property, namely compressibility, which is different from water and liquid metal media. The importance of compressibility may be evaluated through the Mach number parameter. It may be shown that compressibility effects are not significant for values of the Mach number below about 0.1. In helium cooled reactors, the maximum Mach number is typically about 0.02 so compressibility may be completely ignored to a very good approximation. In CO₂ cooled reactors the Mach numbers are higher but the coolant may still be treated as an incompressible fluid.

OUTLOOK FOR THE FUTURE

Just as in the past, future improvements in gas-cooled reactors will depend upon advances in all phases of technology. It has been mentioned, for example, that adopting concrete containment structures allow increasing system pressure which reduces pumping losses. Other benefits arise from concrete vessels since the trend is to enclose the entire primary system inside the containment structure, making for a compact arrangement and elimination of steam generator containment and some piping problems. In the area of heat transfer by means of gases, the main emphasis seems to be currently in two directions: experimental and analytical investigations of roughened surfaces and improvements in analytical methods in general.

Surface roughening is, in many respects, a more subtle way of increasing heat transfer than extended surfaces. By definition, surface roughening does not increase the surface area by any significant degree but acts on the fluid boundary layer to promote turbulence. In fact, surface roughening would be regarded by aerodynamicists as simply a special form of a boundary layer turbulence promoter. Turbulence promoters have been around for some time to bring about desired behavior in aircraft and even such an everyday item as the golf ball.

The fully developed turbulent boundary layer is usually represented as in Figure 9 and is described by referring to a laminar sublayer next to the wall, a buffer layer and then the fully turbulent zone. As an illustration, the laminar sublayer has a thickness of 0.004 inch for flow in a 1-inch diameter pipe at a Reynolds number of 20,000 which corresponds to an air velocity of about 40 ft/sec. The roughening of the surface need only extend just beyond the laminar sublayer to be effective, so it can be seen that the surface changes are, indeed, subtle.

Proposed surface roughening for fuel element designs have included many shapes, such as square threads, sine waves, circular wires, and various other rib cross-sections; see Figure 10. The literature on surface roughening is extensive. (40-45)

In rod-cluster type fuel elements, surface roughening is only applied to the heated surfaces, the rods, and the outer shroud is left smooth. Roughening the outer shroud would only increase pressure losses without any gain in heat transfer where the improvement is necessary. Basic tests on roughening are usually carried out on simpler configurations such as flat plates or annuli. Using test data from the simpler configurations for the complex fuel element designs poses a special problem in how to separate the combined affects of rough and smooth surfaces. This problem was first investigated by Hall⁽⁴⁶⁾ and extended by others.⁽⁴⁷⁾ Work still remains to be done in this problem area.

The greater reliance on the high speed digital computer for solving complex problems has in some ways revolutionized engineering. The solution of three-dimensional, transient heat conduction problems is a formidable task by classical analytical procedures and can only be carried out for very simple geometries and boundary conditions. With computers, it is not at all unusual today to obtain numerical answers to complex three-dimensional, heat conduction problems within one working day. Of course, the development of the computer codes to perform the calculations took months to years and methods for improving the efficiency and generality of these tools in a continuing effort. Certainly much remains to be done to make computer codes better able to solve the complex problems in engineering, and this has a direct bearing on the economics of nuclear power plants. Probably the greatest limitation on computer codes, besides inevitable human errors, is the uncertainty in input data. As an example, it would be the simplest of tasks to program into a code any of the many forms of heat transfer correlations to correct for high film temperature differences or the entry length, but this would be useless as a generally valid approach since each correction is strictly applicable to very special circumstances. Obviously, there remains much to be done on fundamentals in heat transfer that would provide universally applicable data.

In this paper extended and roughened surfaces have been discussed as means for improving heat transfer in gas-cooled reactors. These are not the only methods that have been proposed. Impinging sound⁽⁴⁸⁾ and electric⁽⁴⁹⁾

fields have been considered to a limited extent. An older proposal and one that is still being actively investigated⁽⁵⁰⁾ is to add solids to the gas. The use of solid suspensions has been avoided due to problems of deposition of the solids on cold surfaces which might seriously deteriorate the performance of steam generators. In Reference 50 tests are reported which were conducted with graphite particles (the order of 10 microns in diameter) added to CO₂ gas (with a density the order of one pound per pound of gas) in a cross-flow heat exchanger with no deposition found. However, it seems unlikely that this method of improving heat transfer will be adopted for commercial power plant stations in the near future.

In closing, it might be mentioned that several difficult problems relating to reactor cores have not been discussed. The problem of fuel rod bowing is a coupled flow, heat transfer and elasticity problem which is important in rod-cluster type fuel elements. The problem of acoustic interaction with structures involves the entire primary flow system. Information on these problems may be found in the literature.

NOMENCLATURE

<u>Symbol</u>	<u>Definition</u>
A	Fuel channel cross-sectional flow area
A_c	Core frontal area
c	Minimum clearance between fuel rods
c_p	Specific heat capacity at constant pressure
C_p	Molar specific heat capacity at constant pressure
d	Fuel rod diameter
D	Coolant hole diameter
D_c	Core diameter
D_e	Equivalent hydraulic diameter
f	Fanning friction factor
\dot{g}	Mass transfer rate
h	Heat transfer coefficient
$h(\theta)$	Local heat transfer coefficient about fuel rod circumference
\bar{h}	Circumferentially averaged heat transfer coefficient
$h_m(\theta)$	Local mass transfer coefficient about fuel rod circumference
\bar{h}_m	Circumferentially averaged mass transfer coefficient
L	Core length
m	Coolant mass flow rate
M	Molecular weight
p	Reactor coolant pressure
P_f	Frictional perimeter of fuel channel
Pr	Prandtl number

<u>Symbol</u>	<u>Definition</u>
\dot{q}	Heat flux
Q	Total core power
R	Universal gas constant
Re	Reynolds number
Sc	Schmidt number
S_H	Total core heat transfer surface
St	Stanton number
St_m	Mass transfer Stanton number
t	Local bulk coolant temperature
t_f	Local film coolant temperature
t_w	Local wall temperature
T	Coolant temperature
v	Coolant mean velocity
W	Pumping power
Δc	Concentration difference
Δp	Core pressure drop
Δt	Temperature difference between bulk coolant and surface
ΔT	Difference between core inlet and outlet coolant temperature
θ	Circumferential angle of fuel rod
μ	Coolant viscosity
ν	Void fraction
ρ	Coolant density

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TABLE I

Characteristics of Selected Gas-Cooled Power Reactors

Plant	Hinkley Point A	Wylfa	Dungeness B AGR	Bugey 1 EDF	Peach Bottom HTGR	Fort St. Vrain HTGR
Power, Mw(e)	250	590	610	540	40	330
Efficiency	25.5	31.4	41.8	29.1	34.6	39.4
Power density, kw/liter	0.74	0.87	2.5	1.8	8.3	6.3
Active core length, ft.	25	30	27	29.5	7.5	15.5
Active core diameter, ft.	49	57	31	40	9	19.5
Pressure, psia	200	400	485	620	350	700
Inlet temperature, °F	355	475	605	435	650	755
Outlet temperature, °F	710	775	1245	745	1340	1445
Maximum can temperature, °F	800	840	1340	830	1910	
Maximum moderator temp., °F	720		840	750	2425	1900
Maximum fuel temperature, °F	1100	1055	2960	1020	2425	2300
Steam temperature, °F	685	755	1050	735	1000	1050
Steam pressure, psia	655	718	2315	515	1500	2400
Coolant	CO ₂	CO ₂	CO ₂	CO ₂	He	He
Moderator	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite
Containment vessel	Steel	Concrete	Concrete	Concrete	Steel	Concrete
First power	1965	1968	(1970)	(1971)	1967	(1972)
Country	U.K.	U.K.	U.K.	France	U.S.A.	U.S.A.

TABLE II

Thermal Performance Comparison of Common Gases
Relative to Helium

Gas	He	CO ₂	H ₂	Steam	Air
Thermal Power	1.00	1.22	2.41	1.18	0.70
Heat Transfer	1.00	1.32	1.09	1.60	1.24

Assumptions:

1. Allowable temperatures, pressure, pumping power ratio, and flow area fixed.
2. Gas properties evaluated at 500 psia, and 1000°F average temperature.

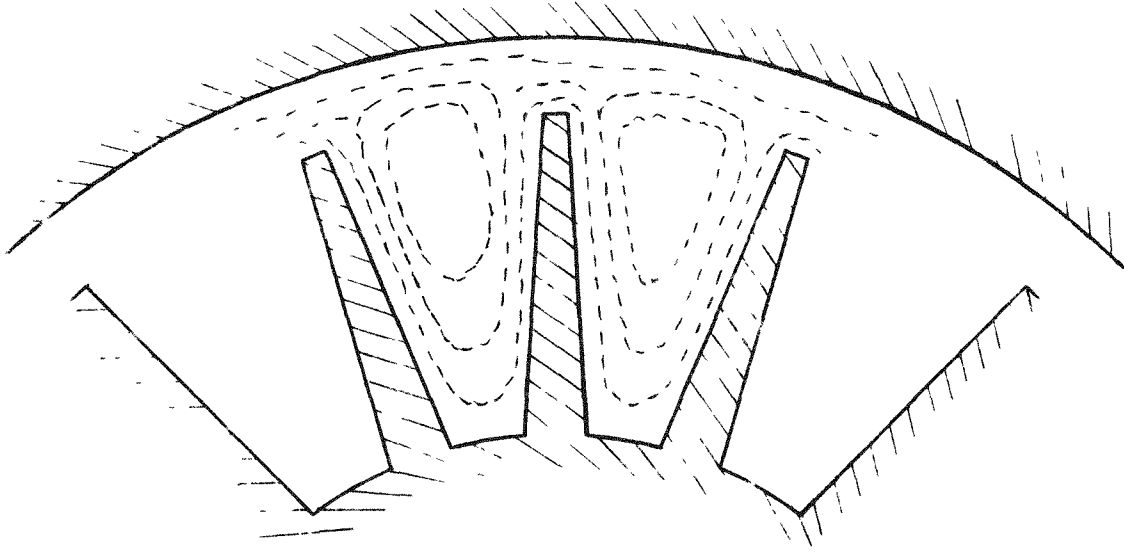


Figure 1 - Sketch of Longitudinally Finned Fuel Element and Approximate Flow Contours

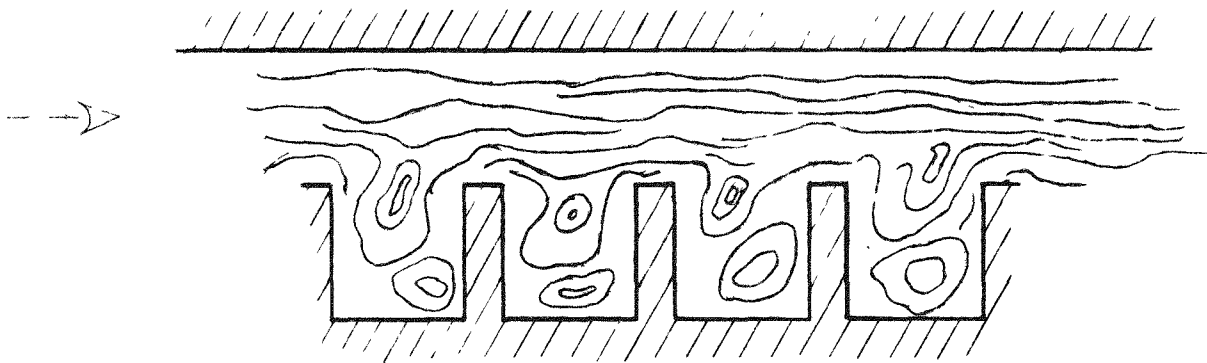


Figure 2 - Sketch of Flow Over Transverse Fins

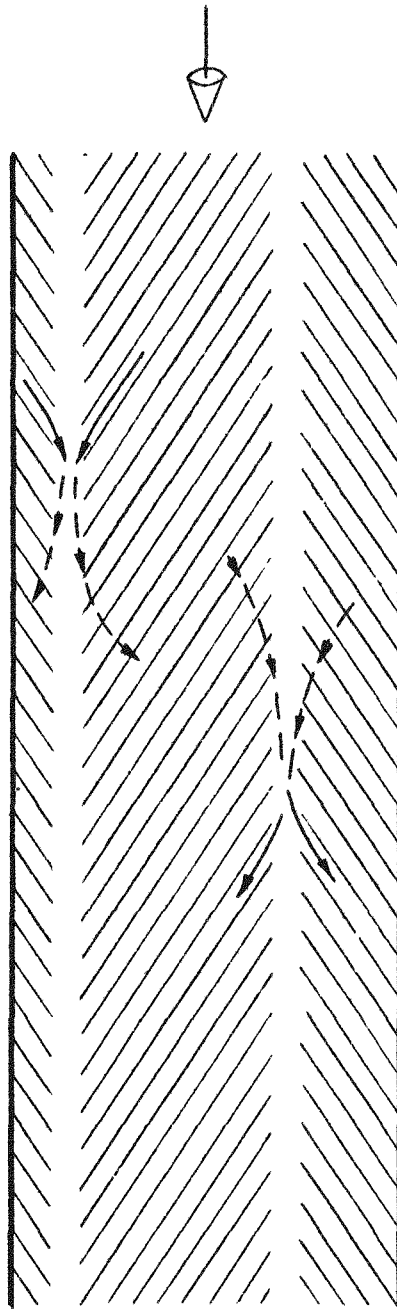


Figure 3 - Flow Characteristics of the Herringbone Fuel Element

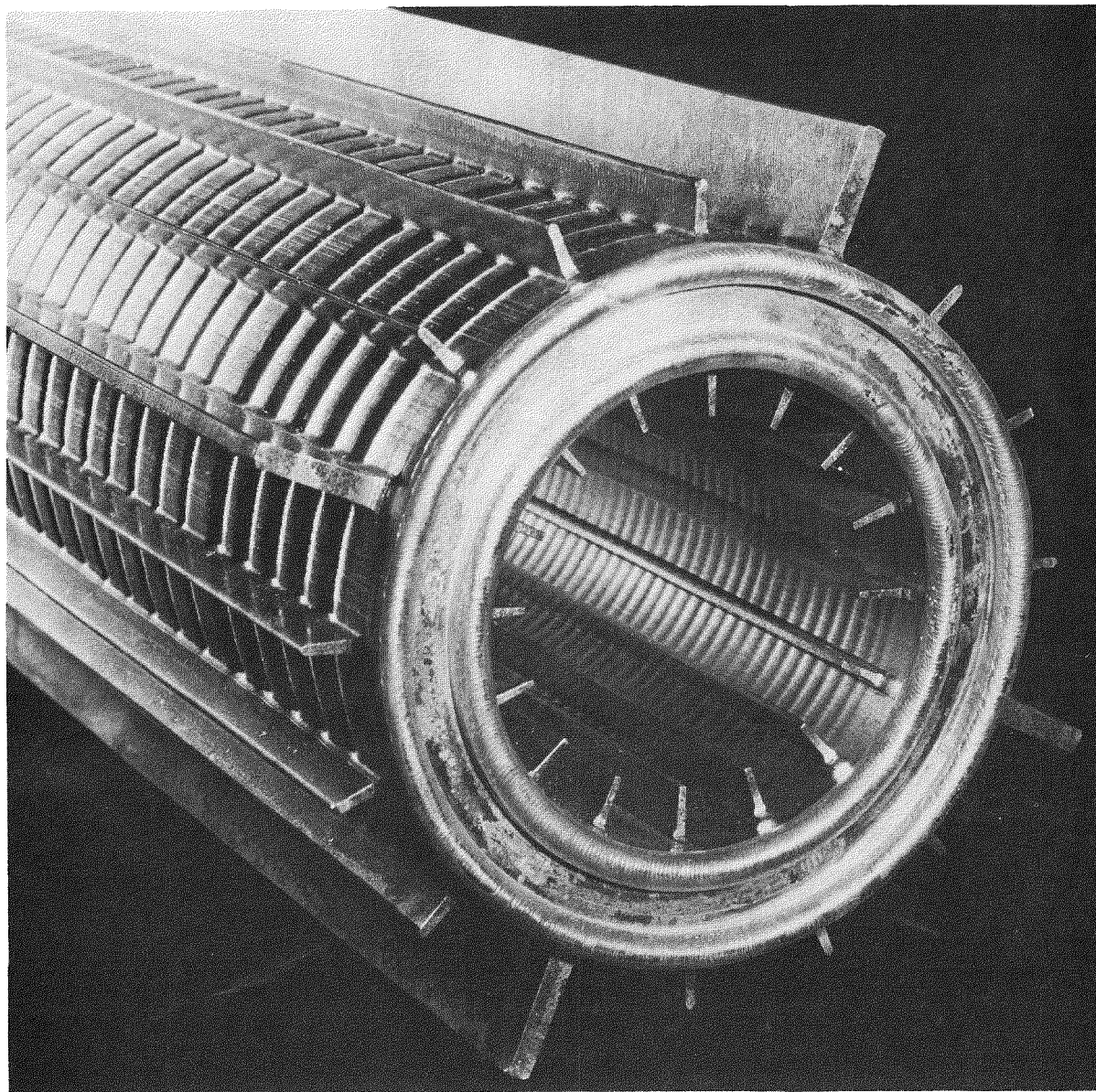
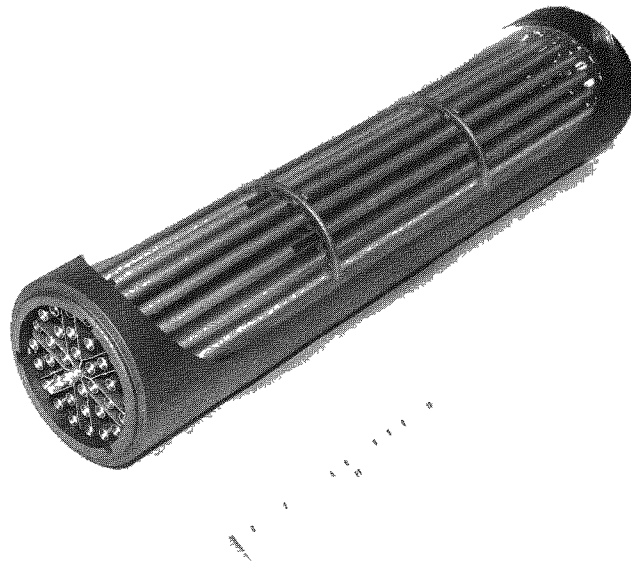
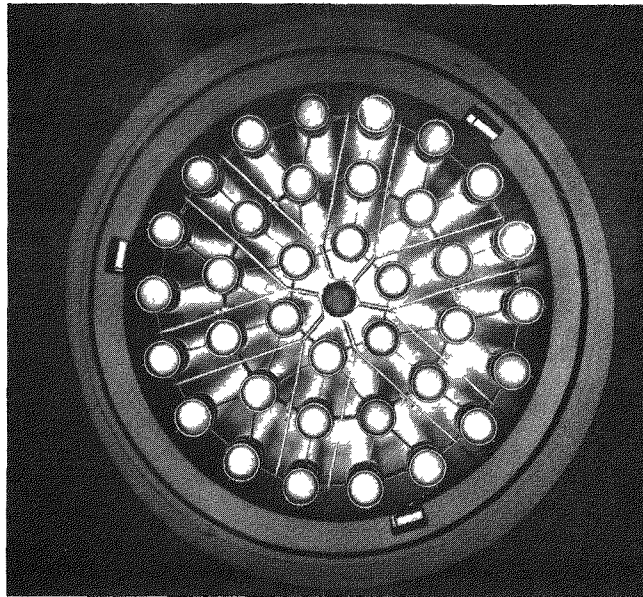


Figure 4 - Annular Fuel Element (being developed)
(courtesy of Commissariat A l'Energie Atomique)



(a)



(b)

Figure 5 - Thirty-six pin AGR Fuel Assembly; (a) cutaway
(b) end view (courtesy of U. K. Atomic Energy Authority)

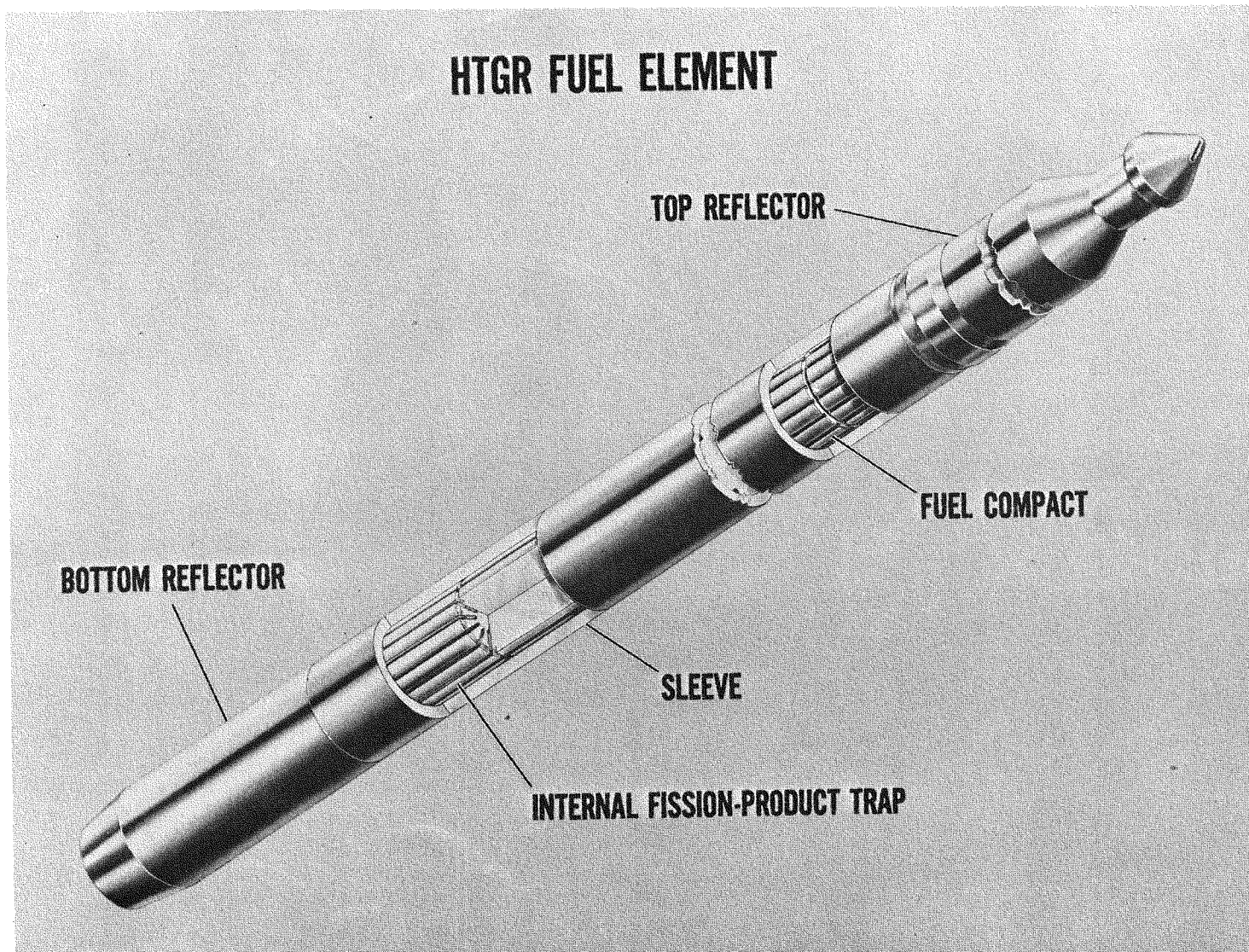
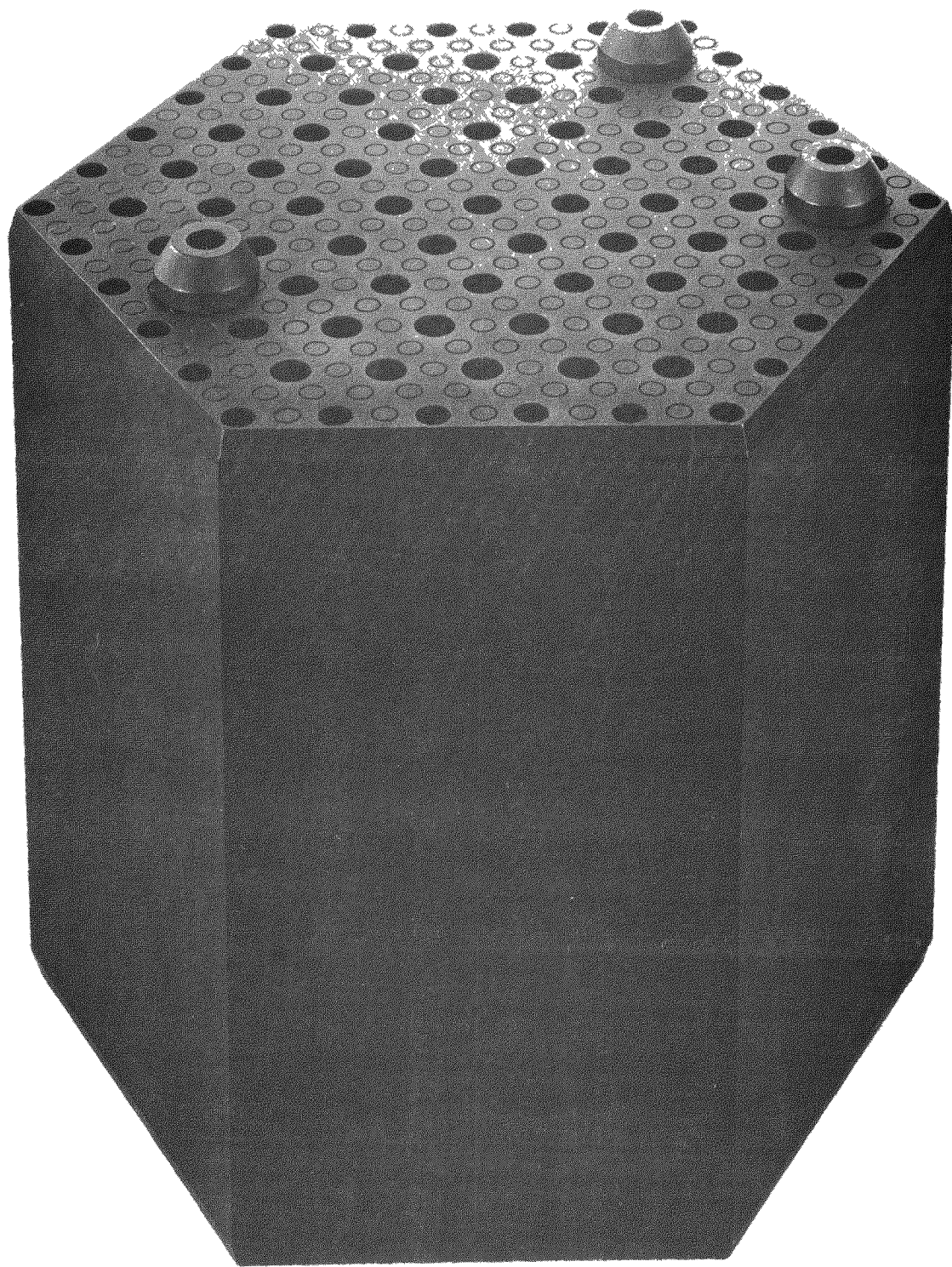


Figure 6 - Peach Bottom Fuel Element



HT-37,406

Figure 7 - Hexagonal Block Fuel Element for the HTGR

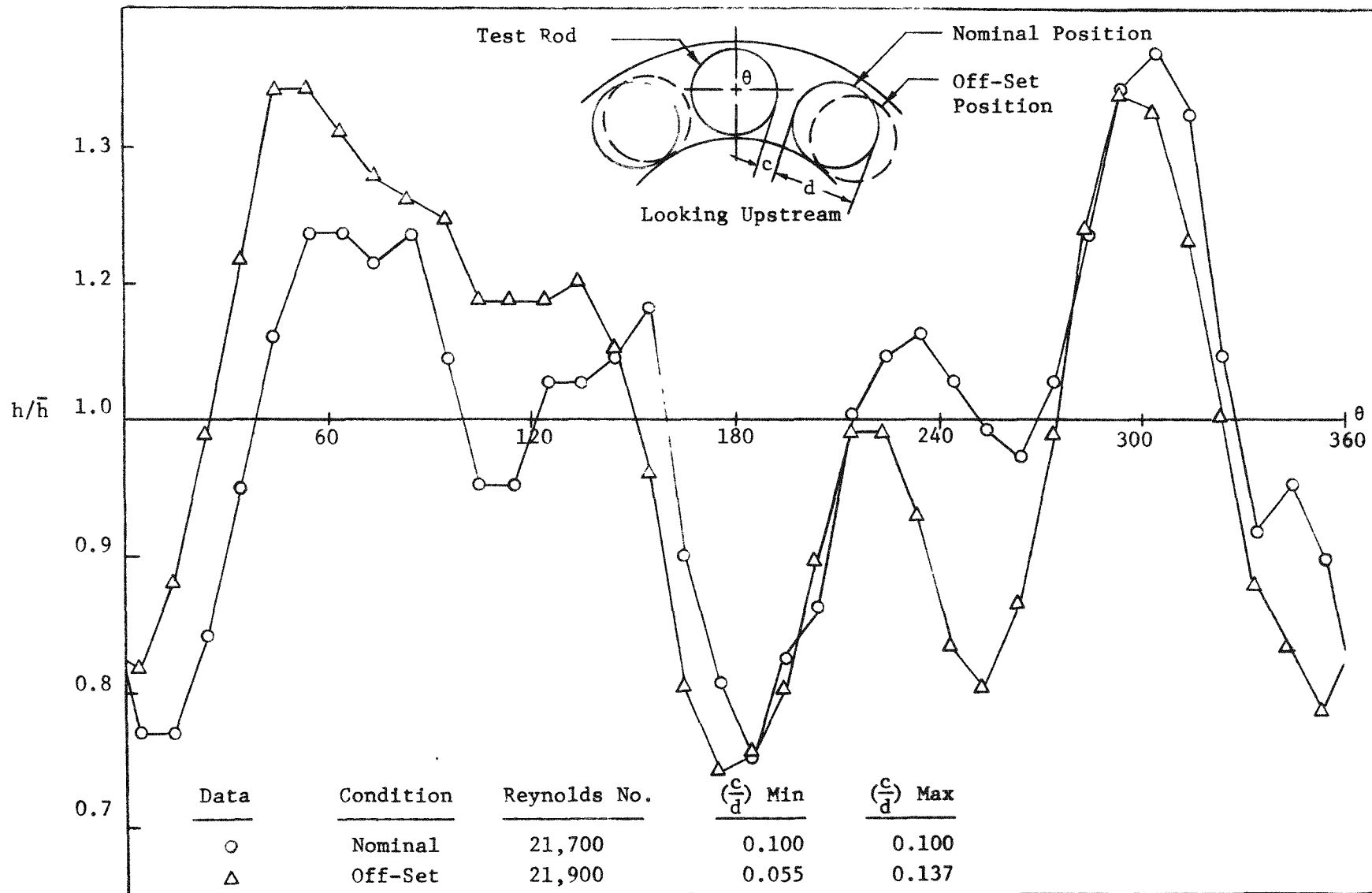


Figure 8 - Circumferential Variation of Heat Transfer for the 12 Rod, Telephone-Dial Fuel Element

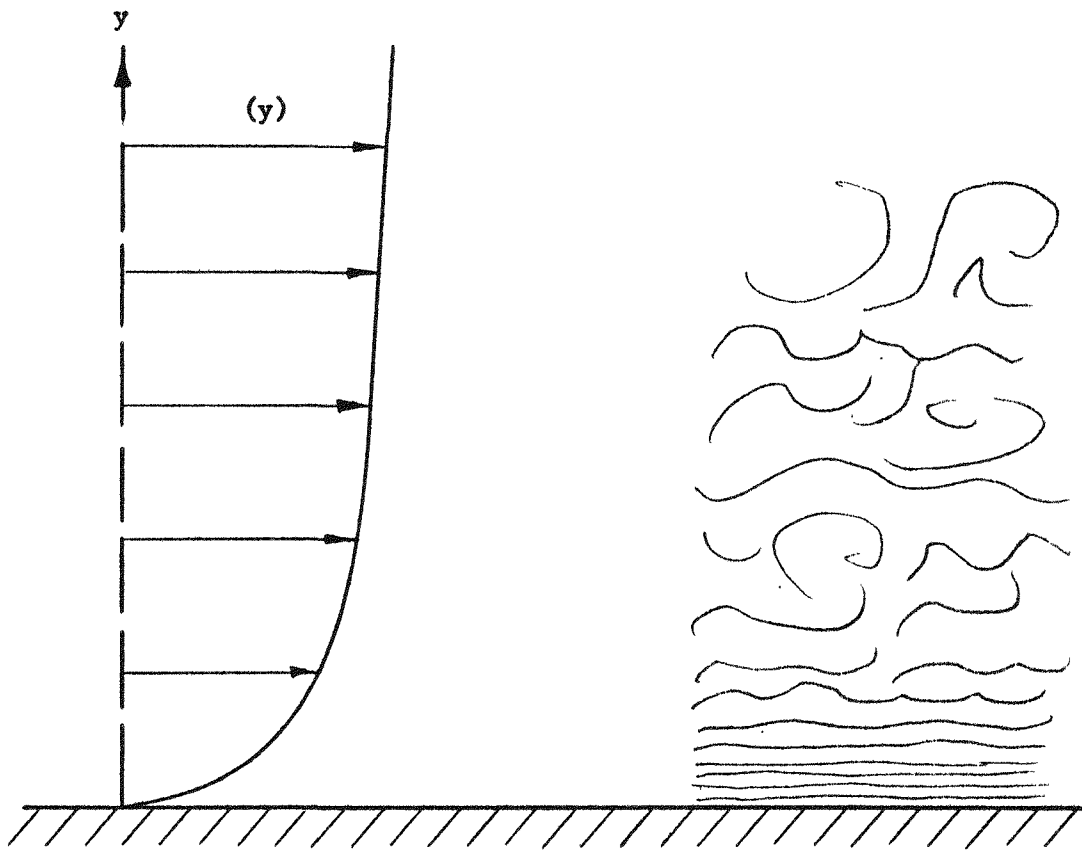


Figure 9 - Representation of Fully Developed Turbulent Boundary Layer

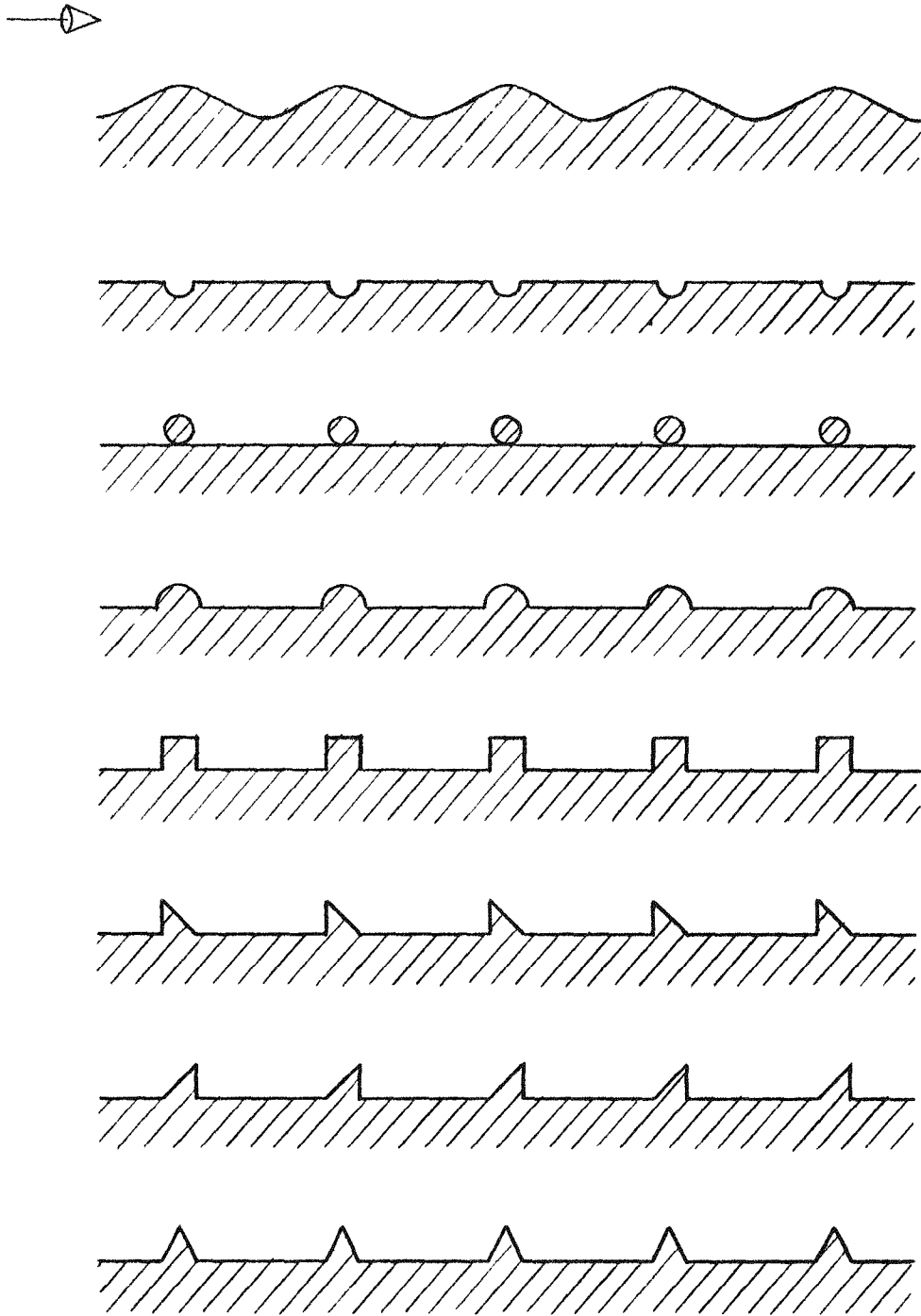


Figure 10 - Various Proposed Roughened Surfaces