MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0361252 5

CENTRAL RESEARCH LIBRARY
DOCUMENT COLLECTION

ORNL-2510

cy. 1

THE ORNL GAS-COOLED REACTOR

ADVANCED CONCEPTS

CENTRAL RESEARCH LIBRARY
DOCUMENT COLLECTION
LIBRARY LOAN COPY
DO NOT TRANSFER TO ANOTHER PERSON
If you wish someone else to see this document, send in name with document and the library will arrange a loan.

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

ORNL-2510

Contract No. W-7405-eng-26

THE ORNL GAS-COOLED REACTOR ADVANCED CONCEPTS

By the Staff

of the

Oak Ridge National Laboratory

DATE ISSUED

SEP 18 1958

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES

3 4456 0361252 5

SECTION 14

DESIGN DEVELOPMENT

14. DESIGN DEVELOPMENT

Contents

14.1 Gas-Cooled Power Reactor Application 14.1

14.1.1 Design Objectives 14.3

14.1.2 Promising Reactor Systems 14.13

14.2 Fuel Element Design 14.21

14.2.1 General Requirements 14.21

14.2.2 Hazards and Maintenance Considerations in Advanced Gas-Cooled Reactors 14.34

14.2.3 Systems with Unclad Fuel Elements 14.37

14.3 Power Recovery 14.39

14.3.1 Steam System 14.39

14.3.2 Selection of Coolants and Interchangeability 14.41

14.3.3 Gas Turbine Applications 14.42

14.3.4 Power-Recovery Component Problems 14.47

14.4 Instrumentation and Control 14.56

14.4.1 Analog Simulator Studies 14.57

14.4.2 Control-Rod Mechanism 14.57

14.4.3 Neutron Detection 14.58

14.4.4 Fuel Element Operating Conditions 14.58

14.4.5 Pressure Vessel Penetrations 14.59

14. DESIGN DEVELOPMENT

14.1 Gas-Cooled Power Reactor Application

In the fall of 1957, when the Atomic Energy Commission undertook an investigation of the possible role of gas-cooled reactors in the United States reactor development program, the attention of the Kaiser Engineers, with their subcontractor ACF Industries, and of the Oak Ridge National Laboratory, was directed to two main questions of immediate importance:

(1) What sort of natural-uranium, graphite-moderated gas-cooled reactor will produce the lowest cost power in the United States, subject to the provision that the reactor be suitable for almost immediate design and construction, with a bare minimum of development effort?

(2) Would a slightly enriched version of this reactor, also suitable for early construction, produce power at more favorable cost?

These questions have now been answered in some detail.[1,2,3,4,5] It has become apparent that the natural uranium reactor, owing to its relatively high capital cost, is not likely to be attractive in the limited States. A similar conclusion was reached earlier in studies by American-Standard Corporation[6] and by Atomics International Company.[7] It appears also that a slightly enriched gas-cooled reactor could be built now that would be generally competitive with other power reactors in existence or under construction; and that potential exists for significant reductions in power

---

[1] The ORNL Gas-Cooled Reactor, ORNL-2500 (April 1, 1958).

[2] Gas-Cooled Power Reactor, Preliminary Design, 55,000 kw Prototype, Natural Uranium Nuclear Power Plant, 58-1-RE, KE-ACF (March, 1958).

[3] Gas-Cooled Power Reactor, Feasibility Study, Optimum Natural Uranium Nuclear Power Plant, 58-2-RE, KE-ACF (March, 1958).

[4] Gas-Cooled Power Reactor, Feasibility Study, 44,000 kw Prototype, Partially Enriched, Uranium Nuclear Power Plant, 58-3-RE, KE-ACF (March, 1958).

[5] Gas-Cooled Power Reactor, Feasibility Study, Optimum Partially Enriched Uranium Nuclear Power Plant, 58-4-RE, KE-ACF (March, 1958).

[6] A. Puishea, D. P. Herron, D. R. Mash, [illegible] W. Webster, Comparison of Calder Hall and PWR Reactor Types, AECU-3398 (March 1, 1957).

[7] W. Banks, G. A. Schneider, W. T. Morgan, and E. B. Ash, An Evaluation of the Calder Hall Type of Nuclear Power Plant, NAA-SR-1833 (January 18, 1957).

cost with future gas-cooled reactors. In addition, the General Electric Company at Hanford has carried out a broad parametric study[8] of graphite-moderated gas-cooled reactors with respect to such variables as reactor power output, coolant composition, power-recovery cycle, and details of core design. This study has established that graphite-moderated gas-cooled reactors will approach their minimum power costs in the range of 1000–1500 Mw(T) per reactor, and that a gas-turbine cycle at 1300°F is no more advantageous than an indirect cycle at 1100°F. The study also indicated that optimum power costs for such systems (8 – 10 mills/kwhr) may not be much greater than for fossil-fuel plants in some parts of the United States, a conclusion that would seem to justify further investigation of the possibilities of gas-cooled reactors in this country.

In the past several years, a number of studies have been undertaken to examine the feasibility of several other types of gas-cooled reactors for power production,[9–15] including heavy-water moderated reactors, graphite pebble-bed reactors, and fluidized-solid reactors. However, in spite of the substantial effort that has been expended in scouting the potential capabilities of a number of concepts, there does appear to be lacking, thus far, in the gas-cooled reactor program, a thoroughgoing attempt to bring the examination of the various gas-cooled reactor concepts to roughly equal levels of reliability, and to establish a comparative evaluation of the several types of gas-cooled reactors on a uniform basis. What is needed is

---

[8] A Parametric Study of the Gas-Cooled Reactor Concept, HW-54727 (March 1, 1958).

[9] A High-Temperature Gas-Cycle Reactor System for Steam Generation, Sanderson-Porter and Alco Products, Inc. (April, 1957).

[10] A. Schock et al., Gas-Cooled Pebble-Bed Reactor for a Large Central Station, ORSORT, ORNL-CF-57-8-12 (August, 1957).

[11] 125 Mwe Pebble-Bed Reactor, Rough Draft, Alco Products, Inc., and Sanderson-Porter (1958).

[12] J. F. Smith, Jr. et al., Gas-Cooled Ceramic Power Breeder Reactor, ORSORT, ORNL-CF-5408-235 (August, 1954).

[13] Feasibility Report on a Fluidized Solids Reactor System, NEA-5401, Diamond Alkali Co., Foster Wheeler Corporation, Pioneer Service and Engineering Co. (January, 1954).

[14] C. L. Teeter et al., Fluidized Bed Reactor Study, ORSORT, ORNL-CF-57-8-14 (August, 1957).

[15] D. Krucoff and F. Norin, The Armour Dust Fueled Reactor Concept. Paper presented to Nuclear Science and Engineering Conference, Chicago, March, 1958, Reprint 100.

a rather rigorous appraisal, not alone of the theoretical capabilities of various core designs or power-recovery systems, but also of the probable engineering difficulties associated with each concept, and of the probable effect on power cost or other figure of merit of the unique problems presented by each system.

The purpose of this report is to discuss a few of the uses for which gas-cooled reactors may be designed and to present a brief commentary on several of the systems that have been suggested for these applications. In addition, attention is drawn to a number of general problems that will accompany the design and construction of high-performance systems. This evaluation is admittedly very preliminary, for it is clear that much solid engineering work remains to be done before it will be possible to identify the most promising lines of development in gas-cooled reactor technology.

14.1.1 Design Objectives: The attractiveness of a reactor design will depend, of course, upon the purpose for which the reactor is intended. A comparative evaluation of several different reactor concepts may be made with respect to any one of several different design objectives; and it is to be expected that a given concept may be well suited for one purpose and unsuited for another. Three major areas of interest may be clearly identified within the scope of the program for developing gas-cooled reactors for electric power. These are:

(1) low cost power in the United States,

(2) reactors suitable for export, and

(3) reactors capable of thermal breeding.

Low Cost Power. This has been the principal objective of the power reactor program for years, and has recently been restated as one of the major goals of the Commission's reactor development program.

Recent studies of gas-cooled reactors[16,17,18] have led to the conclusion that large gas-cooled reactors are probably competitive with other reactor

---

[16] The ORNL Gas-Cooled Reactor, ORNL-2500 (April 1, 1958).

[17] Gas-Cooled Power Reactor, Feasibility Study, Optimum Partially Enriched Uranium Nuclear Power Plant, 58-4-RE, KE-ACF (March, 1958).

[18] A Parametric Study of the Gas-Cooled Reactor Concept, HW-54727 (March 1, 1958).

types under active consideration in this country, and that they show promise of substantial improvement in economy. This situation is perhaps unexpectedly favorable for the gas-cooled reactor, and results from the following considerations:

1. Gas-cooled reactors are not inherently at so great a disadvantage with respect to heat transfer as has often been supposed. Heat-transfer coefficients and heat fluxes comparable to those in present-day liquid-cooled reactors can be achieved. Although the heat-transfer coefficients with gases do tend to be lower than with liquids, for the same relative pumping power, the fuel-coolant temperature difference with gases is not subject to the same limitations as with water, where it can hardly exceed 50 – 100°F (because of the onset of surface boiling). Table 14.1 contains a very rough correlation between the gas pressure and the heat fluxes one may expect to achieve. Obviously the heat flux will depend on other factors as well, such as the properties of the gas, the allowable temperature difference between fuel and coolant, and the geometry of the fuel elements. Nonetheless, experience shows that there does tend to be a rough correlation between the heat flux and the gas pressure for many systems.
2. The tendency of gas-cooled reactors to have large bulk does not necessarily lead to prohibitive capital costs. For sufficiently large power output, that portion of the plant cost that is sensitive to power density in the reactor core can be reduced to the point where it does not exert a decisive influence on over-all costs.[19]
3. Gas-cooling offers the possibility of operation at relatively high temperatures, and there are few other general reactor types that promise as high thermal efficiencies. High efficiency, of course, not only results in reduced fuel costs but also leads to lower installed capacity of equipment per unit of energy generated and consequently may lead to lower capital costs. Cost of power-

---

[19] The ORNL Gas-Cooled Reactor, ORNL-2500, Section 11 (April 1, 1958).

TABLE 14.1

HEAT FLUX OBTAINABLE WITH He or CO_2*
($\Delta T \sim 400°F$)

Pressure psia	$(Q/A)_{max}$ (Btu/hr-ft^2)
300	100,000 – 200,000
600	175,000 – 350,000
900	250,000 – 500,000
1200	300,000 – 600,000
1500	375,000 – 750,000

*Fluxes with H_2 would be roughly twice those given, with N_2 or air, approximately one half.

recovery equipment is reduced, relative to lower temperature systems, also because the operating conditions are closer to, or equal to, those in modern coal-fired plants.

It must be admitted that the natural-uranium-fueled, graphite-moderated reactors of the Calder Hall-type do not possess these characteristics in sufficient degree. There does not appear to be any evidence to induce us to undertake the construction of such plants in the United States. Even the slightly enriched systems that have been suggested as essentially ready for detailed design and construction do not appear to be superior to other reactor types that are more fully developed in this country. The argument in favor of undertaking a major program for the development of gas-cooled reactors, or at least continuing a very serious study of them, rests on the prospect that practical reactor systems of much improved performance may be possible for certain applications, with a reasonable, though not trivial, amount of development.

The approaches that have been suggested for achieving such improved performance are quite numerous. They have in common the difficulty that they all involve more or less substantial extrapolations of existing technology, and that in general the possible disadvantages have been investigated with less care than the possible advantages. Promising possibilities include the following:

1. Fuel elements capable of operating with very high surface temperatures, i.e., 2000°F or higher—

   Most of the suggestions for improved gas-cooled reactors stem from some variation of this approach. They include: (a) graphite-clad uranium-oxide fuel elements; (b) uranium-carbide-impregnated graphite fuel elements; (c) homogeneous cores of graphite, beryllium-oxide, or beryllium impregnated with uranium or one of its compounds, either as a solid block with regular cooling channels, as a bed of small spherical elements, or as a suspension in the coolant; and (d) ceramic fuel elements such as silicon carbide or beryllium oxide.

2. Direct power recovery cycles, which eliminate the need for heat exchange equipment between the heat source and the thermal-electrical converter—

In a non-condensing system, as would be the case with He, N_2, or CO_2, a direct gas-turbine cycle, because of pressure drop in the reactor, intercoolers, etc., will be attractive only if the fluid is heated to 1400 or 1500°F, or even higher, and therefore inherently implies one of the developments listed in (1) preceding. A condensing system, e.g., steam, can be based on materials already at hand, and such a cycle is already under serious study at the Nuclear Development Corporation of America.

3. Improved neutron economy, which may result in very long fuel irradiations, or in negligible costs for fuel enrichment—

In either case, the objective is to cause the fuel cycle costs to approach the absolute minimum set by the cost of raw uranium. These considerations are at the heart of Canada's approach to nuclear power. They lead quite naturally to consideration of D_2O as moderator, and, perhaps less obviously, beryllium as a cladding material. Whether gas-cooled D_2O-moderated reactors will prove to be superior to D_2O-cooled and -moderated reactors, and whether graphite-moderated reactors with Be-clad, slightly enriched fuel will be equally attractive, are questions that cannot yet be answered definitively.

4. High-pressure gas, which may make possible heat fluxes and power densities comparable to the best liquid-cooled reactors now operating—

The principal considerations involved in this approach have to do with the reactor pressure vessel, valves, piping, steam generator, etc. Studies by ORNL and by ACF Industries[20,21] indicate that, to the extent that high pressures imply small cores and relatively small total output, this approach is probably not fruitful. The possibilities have not been

---

[20] A Study of the Design Consideration for High-Pressure Gas-Cooled Reactors with Small Cores, ORNL-CF-58-7-55 (July 31, 1958).

[21] W. S. Farmer et al., Prescoping Optimization for Selection of Enriched Uranium Gas-Cooled Power Reactor Design, ACF-GCPR-101 (February 4, 1958).

thoroughly examined, however, and it may be that longer cores or higher fuel element surface temperatures than those studied so far, might lead to very attractive performance.

5. Hydrogen gas as a coolant—

This is singled out because, while its superior performance is universally recognized, the unique problems associated with its use have led to its rejection in practically all studies of gas-cooled reactors. These problems are difficult, but they do not appear to defy solution, and the matter undoubtedly warrants further investigation.

Having drawn attention to some of the possibilities for high performance with gas-cooled reactors, and in particular to the very respectable heat-transfer performance that may be achieved, it will be well to acknowledge one potentially serious difficulty with gas-cooling, especially for high-performance systems. Since high heat fluxes with gases are generally accomplished with large temperature differences between fuel surface and coolant, comparatively small perturbations in coolant flow and in heat generation in the fuel may result in quite large variations in fuel-surface temperature, so that serious hot spots may result. It may be said that water is a very tolerant coolant, generously allowing (by nucleate boiling) for some ignorance or optimism on the part of the designer, whereas gases are extremely demanding and tend to aggravate any mistake in design or fabrication. This difficulty does not preclude the possibility of high-performance systems, but does place very strict requirements on the design and possibly on the fabrication of fuel elements.

Foreign Market Application. While it is not yet clear what restrictions on reactor design may be imposed by legal requirements or by the attitude of the purchasing government, several points are generally considered to have a bearing on design of reactors for export.

1. There may be a significant advantage in reactors that are fueled with natural uranium, or in reactors in which the first core is slightly enriched, but which may be refueled with natural uranium and recycled plutonium. The incentive, of course, is that the superior performance of slightly enriched reactors may be achieved with a widely available fuel supply. The objection is sometimes

raised that plutonium recycle is not economically feasible unless the total through-put of material is very substantial, e.g., greater than one ton per day. With a fuel exposure of 10,000 Mwd/T, this would imply an installed electrical capacity of three to four million kilowatts in reactors from which the plutonium is to be recovered.

This objection is readily answered by the observation that targets for nuclear power plants greatly exceed 4×10^6 kw at a very early date (Euratom's target is 15×10^6 kw by 1967, possibly 50×10^6 kw by 1975[22]) and that there are already planned, or in existence, in Europe fuel reprocessing plants of the necessary capacity. The more serious objection is raised that fuel refabrication costs may be prohibitive with plutonium recycle. These costs will undoubtedly remain for some time subject to considerable uncertainty, at least as great as that surrounding fabrication costs of uranium fuel elements. There is some indication, from a study by American Standard Company,[23] that the fabrication cost may be substantially greater than for similar plutonium-free elements, without causing the recycle concept to be economically unfeasible. It is worth noting that most of the reactors under intensive development in the United States require enriched fuel, and the question of enriched fuel supply for exported reactors is very general.

2. The possibility or the desirability of exporting helium in large quantities for reactor cooling is certainly open to debate. From the point of view of the United States, this is probably not a serious problem. Based upon a supply requirement of 10^6 scf per year for a capacity of 200,000 kwe, e.g., one helium inventory change per year for the ORNL GCR-2, 10% of an annual helium production rate of 500×10^6 scf would support 10×10^6 kw of installed capacity. This calculation is only intended to show

---

[22] L. Armand, F. Etzel, and T. Giordani, A Target for Euratom, Report submitted at the request of the Governments of Belgium, German Federal Republic, Italy, Luxembourg, and the Netherlands (n.p.) (1957).

[23] L. J. Barbieri, J. W. Webster, and K. T. Chow, Plutonium Recycle in the Calder Hall-Type Reactor, ASAE-S-8 (January 1, 1958).

that a very reasonable amount of helium would be adequate for a very substantial power complex, and that supply problems need not exclude helium for consideration as a reactor coolant. The question of foreign supplies of helium is very much in the dark, but could well have an important bearing on future markets for helium-cooled reactors.

3. The dependence of power cost on power output is likely to be different for different types of reactors. One type may appear to be more advantageous in the range up to 100 Mw(E), another in the range above 300 Mw(E), for example. It is sometimes said that the market in Europe for nuclear plants above 100 – 200 Mw(E) will be very limited. This statement is open to serious question. The rate of increase of power demand in Euratom countries is expected to be comparable to that in this country--roughly a factor of two per decade. By 1975, the installed capacity in these countries will be comparable to the present U. S. capacity, about 150×10^6 kw, and about a third of it will be in nuclear plants. With the concentrated load centers found in Europe, and with the projected rate of installation of nuclear capacity in the Euratom countries of 5×10^6 kwe/yr by 1970, there is every reason to believe that plants of several-hundred-megawatts capacity will find a market.

These arguments presumably do not apply to certain other potential customers, such as the Latin American countries, and it may well be that reactors of quite different types may prove most suitable for installation in these different areas.

Thermal Breeding. It is well known that breeding in thermal reactors is a practical possibility only with the Th-U^{233} cycle. Interest in thorium breeders arises from two incentives, the achievement of low fuel costs, and the recovery of the energy potentially available in thorium. The distinction between these two is in a sense artificial, since the need to tap the energy of thorium is ultimately a reflection of the cost of fossil fuels or other alternative energy sources. The first incentive, however, is customarily discussed in terms of current economic factors.

In this case there is no qualitative difference between a conversion ratio slightly less than one and a conversion ratio slightly greater than one, nor is there any obvious relationship between the performance of the reactor under study and other reactors in the nuclear power system. In the second case, it is supposed that future economic conditions will make it important to recover a substantial fraction of the energy available in thorium. Furthermore, if this energy is to contribute significantly to the over-all demand for energy, as implied by the urgent need to utilize the thorium, it follows that a breeding system must not only produce more fissionable material than it consumes, but it must do so at a rate adequate to supply an expanding power economy. In this case, then, the system is to be judged not only by its prospective power cost, but also by its doubling time, i.e., the time required for the entire system, consisting of reactor, fuel-recovery plant, fuel-fabrication plant, etc., to double the total inventory of fissionable material. Another way of saying the same thing is that if the doubling time is too long, the demand for fissionable material will exceed the supply, the cost will rise, and inventory charges will rise until the reactor designer finds it expedient to design into the system a rate of increase of fissionable material that will bring supply and demand into balance. Since the doubling time for electrical capacity in the United States, in Canada, and in Europe is approximately ten years, and since it is to be expected that nuclear power plants will account for an increasing fraction of the total capacity, it is reasonable to expect that a successful breeding system will require a doubling time no longer than five or ten years.

There is very little doubt that fixed-fuel thermal reactors, based on the Th-U^{233} cycle, can be made to have a conversion ratio greater than unity. It has not yet been established, however, that a heterogeneous reactor can also satisfy the requirement of short doubling time, and it may well be that this requirement, if taken seriously, will distinguish more sharply among various reactor types than will the cost of power as determined by present economic factors.

The difficulty of satisfying this requirement with a heterogeneous reactor can be seen quite readily from a few simple considerations. It can easily be shown that the doubling time of the over-all system depends upon

the conversion ratio, on the specific power of the system in kw/kg of total fissionable material inventory, on the rate of fuel reprocessing, and on losses in reprocessing. In fact, if the total fuel inventory in the entire system is I kg of U^{233}, the rate of increase of fissionable material is given roughly by

$$\frac{1}{I}\frac{dI}{dt} = C(R-1)\,p - \beta\cdot\rho \equiv \lambda,$$

where

$C = 1.15 \times 10^{-6}$ kg/kw day,

R = conversion ratio,

p = specific power, kw/kg,

ρ = fraction of inventory processed per day,

β = fraction of processed material lost.

For a five-year doubling time ($\lambda = 3.85 \times 10^{-4}$/day) and R = 1.15, even if $\beta = 0$, the above relation gives p = 2200 kw/kg. On the assumption that not more than 20 or 25% of the system inventory is in reactor cores, the specific power in the core would have to be 8000 – 10,000 kw/kg, requiring an average neutron flux of about 2×10^{14} n/cm^2 sec. On the other hand, the loss of U^{233} by neutron absorption in its precursor, Pa^{233}, can only be kept low if the fertile material is in a low neutron flux. Since the pile cross section of Pa^{233} is in the neighborhood of 150 barns, and its half life is 27 days, a 1% loss of Pa^{233} by absorption takes place with a neutron flux of about 2×10^{13} n/cm^2 sec. This actually represents a 2% loss of neutrons, however, because of the neutrons captured in thorium that led to the unproductive protactinium. This represents 0.02 of the approximately 0.3 neutrons theoretically available for a net positive breeding gain. Obviously a flux much greater than 2×10^{13} n/cm^2 sec could not be tolerated in the fertile material. It follows that nearly complete separation of the fissile and fertile materials is required if the doubling time requirement is to be satisfied. It is still not clear, however, that the necessary specific power in the fissionable material can be achieved with a heterogeneous system. The difficulty of doing this can be seen by considering the case of uranium oxide-thorium oxide fuel elements. The conductivity and melting point of such elements, together with practical limits of fabricability, limit the power density in the fuel elements to about 1000 – 1200 w/cm^3. The limiting

specific power, in kw/kg of fissile material, is then inversely proportional to the concentration of the fissile component of the fuel. However, to limit the loss in Pa^{233} discussed above, the contribution of the core to the breeding process must be held down, and the Th-U^{233} ratio in the core can hardly be as great as 30 to 1. Thus, the maximum specific power in such elements cannot exceed about 3600 kw/kg, and the average probably not more than half this value. The conclusion is reached, then, that the doubling time requirement cannot be met by a reactor fueled with oxide fuel elements. It is possible that a high conductivity diluent such as graphite can be used as a carrier for U^{233}, so that the required power density can be achieved. The whole question involves also such matters as the blanket processing cycle and inherent delays in the process flow sheet outside the reactor. The whole problem is a very complex one, which has not yet been sufficiently investigated to establish the possibility of satisfying the doubling time requirement with a heterogeneous reactor.

It is possible, of course, to forego the requirement of a five-or ten-year doubling time, and to evaluate high-conversion ratio systems solely on the basis of the promise they have of yielding very low fueling costs. In this event, there is very little choice but to adopt present price schedules, so far as they are established. There is then no special interest in breeding as such, and it may well be that the optimum conversion ratio in heterogeneous thorium reactors will be less than unity. It is certain that it will be less than the maximum achievable. It is probable also that a significant fraction of the fissionable material supplied would have to be U^{235} or Pu. The whole question of the fuel cycle, in this case, becomes simply another aspect of the search for low-cost power in today's market.

14.1.2 Promising Reactor Systems: An attempt is made in this section to discuss those aspects of several promising gas-cooled reactor systems that appear to hold the key to their future success or failure.

Since it will hardly be feasible to pursue the development of all of the reactors that appear at the bar for consideration, it is important in discussing the pros and cons of various reactor types to distinguish between those advantages and disadvantages that are unique to the particular reactor concept, and those that are held in common with other reactor concepts.

High-Temperature Systems. The advantages of high-temperature fuel elements, i.e., those capable of operating at 2000°F or above, are obvious. Very high-temperature differences between fuel and coolant may be achieved, leading to high-heat fluxes and high-power densities, which in turn tend to produce both low capital costs and low fuel inventory charges. Alternatively, the coolant may be allowed to reach a high temperature, perhaps 1400 – 1600°F, making possible the use of a direct power-recovery cycle; with an indirect cycle, much lower steam generator costs are possible, by virtue of high-temperature difference on the gas side. Very high thermodynamic efficiencies are possible; 40% net electrical efficiency is not out of the question, and the savings in capital and fuel costs may be quite significant. While all of these gains may not be fully realized simultaneously, there is no doubt that high fuel-surface temperatures can lead to marked improvement in power plant performance.

Unfortunately, the prospects for achieving such performance with metal-clad fuel elements are not good, and the materials that do have promise for this application, i.e., ceramics and refractories, do not appear to have very good fission-product retention at elevated temperatures. While the extent of fission-product leakage will vary from one material to another, it will be prudent to assume, until convincing evidence to the contrary is at hand, that all of the high-temperature systems will be extensively contaminated with fission products. Since hazards, maintenance problems, reliability requirements, and other questions associated with the use of contaminated coolants ultimately reduce to a matter of cost, a thorough study of the cost of living with contaminated gas-cooled systems is very much needed. It is doubtful that much progress can be made in the evaluation of high-temperature systems until these issues are squarely faced.

Other general disadvantages of high-temperature systems are the increased difficulty and cost of fabrication, the need for costlier materials of construction, and the more complex and costlier design of components that may sometimes be required. These matters are discussed in Section 14.3.4.

Possibly the simplest approach to high-temperature fuel is to replace metal-clad fuel elements in heterogeneous graphite-moderated reactors with somewhat similar graphite fuel elements. The possible forms are quite

numerous, e.g., plates, tubes, rods, fluted cylinders, etc. It is possible that an adequate degree of fission product retention can be achieved by high-graphite density, or by collecting the fission products in an interior void in the fuel elements, connnected to an external storage system. The feasibility of such schemes would have to be established by a thorough test program before they could become the basis for serious reactor proposals.

An advantage of such elements, shared with other graphite-based fuels, is relative ease of fabrication and reprocessing which results from modest dimensional requirements, and from the tractable nature of graphite. A unique potential advantage may be the possibility of bleeding off gaseous and other fission products, without distributing them throughout the coolant circuit. Another advantage for fuel elements of this type may be that the methods for support and handling of the elements, while not the simplest in principle (compared, for example, to the pebble bed) may prove to be the most straightforward in practice.

The pebble-bed reactor, consisting of very many small spheres of fuel-impregnated graphite (or other refractory material), claims the unique advantage that the fuel can essentially be poured into the reactor, like grain into a grain elevator. Such an advantage could be very important. There are real difficulties associated with continuous on-stream loading of large fuel elements, and the time lost in shutdown can be quite serious in batch-loaded systems if the power density is reasonably high, and the fuel cycle correspondingly short. However, it is not at all clear that such spherical fuel elements will be as well-behaved as one would like. At the points of contact between neighboring fuel elements, there will exist temperatures substantially higher than the average surface temperature, as well as high contact pressures. Suggested coatings, such as siliconized silicon carbide may be much more susceptible to self-welding than is graphite itself. With uncoated graphite, there is a real possibility that impurities in the helium may promote the deposition of solid carbon at the contact points. Several balls fastened together in this way could form a solid much more resistant to breakage than any single bond would be, and could result in plugging the unloading mechanism. The chief problems, then, in this potentially attractive system, have to do with fuel handling and with massive contamination of the coolant loop.

The fluidized-bed reactor is similar in principle to the pebble-bed reactor, except that the fuel pellets are sufficiently small and the gas velocity sufficiently high that the particles become suspended in the coolant, and the whole mixture acquires many of the properties of a liquid, such as buoyancy. Many variations are possible, including those in which the solids are carried through the entire coolant loop and those in which the solids remain in the reactor. Generalizations are difficult, therefore, and some of the following comments may not apply to all fluidized-solids reactors.

The stability of such systems appears to be very much in doubt, and their control may be very difficult. Turbulence and sporadic local density changes will lead to unsteady power levels. There will be a tendency for larger particles to break down, producing fines and altering the characteristics of the fluid.[24] The fines must be removed at a rate not necessarily equal to the desired fuel-reprocessing rate.

On the other hand, the possibilities of continuous fuel processing and of high-power density at very modest gas pressure cannot be lightly dismissed, and although at least one study team has reached the conclusion that a fluidized-solids reactor is technically unfeasible,[25] it may be worthwhile to review the matter once more to see whether any development effort is warranted.

Solid "homogeneous" cores do not differ in principle from other fuel-impregnated, graphite structures and therefore may have in common with the others many of the same advantages and disadvantages. Two features characterize a reactor of this type: Nearly all of the graphite in the core is impregnated with fissionable and possibly also with fertile material, and very little support structure is required within the core other than the graphite itself. This might include, for example, large blocks of graphite penetrated by many cooling holes, or flat-plate fuel elements similar to those of the MTR. As distinct from other impregnated-graphite elements, these probably have somewhat greater sensitivity to dimensional instability under irradiation and may pose rather serious problems in fuel handling. In addition, they suffer the disadvantage that all of the core graphite must be reprocessed along with the fuel. This penalty is quite unnecessary from a

---

[24] Feasibility Report on a Fluidized-Solids Reactor System, NEA-5401, Diamond Alkali Co., Foster Wheeler Corp., Pioneer Service and Engineering Co. (January, 1954).

[25] Ibid

reactor physics standpoint, but may be justified by greater heat-transfer surface per unit volume of core. The only apparent advantage of such a system is a convenient and controllable way to maximize the area of heat-transfer surface in the core. However, the choice between a core of this sort and a pebble bed may well be made in the final analysis on the basis of fuel handling techniques, and it is not clear at present which way the practical advantage will ultimately lie.

Small High-Pressure Cores. Preliminary studies have been carried out by ORNL, Kaiser-ACF and GE-Hanford[26,27,28] on the prospective advantages of a more compact core which could be operated at higher gas pressures. The advantages claimed for such a system are: (1) lower uranium inventory requirements, (2) shop fabrication of major components, and (3) more efficient heat-transfer performance. The several studies carried out revealed that with current limitations on operating conditions (e.g., 1000°F gas temperature, 1200°F fuel element temperature, and CO_2 and helium coolants), these designs were not competitive on the basis of lowest net power costs with the GCR-2 design.

A brief summary of some of the results of the ORNL study is shown for purposes of illustration in Table 14.2. Two cases are shown of several that were studied.

The primary difficulty arose from excessive capital costs, although fuel costs were also a contributing factor. The major problems were:

1. When the vessel sizes are limited by the requirement of shop fabrication, the diameter of the reactor could not be greater than 14 ft. Even with a 19-rod fuel element cluster the small core could not approach the power rating of the GCR-2 reactor.

---

[26] A Study of the Design Considerations for High-Pressure Gas-Cooled Reactors with Small Cores, ORNL-CF-58-7-55 (July 31, 1958).

[27] W. S. Farmer et al., Prescoping Optimization for Selection of Enriched Uranium Gas-Cooled Power Reactor Design, ACF-GCPR-101 (February 4, 1958).

[28] A Parametric Study of the Gas-Cooled Reactor Concept, HW-54727 (March 1, 1958).

TABLE 14.2

COMPARISON OF SMALL CORES WITH GCR-2

	GCR-2	Small Core I	Small Core II
Core diameter, ft	30	9	9
Core length, ft	20	14	14
Vessel shape	sphere	cylinder	cylinder
Vessel diameter, ft	50	13	13
Helium pressure, psia	300	1350	1350
Thermal rating, Mw	687	448	421
Net electrical output, Mw	225	133	129
Number of channels	1597	212	107
Number of fuel rods per cluster	7	19	19
OD of fuel slug, in.	0.75	0.31	0.62
ID of fuel slug, in.	0	0	0.31
Maximum heat flux, Btu/hr-ft^2	96,000	660,000	620,000
Fuel enrichment, % U^{235}	2.0	3.0	3.0
Power cost, mills/kwhr:			
Fixed costs, incl. fuel inventory	7.55	8.20	8.12
Fuel costs	1.73	2.30	2.20
Operation	0.89	1.00	1.00
Total mills/kwhr	10.17	11.50	11.32

NOTE: These costs are based on 1958 levels and top charges of 47% of direct capital costs. Escalation or a different schedule of top charges would change the final power cost, but would have very little effect on the comparison.

2. The fuel element arrangement is basically more costly because of the need for greater fuel element surface. In addition, the fuel element "hot-spot" problem would be much more severe.
3. The containment of the high pressure (e.g., 1350 psia) in the system causes a very complex steam-generator pressure-shell fabrication problem as well as a blower-containment problem. These difficulties result in a disproportionate increase in capital costs for the pressure vessels.
4. On the high-pressure side of the system, all vessel and pipe walls must be cooled. A concentric coolant-pipe system is preferred so that the hot-gas piping need carry only the differential pressure between hot- and cold-gas piping. The field assembly problem is quite complex.
5. Provision for thermal expansion between the reactor and the steam generator might be difficult, if not impossible, to accomplish with expansion joints.
6. The high-pressure systems may necessitate additional containment facilities and give rise to more complex core-cooling problems.

In spite of these objections, the basic concept of a small core cannot be discarded. Essentially, this is the direction toward which all advancements in reactor design lead. If gas temperature can be elevated to 1100°F or higher, the equipment can be further compacted. With smaller sizes of equipment the pressure vessel fabrication problem may be less difficult and costs can be further reduced. It thus follows that while the small high-pressure core is not applicable to GCR-2 parameters, an increase in permissible outlet-gas temperatures would require a re-examination of this design concept.

<u>D_2O Moderation</u>. The potential advantages of D_2O as a moderator are well known. Because its moderating ratio, $\xi\Sigma_s/\Sigma_a$, is some 70 times that of graphite or beryllium, and 170 times that of water, heavy water is sometimes called the perfect moderator. (In the above ratio, Σ_s and Σ_a are the macroscopic scattering and thermal neutron absorption cross sections, respectively, and ξ is the mean logarithmic energy loss of a neutron per collision.) Even with the addition of several volume per cent of aluminum or

zirconium to separate the fuel from the moderator, the moderating ratio may remain many times that of carbon. The advantage of a high-moderating ratio is that much greater flexibility is allowed in the distribution of neutrons between fissionable and fertile material, while the moderator absorbs a nearly negligible fraction of the neutrons. As a result, operational characteristics such as long burn-up of fuel can be obtained with natural uranium in a D_2O-moderated reactor, whereas similar characteristics in a graphite-moderated reactor would require appreciable enrichment of the uranium. The chief consequence of this superior neutron economy is likely to be low fuel cost, both because the initial value of the fuel (e.g., natural uranium) may be lower, and because much longer burn-up of the fuel may be achieved before reprocessing (e.g., 10,000 Mwd/T with natural uranium). Indeed, it is probably fair to say that the interest in D_2O as a moderator stems largely from the expected reduction in fuel cycle costs, and that the effect of D_2O on capital costs will not be a major factor in determining its usefulness. Studies at Chalk River[29] indicate that fuel cycle costs as low as 1.85 mills/kwhr may be achieved, as compared to about 2.4 mills/kwhr for a Calder Hall type of reactor[30] and 2.5 mills/kwhr for GCR-2.[31] It must be pointed out that the Chalk River figure is based upon a once-through fuel cycle, with no plutonium credit and no reprocessing charge. Hence, no reliance is placed upon the future development of means for recycling the plutonium.

There are, however, potential disadvantages to the D_2O system which must be weighed along with its merits. The limitations on loss of D_2O from the system are actually a good deal more severe than the comparable limitations on He loss from a He-cooled reactor. In the case of GCR-2, an annual make-up requirement of 10^6 scf corresponding to a leakage rate of about 0.3% per day would add about 0.016 mills/kwhr to the power cost. In a D_2O system containing 250,000 lb of D_2O and producing 200 Mw(E), a leakage rate one-tenth that of

---

[29] The Canadian Study for a Full Scale Nuclear Power Plant, AECL No. 557, Chalk River (January, 1958).

[30] L. Armand et al., A Target for Euratom (May, 1957).

[31] The ORNL Gas-Cooled Reactor, ORNL-2500, Section 11 (April 1, 1958).

the above, or 0.03% per day, would contribute about 0.1 mills/kwhr to the power cost. That is, for equal supply cost, the fractional loss per day of He may be 100 times that of D_2O. This computation is intended only to draw attention to the great importance of maintaining a low loss rate of D_2O.

The principal problems of the D_2O system appear to center around details of mechanical design. The need to separate the moderator and the coolant may lead to quite serious thermal stress problems in a power reactor. Differential thermal expansion between hot headers and cold pressure tubes, or between hot pressure tubes and cold moderator tank walls appears to pose difficult problems. Radial, as well as axial, expansion will present problems if a header is used. In short, it is not yet clear from the designs prepared thus far, that difficult mechanical problems are not inherent in the D_2O system. Such questions can probably be resolved only by considering a number of detailed designs.

14.2 Fuel Element Design

14.2.1 General Requirements: The fuel element is one of the more important components of any reactor system. As the point of origin of the heat produced by the reactor, the fuel element is also the point of highest temperature in the system. It is not possible to allow the peak fuel element temperature as calculated by heat balance to reach the maximum allowable material temperature without exceeding the limiting temperature at some point. Some margin must be allowed for the various factors which cause certain "hot spots" to exceed the average local temperature.

When liquid coolants are used their characteristic high conductivities and volumetric heat capacities tend to minimize the necessary hot-spot allowance. Water, because of its local boiling phenomenon, is a particularly forgiving coolant. Gases, although possessing some desirable properties as coolants, are not so tolerant of designers' errors. When a region of a gas-cooled reactor tends, for some reason, to run hotter than the average, the resulting changes in physical properties will reduce the flow of coolant to that region resulting in further temperature increase. The hot-spot allowances are also related to the temperature difference between the fuel element surface and the coolant. This difference tends to be large in gas-cooled reactors. As a result, gas-cooled reactors, in general, require

large hot-spot allowances. In the GCR-2 design study the allowance was arbitrarily set at 300°F, and subsequent work indicates this to have been a good choice subject to acceptable performance of the fuel element in other respects.

From a standpoint of reactor economics and thermodynamics it is desired to run the fuel elements as hot as possible without exceeding the temperature limits imposed by the fuel element materials. For given materials this means, in effect, reduction of the hot-spot allowance. Since, for power reactors, reliability and long service life are required, the hot-spot allowances must not be reduced arbitrarily but, rather, by a careful analysis and evaluation of the causative agents and subsequent design effort to reduce the hot-spot factors.

The basic geometry of a fuel element has a profound effect on the magnitude of its hot-spot factors and its reaction to them. Six criteria are presented as a first attempt to develop a basis for judging fuel element shapes for gas-cooled power reactors. Item (1) fabricability, is self-explanatory, and items (2) through (6) are all related to hot-spot effects. The following items are not necessarily listed in order of importance: (1) fabricability, (2) surface/volume ratio, (3) structural stability, (4) internal thermal path, (5) external thermal path, and (6) thermal stability of structure.

1. The importance of fabricability is obvious. The difficulty of making a fuel element is directly related to its cost. Since the fabrication cost is an appreciable part of the fuel cost which is, in turn, a major item in the cost of power, the cost of fuel fabrication represents a sizable fraction of power cost. For the GCR-2 study, fuel costs represented 28% of power cost, and 9% of the power cost was chargeable to fuel fabrication. The fabrication details of the fuel element also affect the cost of fuel reprocessing.
2. The surface/volume ratio must be such that the element contains enough fuel from a nuclear standpoint and has enough surface area to permit the heat to be picked up by the coolant within the limits of economically obtainable heat-transfer coefficients.

Surface/volume ratio can be affected by choice of the basic configuration and its size, or by the use of extended surfaces, i.e., fins. Where clad materials are used which have a high neutron absorbtion cross section, there is a strong incentive to obtain configurations which make the utmost use of the cladding surface for heat transfer purposes. In this sense it should be remembered that a given quantity of material in the form of fins will have an effectiveness approaching twice that of the material in the base capsule, since both sides of the fin material are available for heat transfer.

Another aspect of surface/volume ratio occurs in non-homogeneous reactors having low fuel enrichment. The ratio of fuel volume to a hypothetical surface which "sees" the moderator affects the criticality of the reactor by influencing the number of neutrons absorbed in U^{238}. Where criticality is at a premium there is an incentive to minimize the surface which "sees" the moderator.

3. The basic requirement that the fuel element be structurally capable of carrying the loads imposed on it by service is an obvious one, but there are some subtle implications which are easy to overlook. In addition to the static loads imposed by the weight of the fuel element, there are also dynamic loads due to the impact head of the coolant; in GCR-2 these were of the order 0.5 psia. Had they been higher by a factor of two, it would have been possible for the dynamic load to swing the fuel capsules sidewise causing them to "wind up" as shown in Fig. 14.1

With a factor of two margin of safety the GCR-2 element is probably satisfactory from this standpoint. The study, however, showed that lower power costs would result if the reactor heat output were increased from 700 to 1100 Mw(T). With the increased coolant flow at the higher rating the GCR-2 fuel element as designed would be dynamically unstable. This particular difficulty could be overcome by restraining the capsules against swinging sidewise or by reversing the flow of coolant so that the dynamic

UNCLASSIFIED
ORNL-LR-DWG 31566

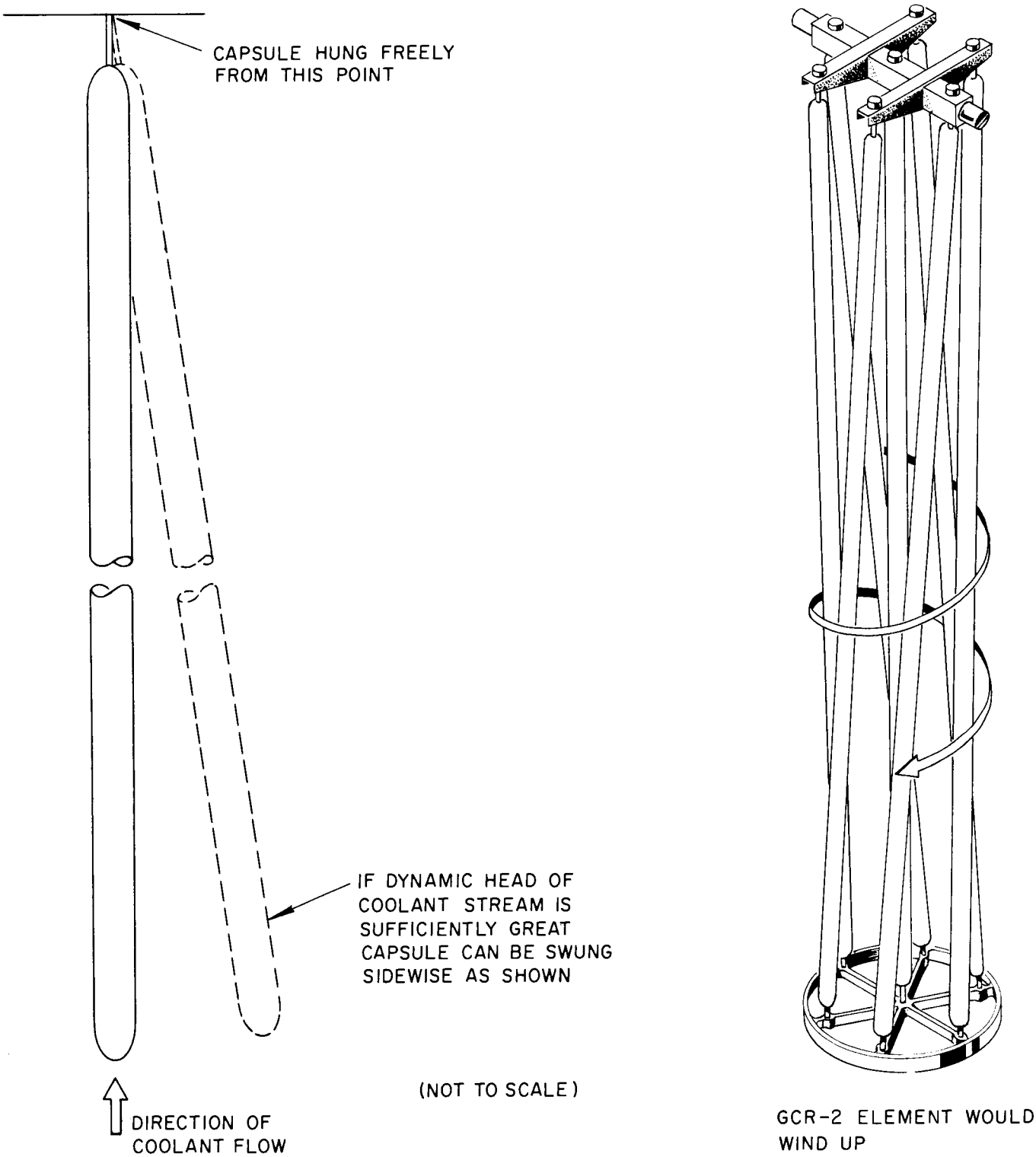


Fig. 14.1. Effect of Dynamic Pressure on GCR-2 Fuel Element.

forces tend to straighten the capsules. These examples are used for illustration only and are not specific recommendations. In the latter case, for instance, there would exist the possibility of an induced flutter.

4. The nature of the internal thermal path influences the temperature structure within the fuel body. This, in turn, determines the maximum internal temperature and the thermal stresses which will exist in the fuel element. With fused UO_2 fuel, differences of opinion exist as to the maximum allowable material temperature and the seriousness of thermally induced cracks in the fuel body. Some limits, however, must exist. For UO_2 fuel an acceptable peak internal temperature seems to be expressed by the equation:

$$T_{max} = T_{wall} + 0.9(T_{melt} - T_{wall})$$

With regard to thermal cracking, designs are to be avoided in which the temperature pattern tends to induce "spalling" which would produce small particles having a tendency to "rachet" with temperature cycling. Gross cracks resulting in large particles which remain in place are not considered serious.

The point-to-point temperature differences existing in a given fuel element are directly proportional to power density and inversely proportional to the heat conductivity of the fuel. Uranium dioxide is a poor heat conductor. Its specific conductivity is approximately 5% that of stainless steel or graphite. If a fuel material were available with higher conductivity by even a factor of two, the benefit would be great. Hence, there is considerable incentive in investigating other materials, such as uranium carbide, uranium-uranium oxide, or uranium oxide-graphite mixtures.

5. The external thermal path is the result of the shape of the coolant passage which surrounds the fuel element. It influences the variation of fuel element surface temperature at a given location in the reactor. A less obvious but nevertheless real effect is to increase the <u>average</u> fuel element surface temperature over that calculated by a running heat and material balance along a coolant

channel. This effect was brought out in an analysis of the temperature structure of the GCR-2 fuel elements[32] and is due to a passage shape efficiency factor which is analogous to fin efficiency in solid conductors. For configurations which have azimuthal symmetry or a uniform temperature around the periphery of the individual elements this factor would be 1.0, and the surface temperatures at a given location would be constant at the theoretical value calculated by heat balance. Three-dimensional configurations will give a factor less than 1.0. The best spacing of the GCR-2 capsules relative to channel walls was calculated to give a passage efficiency factor of 0.85 with a resulting surface temperature variation of ± 100°F from the mean.

The external thermal path of the GCR-2 fuel element configuration forces the mean fuel element surface temperature to run approximately 100°F lower than could be allowed with a two-dimensional configuration, keeping the same maximum hot-spot temperature limit (i.e., 1500°F for GCR-2) and the same allowance for hot spots from other causes. Since the maximum heat-balance temperature for the GCR-2 fuel element was 1200°F, a two-dimensional fuel element could be run at 1300°F. The GCR-2 study indicated the minimum power cost at 1200°F to be 9.6 mills/kwhr while the minimum cost at 1300°F was 9.0 mills/kwhr. The reduction in power cost results only in part from an increase in total power output. The fuel-fabrication costs amounted to approximately one mill/kwhr. We could, therefore, pay up to 60% more for fuel fabrication in order to obtain fuel elements with a more efficient external thermal path.

The effect of the external thermal path is influenced by the thermal conductivity of the fuel. Surface-temperature differences tending to exist because of details of the external thermal path would be dampened by fuel having high conductivity and in good thermal contact with the inside of the capsule.

---

[32] L. G. Epel and W. T. Furgerson, Temperature Structure in Gas-Cooled Reactor Fuel Elements and Coolant Channel, ORNL-CF-58-5-97 (May 27, 1958).

The effect of mechanical misalignment, both of the individual components of the element relative to each other and of the fuel element assembly relative to the channel, must be considered. It is expected that any fuel element configuration will be sensitive to misalignment in some degree. It must be determined which designs are least sensitive. A study of this point is currently under way. Pending results of the study, it can be postulated that if the fuel element design is such that some point on the surface runs at higher than average temperature, that point will be more sensitive to misalignment than the remainder of the fuel element. It follows, then, that the configurations least sensitive to misalignment are those which either have no points at higher than average temperature, or if such points exist they are so located that the adjacent layer of coolant is not subject to "pinching" by misalignment. An example of the first case is a two-dimensional element which has no surface-temperature variation. An example of the second case is a single longitudinally finned cylinder in a cylindrical channel. The hottest region, i.e., the capsule wall, is farthest from the channel wall and least sensitive to misalignment of the assembly in the channel.

6. The problems of thermal stability of fuel and capsule materials are universally recognized. It is not so widely realized that certain mechanical shapes are inherently stable, i.e., tend to retain their shape under fuel element operating conditions, and some are inherently unstable. This is a property of the shape rather than the material, although material properties can modify the effect.

Consider an MTR-type fuel element consisting of a series of parallel slabs, spaced at the edges, with the coolant flowing longitudinally, as shown in Fig. 14.2a. If, for some reason, the temperature in one coolant channel runs hotter than its neighbors, the adjoining slabs will deflect thermally so that their hotter sides are convex. Since the slabs are spaced at both edges, this results in their movement toward each other, as shown in Fig. 14.2b.

UNCLASSIFIED
ORNL-LR-DWG 31567

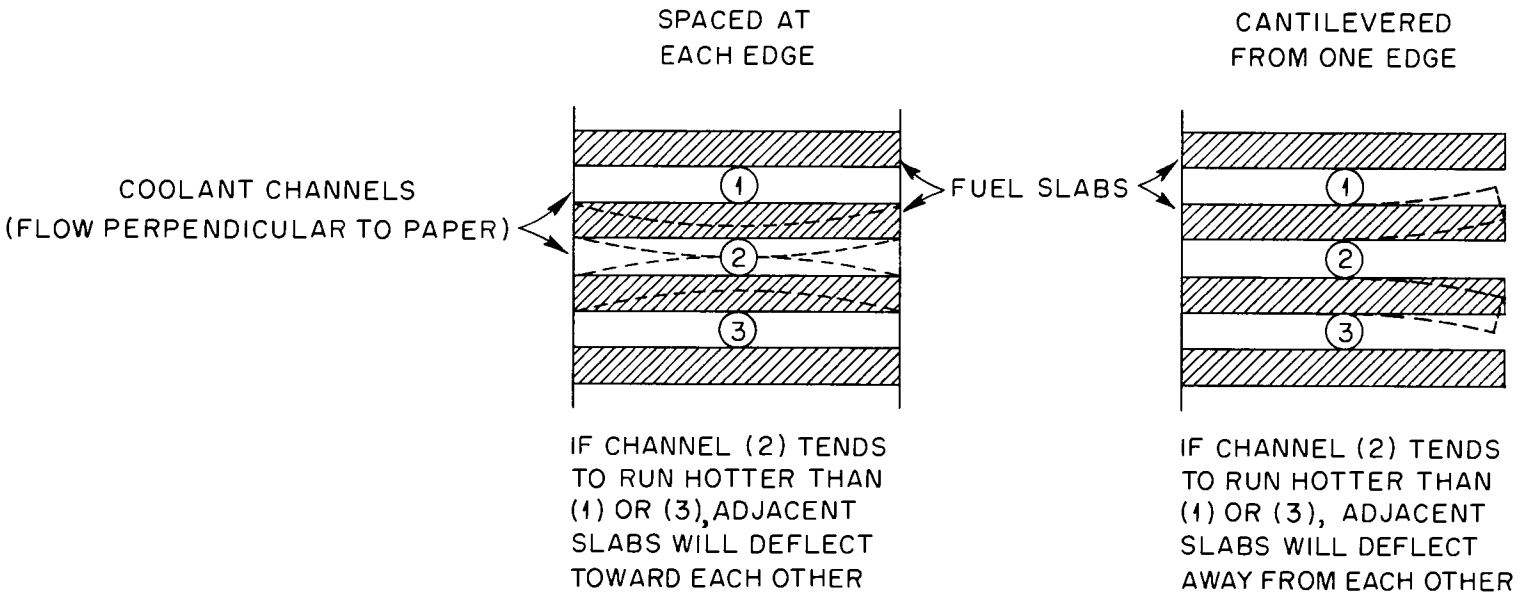


Fig. 14.2a

Fig. 14.2b

The flow passage is "pinched" and therefore suffers a further temperature increase giving more deflection, etc. Such a configuration may be said to have a "positive feed-back". It is possible to change this into a "negative feed-back" so that thermal distortion tends to open the hot channel by simply removing the spacing at one edge and supporting the fuel slabs as cantilevers from the other edge, as shown in Fig. 14.2b. Adjacent fuel slabs will then deflect away from a hot channel. The feed-back effect, either positive or negative, may be insignificant if the fuel elements are short, if the materials have low coefficients of thermal expansion, or if the coolant-passage configuration is such that the fuel element surface temperature is insensitive to deflection. In the other extreme, the feed-back can be divergent if the surface temperature differences brought about by a deflection are greater than the original temperature differences which produced it. Then, if the feed-back is negative the fuel elements will "hunt"; if it is positive they will continue to deflect in the original direction until they are restrained in some way, probably by contact with other parts of the fuel element-channel assembly. On the basis of the temperature analysis of CF-58-5-97, it is strongly suspected that the GCR-2 fuel elements as described in ORNL-2500, Section 4, will be in the latter category.

It should be noted that stiffening the fuel element by increasing the moment of inertia of the capsule cross section is not effective in decreasing the deflection of a fuel element of a given length which is subjected to a given transverse temperature gradient. The moment of inertia of the section influences the stiffness of the capsule and the moment tending to deflect it in the same degree, and the two effects cancel.

The use of spacers to restrain thermal deflection of fuel element members is regarded with extreme suspicion. If the forces which would produce deflection in an unrestrained assembly still exist something must yield. When it is remembered that the capsule walls are approximately 0.020 in. thick and are at temperatures up

to 1500°F, it is not difficult to imagine what part it will be. Experience at the Oak Ridge National Laboratory indicates that a "soft" structure was required rather than a stiff one. The principal difference between the radiator and a reactor fuel element is that in the former the maximum temperature is limited to the temperature of the heat-transfer medium.

All of this discussion has been based on consideration of steady-state temperatures. Variation of temperature with time is to be expected when the reactor is undergoing changes in power level, and high-frequence variations can occur due to local non-steady flow of coolant even though the reactor is operating at constant power. Support members which are transverse to the direction of coolant flow and certain types of fuel element configurations will give rise to trailing vortices which can result in time variations of the local heat-transfer coefficient of approximately ± 25% of the time average value. The Calder Hall-type transverse-fin fuel element is an example of those types in which time variations of temperature exist. These are apparently satisfactory from a standpoint of thermal cycling at Calder Hall conditions. It is by no means certain, however, that this fuel element design could be extrapolated to the higher temperatures and heat fluxes using the lower conductivity capsule materials being contemplated in the ORNL gas-cooled reactor studies.

The basic fuel element configuration for GCR-2 was chosen primarily from the standpoint of fabricability. The surface-to-volume ratio requirements were met by allowing the diameters of the capsules to vary, and structural requirements were limited to consideration of weight loads. The internal thermal path was considered briefly, and the external thermal path and structural thermal stability were not considered at all.

The resulting fuel element design rates high from a standpoint of fabricability; it is satisfactory with regard to surface-to-volume ratio and internal thermal path; structural suitability is marginal, and much is to be desired from a standpoint of external thermal path and thermal stability.

If the problem of fuel element design is approached with all the requirements in mind, it is possible to examine various shapes and apply relative ratings on the six points of consideration.

In Table 14.3, an attempt is made to begin such a comparison. The ratings 1, 3, and 5 mean good, marginal, and bad, respectively.

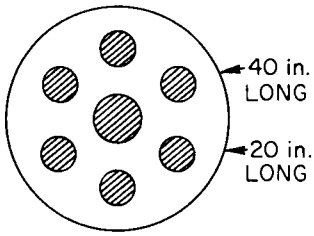
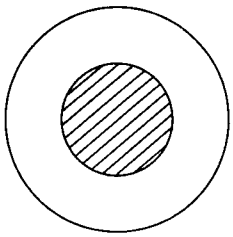
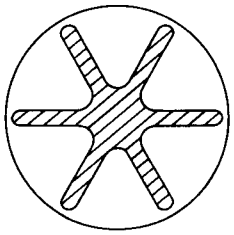
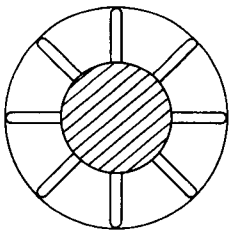
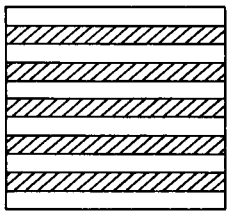
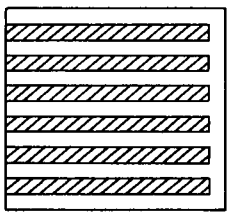
The seven-capsule, cluster-type fuel element chosen for the GCR-2 has been studied analytically to determine the temperature structure which will exist both within the fuel and in the coolant gas stream. It was found that circumferential variations of capsule-surface temperature occurred which could be expressed by the dimensionless group, $\Delta TK/\Phi r$, which was in turn, a function of capsule radius, capsule spacing, and channel size. In this expression ΔT is the difference in temperature between the innermost point of a peripheral capsule and a point diametrally opposite, K is the "total" conductivity of the coolant gas, i.e., molecular plus eddy diffusivity, Φ is the heat flux and r is the capsule radius.

In a channel of a given diameter the location of the peripheral capsules can be adjusted to minimize the temperature variation on the capsule. This minimum variation for conditions typical of GCR-2 was approximately ± 100°F. The temperature structure was found to be quite sensitive to capsule location in the array. A movement of 10 mils from the point of minimum-temperature variation can be expected to increase the temperature difference across a capsule by approximately 20°F. A 20°F temperature difference maintained throughout the length of a 40-in.-long capsule will result in a maximum deflection at the mid-length of approximately 50 mils toward the hot side. Even though it is not expected that this temperature difference would be maintained along the entire length of the capsule, it can be seen that there is considerable danger that the fuel element configuration proposed for GCR-2 would be divergent from a standpoint of thermal stability. Any asymmetry in coolant flow or fuel element geometry could result in temperature asymmetries which would deflect the affected capsules in a manner tending to further increase the temperature asymmetries until finally the capsules would touch each other or the channel wall. The estimates are based on a heat flux of 18 $Btu/ft^2 \cdot sec$. An increase in the heat flux would increase the temperature variations proportionately. Also, the total conductivity, K, was calculated on the basis of helium at a Reynolds number of approximately

UNCLASSIFIED
ORNL-LR-DWG 31483

TABLE 14.3

REQUIREMENT:

CONFIGURATION	(1)	(2)	(3)	(4)	(5)	(6)	REMARKS
 40 in. LONG	1	1	3	1	4	5	RODS LOCATED AT OPTIMUM RADIUS SUSPENDED FROM ABOVE, SPACED TOP AND BOTTOM
20 in. LONG	2	1	2	1	3	3	
	1	2	2	1	1	3	DOWN-RATED ON (2) FOR NUCLEAR REASONS
	2	1	1	2	2	1	FINS CARRY FUEL
	2	1	1	2	3	1	FINS FOR HEAT TRANSFER ONLY
	2	1	1	1	1	2	SLABS SPACED AT EACH EDGE
	2	1	1	1	1	1	SLABS CANTILEVERED FROM ONE EDGE

6×10^4. If carbon dioxide were used at a Reynolds number of 5×10^5, as determined by heat balance, the resulting temperature differences would be increased by a factor of approximately 1.2.

Much remains to be done in the development of a satisfactory fuel element configuration. As mentioned in ORNL-2500, Section 6.2, shortening the fuel element will decrease the deflection resulting from a given temperature difference; this may also be achieved by decreasing the circumferential temperature difference likely to occur. However, the sensitivity of the seven-capsule configuration to asymmetry appears to be such that decreasing the length by a factor of two may not completely remove the danger of temperature divergence; further decreases in length would be undesirable from a standpoint of the extra poison introduced in the form of capsule ends and support brackets and the additional fabrication costs involved. Temperature differences may be decreased by use of "turbulator" devices to mix hot and cold coolant strata.

Another result of the calculation of temperature structure was the determination of the existence of a shape factor resulting from the coolant-passage cross section details; this is analogous to the fin efficiency obtained with extended heat-transfer surfaces. Whereas, a heat balance along a channel assuming a symmetrical fuel element would give a certain "average" fuel element temperature at a given axial location, the average temperature obtained from the temperature structure calculations was systematically higher. The heat-transfer equation may be written $\Phi = \eta_s h(\overline{T}_w - \overline{T}_c)$, where Φ is the heat flux, η_s is the shape factor, h is the heat-transfer coefficient, $\overline{T}_w$ is the mean wall temperature, and $\overline{T}_c$ is the mixed mean coolant temperature. The value of η_s for the configurations studied varied from 0.6 to 0.9.

A basic conclusion to be drawn from the temperature-structure calculations is that an ideal fuel element should, in addition to satisfying the obvious requirements of surface-to-volume ratio and short thermal path, be as generally symmetrical as possible in either polar or rectangular geometry. Good examples are single rods or single annulae in circular channels or flat plates in rectangular channels. The possibility of improving the seven-capsule cluster configuration, which is symmetrical about three axes only, is suggested if one envisions seven single rods in separate channels located on

the corners and center of a hexagon. If the hexagon is made smaller until the channels begin to merge, the resulting configuration, as shown in Fig. 14.3, will have a greater degree of symmetry than the circular channel.

The foregoing discussion is based on the results of numerical calculations and is subject to confirmation by experiment. It is hoped that by this means the experimental effort can be directed into the most profitable channels.

14.2.2 Hazards and Maintenance Considerations in Advanced Gas-Cooled Reactors: The GCR-2 design was sufficiently conservative and inherently safe that an over-all containment vessel was not required. This is due in part to the double containment barrier effected by the fuel capsule-coolant circuit combination, and in part, to the relatively low activity level in the gas stream. Because of this latter feature, it was deemed feasible to plan for direct maintenance on components of the coolant circuit external to the reactor. These considerations of maintenance and containment are essential to a complete hazards evaluation of any new system. It is quite conceivable that maintenance access and plant safety may be over-emphasized and thus cause an excessive capital cost penalty in the evaluation. It is pertinent for the GCR-2, as well as for any advanced reactor system, to determine the relationship between hazard and maintenance considerations on the one hand and capital costs on the other. The resolution of questions such as this is at this stage of the technology primarily a matter of engineering judgement and cannot be firmly established until a substantial body of test and operational experience has been accumulated.

The hazards philosophy adopted for the GCR-2 was that two independent and improbable events, i.e., simultaneous leaks in fuel element capsule and coolant-gas circuit, either of which is immediately apparent to the operator, are required to produce a hazardous situation. The maintenance philosophy was to hold the gas-system contamination to such a level that maintenance work would be feasible on the entire system external to the reactor within a short time after shutdown. The number of contaminated areas to be serviced would be minimized and only the reactor core would require remote servicing equipment.

Although both of these criteria are difficult to evaluate in terms of the economics of the plant, the designer does have at his disposal a number

UNCLASSIFIED
ORNL-LR-DWG 31455

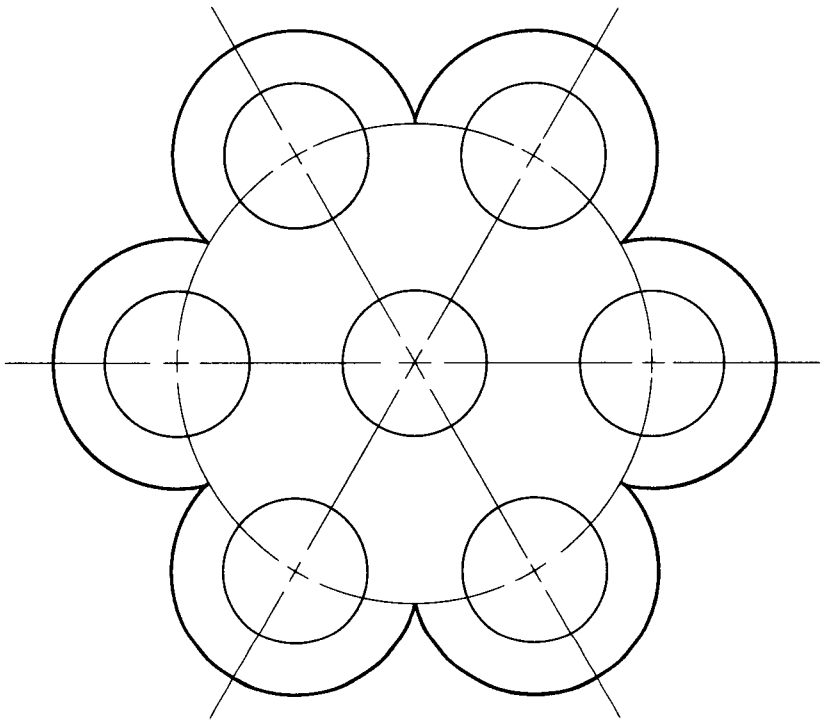


Fig. 14.3 Advanced Fuel Element Design Proposal.

of alternative means of improving the design with respect to each of these objectives. An evaluation of maintenance problems may develop some feeling for the costs involved, but where remote or semi-remote maintenance of radioactive systems is involved, the cost estimates tend to be more uncertain than those for the construction of nuclear plants themselves.

The various reactor systems may be loosely divided into two categories, "clean" or "dirty", depending upon the activity level of the coolant. A clean system may contain radioactivity, but the level is sufficiently low that shielding of the external coolant loop is not required and direct maintenance of the external coolant loop is possible. A dirty system is one in which the activity level is allowed to build up to such levels that shielding is required and direct maintenance a practical impossibility. It is not always a clear-cut issue, however, to decide into which category to place a particular reactor system. In this discussion the effects of the two approaches on reactor design are outlined.

In low-contaminated, or clean systems, nearly all of the fission product activity is confined within the fuel element. In the GCR-2, for example, the stainless steel capsule retains the fission products. The coolant and coolant-loop materials were selected in part so as to minimize the activity in the coolant. Although some such activity is expected, the degree to which it can build up is limited by the purification system to levels which permit direct maintenance on the coolant loop external to the shield soon after shutdown.

With the advent of new design concepts, a favorable situation such as exists within the GCR-2 might not be practical and the effects of various design changes which would help maintain the system "clean" deserve consideration. These include:

1. Provisions for venting fission product gases so that the fuel element capsule need not hold pressure.
2. An enlarged absorption system in which fission products released from the fuel could be removed from the coolant system.

Adaptations of these principles might permit use of ceramic fuel element capsules which currently are not considered reliable for containing gaseous activity. At present, it is not clear how such a system might complicate the reactor design. Several arrangements must be examined in order to

determine the technical and economic feasibility of these schemes and their effects on the hazards containment provisions.

14.2.3 Systems with Unclad Fuel Elements: The alternative to a clean system is a "dirty" or contaminated one. The first question to be answered is, what level of contamination would require an additional containment vessel. The second question is whether there are virtues in permitting this condition to exist. Some advantages that can be visualized are an increase in the reactor output, reduction of fuel costs, as well as the reliability requirements of fuel elements, and elimination of the fuel element leak detection and coolant-purification systems. In the conservative GCR-2 design it was assumed that only limited contamination would be permissible. The penalties for relaxation of these standards are yet to be examined.

A "dirty" system may be defined as one in which the total activity level of the primary coolant is from 10^4 to 10^8 curies total. The unclad-fuel GCR, the pebble bed, and the fluidized fuel or dust reactors imply this condition. In each case one must pay a penalty in additional containment and equipment for decontamination and/or remote maintenance. Thus, a much larger portion of the plant becomes highly contaminated and induced activities in the coolant loop and power-recovery equipment require supplementary shielding. In each case the penalty must be weighed against improved reactor performance. Some of the gains to be realized are: (1) lower fuel costs, (2) lower capital charges, (3) increased gas-outlet temperature, and (4) the elimination of the fuel element "failure" problem.

The lower fuel costs can be achieved because:

1. More fuel element surface area can be utilized; the fuel surface temperature can be increased; and the temperature rise through the reactor can be increased; thus, greatly increased specific power is possible. This results in a lower fuel inventory charge.
2. The elimination of the fuel element cladding may improve the neutron economy and result in longer fuel element life and lower enrichment. Thus, the fuel inventory cost is reduced, and due to the higher conversion ratios which may be attainable, a reduction in burn-up charge is possible.

3. A reduction in fabrication costs may be expected by the elimination of the cladding.
4. Higher gas temperatures possible in these systems will yield more efficient steam conditions with higher net energy output for the same fuel burn-up.

The lower capital charges can be achieved because:

1. Higher power densities resulting from the greater fuel surface area and higher permissible surface temperatures will allow substantial reductions in plant capital costs per kilowatt of capacity.
2. Some savings can be realized by the elimination of the gas-purification system and the complicated and expensive burst-fuel-element detector system.

The increased gas-outlet temperature would reduce the installed unit cost of blower capacity and would decrease blower unit power consumption as well as improving plant thermal efficiency. And finally, the problem of accommodating fuel element failures would be essentially eliminated since the entire plant would be designed to contain and shield against all of the activity resulting fission, particularly that due to fission products released by the elements directly into the coolant stream.

Inasmuch as the ultimate advantage anticipated from the contaminated plant is the reduction in power cost, the evaluation of the factors outlined above could only be accomplished through detailed engineering studies of a number of possible reactor plants. Such a study would require careful consideration of the following items:

1. The design of suitable fuel element configurations which take full advantage of the contaminated coolant concept.
2. A complete reappraisal of the reactor physics and heat-transfer characteristics of the plant based on these configurations.
3. The estimation of the activity and dose levels in and around the primary-coolant circuit.
4. The determination of the shielding requirements to achieve tolerable conditions for plant maintenance and servicing, including an analysis of decontamination and remote handling facilities.

5. The determination of plant designs to accommodate these requirements.
6. The estimation of plant and power costs.

It is expected that in undertaking a program of this nature, it would not be necessary to carry out a complete and new reactor design for each fuel element configuration, but that in many instances the essential engineering features could be obtained by performing perturbation analyses of various available reference plant designs.

14.3 Power Recovery

14.3.1 Steam System: It is customary in discussing nuclear power plants to refer to the power recovery portion of the plant as the "conventional" portion. While there is a clear incentive to utilize standard turbines, condensers, and so forth, to as great an extent as possible, it is important to recognize that the steam-raising part of a gas-cooled reactor power plant is not conventional at all. There is probably no directly comparable industrial equipment. The variety of possible steam-generator designs is great, while the relevant experience on which to base an evaluation of the possibilities is meager. The list of steam-generator types that suggest themselves for serious consideration includes the following:

1. Once-through boiler without reheat: Principal virtues are relative simplicity of design and construction, and minimum number of pressure vessel penetrations and tube-to-header welds.
2. Once-through boiler with reheat, or dual-pressure cycle: The incentive is improved cycle efficiency obtained at the expense of more complex configuration, more vessel penetrations and larger pipe runs.
3. Supercritical Boiler: Very little experience with supercritical steam systems has been accumulated. Two plants (Philo and Eddystone) are presently operating. Material problems at high pressure are of great interest. A higher probability of failure exists and the amount of stored energy in the high-pressure steam is very great. The question is again one of capital costs versus efficiency.

4. Loeffler system (bleeding off enough steam to evaporate feed water): The possibility of using the reactor as a superheater, replacing expensive steam generators with less costly one-fluid evaporators, exists.

Significant differences in capital cost among the several possible systems are already apparent. It is not yet clear, however, which type of steam generator may prove to be most advantageous for a given reactor when all factors are taken into account. It appears that analytical and design studies directed explicitly to the steam-raising part of the plant will be helpful in identifying and reducing the areas of uncertainty in steam-generator design, and hence in the evaluation of advanced power plants. Uncertainties include the following points:

1. Effect on steam-generator size, performance, and cost of tube-support structure; degree of utilization of internal volume consistent with limitations on tube-bending radii, thermal stresses, etc.
2. Effect of increase in gas or steam temperature on required properties of materials and cost of materials and fabrication.
3. Effect on steam-generator design, performance, and cost of requirements for reliability, repairability, and ease of maintenance, particularly as affected by: (a) level and nature of radioactivity in the coolant, and (b) urgency of the need for keeping water vapor out of the coolant. Assuming the long-term objective is to develop designs for which ASME and state code groups will extend code coverage, the inspection problem becomes significant in terms of activity in the system. If inspection is prohibited by radioactivity, a gradual derating of the plant dictated by the code as an alternate for periodic inspection could be disastrous in terms of mills/kwhr.
4. Effect of coolant-steam temperature difference on control aspects of the steam plant. Large design ΔT implies large swings of steam or coolant temperature with changes in load. While equilibrium conditions can undoubtedly be taken care of quite readily, the transients may impose severe requirements on the control system.

The steam generators represent an appreciable fraction of plant cost. Therefore, progress on evaluation of advanced gas-cooled reactor power plants will require at least a partial resolution of these and other problems associated with the steam equipment.

14.3.2 Selection of Coolants and Interchangeability: Inasmuch as the first gas-cooled reactor which is built in the U. S. will be operated largely to gain engineering experience, there is the possibility of using the plant to study the influence of coolant choice on performance and maintenance requirements. Some aspects of this matter were discussed previously.[33]

If two or more gases are to be used interchangeably in a reactor system and if rated power and temperature conditions are to be maintained without alteration of equipment, it is necessary to satisfy the following requirements:

1. The alternate gas or gases must produce the same heat-transfer coefficient in the reactor, and further, must result in temperature variations in the coolant channels which are no greater than those existing with the original coolant. The channel-diameter ratios calculated in Section 15.2.1 give an indication of interchangeability in this respect.
2. The volumetric heat capacity of the alternate gases must be the same as the original gas. This requirement is imposed by blower capacity. The molar heat capacity can be used as an index of interchangeability under this requirement. It should be noted that a degree of control can be exercised over this variable through control of system pressure, although limits are imposed by pressure vessel structural design and the effect on heat-transfer coefficient.
3. It is necessary that pumping-power requirements stay within the limits imposed by the blower drive motors. The pumping-power ratios previously calculated are indicative of interchangeability on this point.

The three parameters referred to above are listed for several possible coolant gases. All values are referenced to helium. Gases having the same value for all three parameters would be interchangeable. The comparison is

[33] The ORNL Gas-Cooled Reactor, ORNL-2500, Section 6.1, (April 1, 1958).

rough; however, a final answer on interchangeability would require a much more detailed examination. (See Table 14.4) It can be seen that the only gases which are interchangeable without adjustment of the system are the three common diatomic gases which are listed together. Others might be adapted by suitable system modification. Hydrogen, for instance, might be used in a system designed for helium provided new blowers having lower volumetric capacity and head were installed. Note that the head and volume changes are not such that they can be obtained by changing the speed of the helium blower; a new blower design is called for. The use of hydrogen in this manner will result in increased performance over that obtained with helium, but it will not develop the full potential of hydrogen. Similarly, by changing the blower and drive motor, helium could be used in a system designed for hydrogen. In this case, the reactor would have to be operated at a power level below that of a system designed specifically for helium. Carbon dioxide and water vapor show some promise of interchangeability if blowers and fuel elements are changed. The use of carbon dioxide in a helium system would necessitate changing blowers, drive motors, and fuel elements. Because of the lower heat-transfer coefficients of carbon dioxide, it would also be necessary to derate the reactor or use fuel elements of greater surface area in order to stay within temperature limitations on the fuel element capsules.

For small reactor systems where the required blower capacity was not greater than ~ 10,000 cfm each, it might be advisable to use positive displacement blowers if coolant interchangeability is contemplated. In these machines the head is independent of the revolutions per minute; and volumetric capacity can be changed simply by changing speed.

It is important to point out that the thermodynamic properties of the gases most suitable for reactor coolants are such that an attempt to design a gas system capable of operating with either of two gases would necessitate compromises, and the resulting system would not have the thermal efficiency obtainable in a system designed for a specific coolant.

14.3.3 Gas-Turbine Applications: Whenever it is proposed to remove heat from a reactor with a gaseous cooling medium, the question of applicability of gas-turbine-power cycles arises. Gas-turbine power plants fall into three basic categories: the open cycle, the semi-closed cycle, and the closed cycle. These are illustrated on Fig. 14.4. Any of these can be

TABLE 14.4

INTERCHANGEABILITY OF VARIOUS COOLANT GASES*

	Helium	Hydrogen	Carbon Dioxide	Air, Nitrogen, or Carbon Monoxide	Water Vapor
D/D_h	1.0	1.077	0.87	0.85	0.79
MC_p/MC_{ph}	1.0	3.53	4.65	2.97	3.63
W/W_h	1.0	0.114	2.6	7.0	2.6

*Where D is the equivalent channel diameter, M is the molecular weight of the gas, C_p is the heat capacity, and W is the total pumping power required.

UNCLASSIFIED
ORNL-LR-DWG 31456

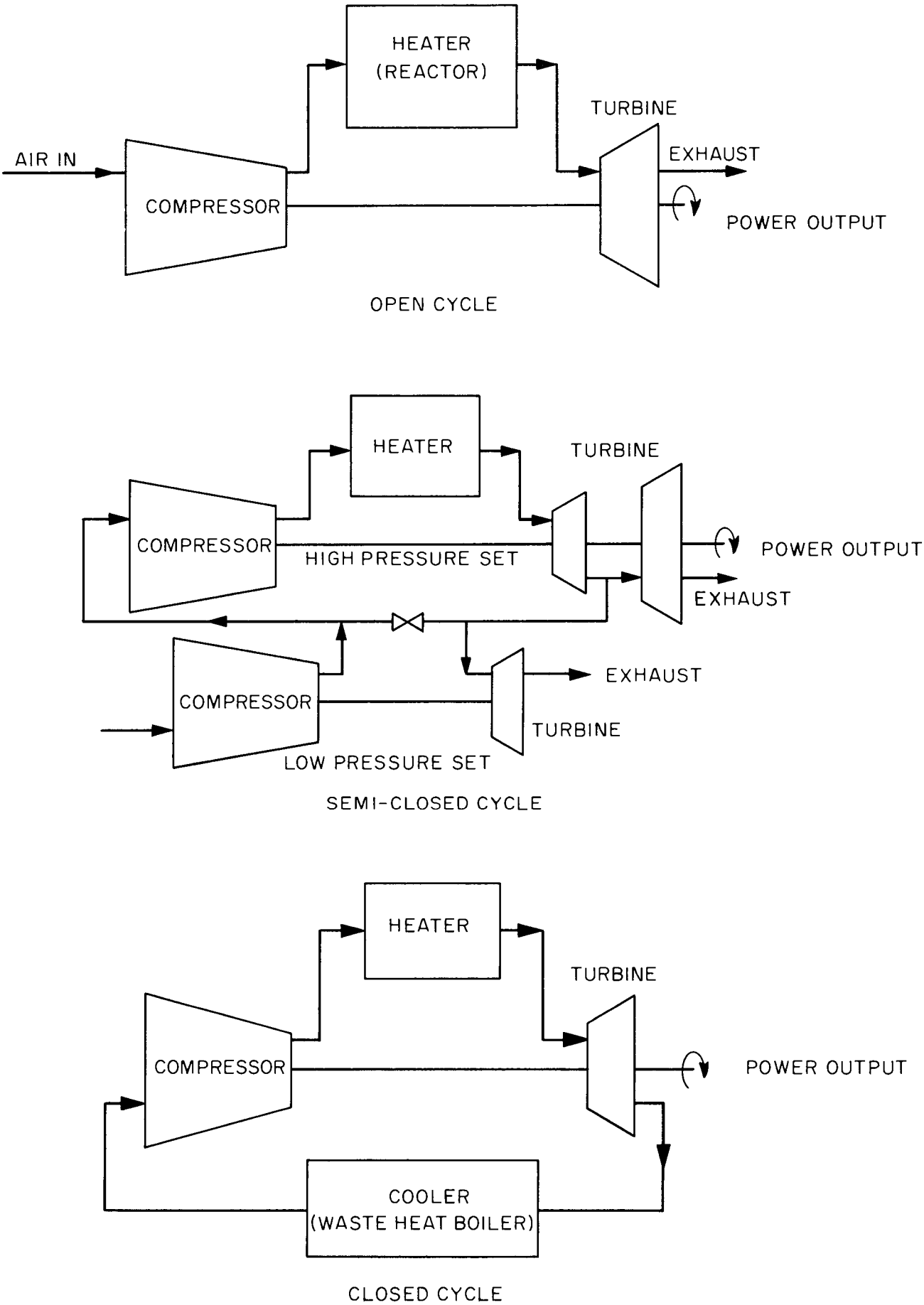


Fig. 14.4 Gas Turbine Cycles

modified by use of inter-coolers, regenerators, and waste-heat boilers to improve the cycle efficiency. The open cycle has the advantage of greatest simplicity since the basic cycle requires no heat exchangers. It is less efficient and less flexible in meeting demand power than the other cycles, however, and is limited to air as a working fluid. The semi-closed cycle is somewhat more efficient and more flexible, but it is still limited to air as the working fluid. The closed-cycle gas turbine permits a choice of working fluids and is able to operate efficiently over a wider range of loads than any other power plant with the possible exception of hydro-electric plants. Since the closed cycle permits control of the coolant density independently of the compressor-pressure ratio, it is possible to optimize the latter to obtain highest cycle efficiencies.

Cycle efficiencies are dependent on turbine-inlet temperature as shown on Fig. 14.5. The two curves show approximate efficiencies for the basic closed cycle with and without recovery of waste heat from the turbine exhaust using helium as a working fluid. It may be seen that, at gas temperatures in the GCR-2 range, i.e., 1000°F turbine-inlet temperature, the gas turbine has little to recommend it over a conventional steam cycle with the coolant circulated by blowers. At this temperature approximately half the power output is through the waste-heat steam cycle. As temperatures are increased, the cycle efficiency is improved, and a smaller fraction of the power output is involved in the waste-heat cycle.

Gas-turbine machinery is currently limited to approximately 20 Mw output; whereas, steam turbines of ten times this capacity are not uncommon. The aerothermodynamic technology for larger machines is available, but the field is relatively new, and the demand for larger units does not presently exist.

In addition to flexibility in meeting power demands the gas turbine has another advantage when coupled to a reactor as a heat source. Preliminary calculations indicate that by going to high system pressures and by using appropriately designed fuel elements, the power densities in gas-cooled reactors can be increased over those of the GCR-2 by a factor of approximately five. It is more difficult, however, to reduce the size of the steam generators. This raises the problem of construction of pressure shells 15 ft or more in diameter to contain pressures over 1000 psia. Use of a gas turbine alleviates the problem in two ways: first, by decreasing the heat

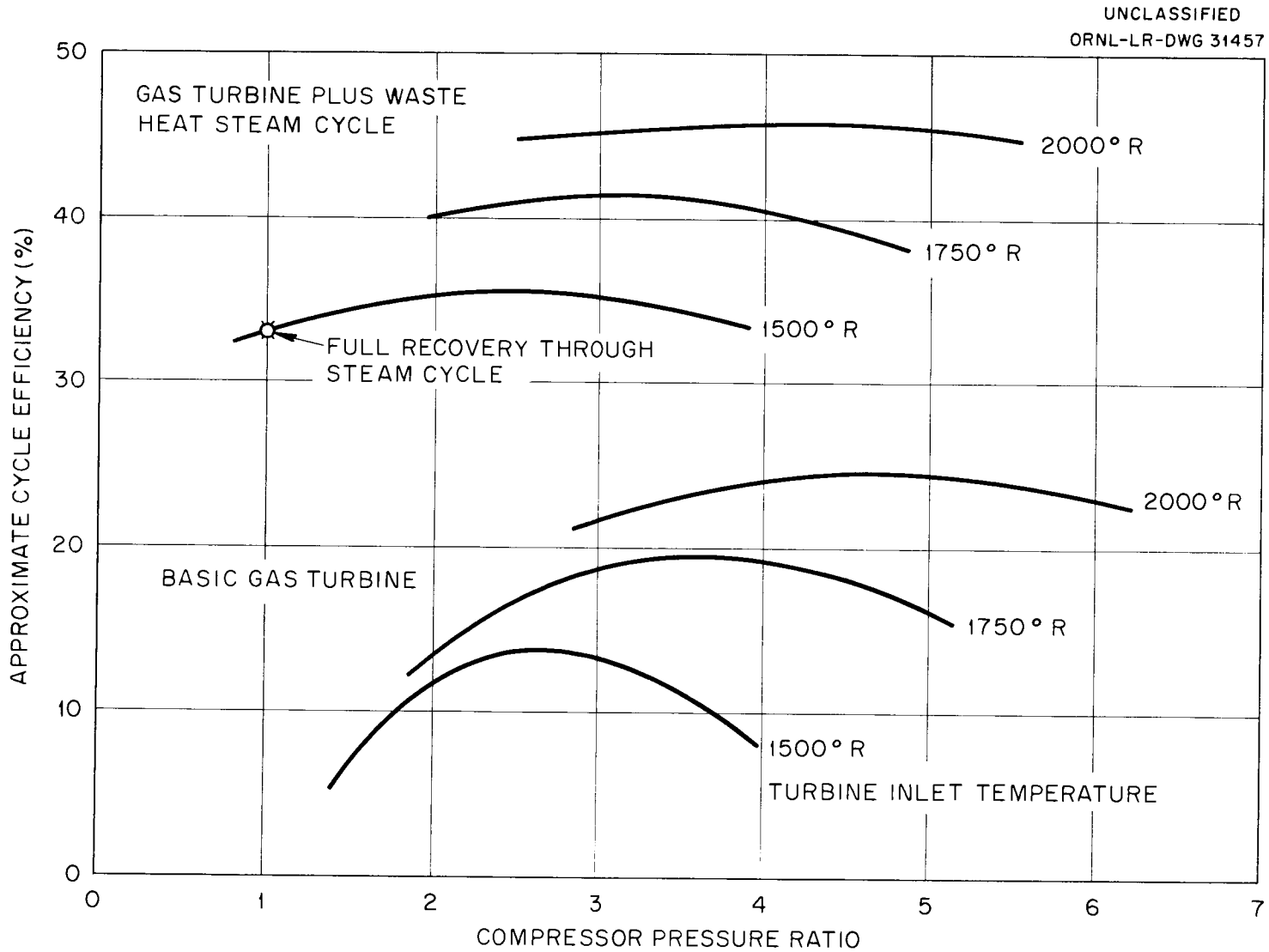


Fig. 14.5. Gas Turbine cycle Efficiencies for Helium Coolant.

load, and therefore the size of the steam generators or other heat exchangers, and second, by affecting a reduction of pressure between the reactor and heat exchangers equal to the pressure ratio of the turbine. This would permit a small, high-pressure reactor to be coupled to large, low-pressure heat exchangers.

An interesting adaptation of the gas-turbine cycle is the use of a gas turbine to drive the coolant-gas compressors. As mentioned previously, in certain gas-turbine-cycle arrangements the major part of the power output may come from the waste-heat-recovery cycle. If carried to the extreme, this would result in a system in which the gas turbine produced only enough shaft power to turn the coolant-circulating compressor with all the net output power coming from "waste" heat recovery. Such systems have the advantage of permitting the reactor to run at higher pressure than the steam generators, although not with as great a pressure difference as would be possible with a power turbine; also, the main power-generating equipment would be that of a standard steam plant. The gas turbine and compressor would be smaller and simpler than in the previous case.

Flow of the coolant could be controlled by variations in the speed of the turbo-machinery. There are some disadvantages associated with this approach, however, if two or more units are to be operated in parallel, as would undoubtedly be the case. The use of turbo-compressors in parallel has attendant problems in matching and cross-coupling instability.[34,35] The problems are complicated by variable speed, especially so if the driver is aerodynamically coupled to the compressor. It should be noted, however, that these remarks do not apply if the turbo-machinery is coupled to alternating-current generators. The speeds of the several parallel units will then be fixed and synchronized through the electrical circuit, except during start-up and shutdown operations.

14.3.4 Power-Recovery Component Problems: The relatively simple coolant-system arrangement of the ORNL GCR-2 was directed toward a design which could utilize current technology for the immediate construction of a nuclear reactor. It is not necessarily applicable to the more advanced concepts. The "dirty"

---

[34] W. T. Furgerson, Notes on Use of Turbo Blowers, ORNL-CF-57-12-24, (April 29, 1958).

[35] W. T. Furgerson, Blower-System Dynamics, ORNL-CF-58-4-80 (May 2, 1958).

versus "clean" reactor design implications are discussed in Section 14.2.2. Aside from the contamination considerations, the component problems of each concept are affected by: (1) coolant composition, (2) coolant temperature, and (3) coolant pressure.

The design and development of major power-recovery components to match the coolant conditions are important portions of a well-rounded gas-cooled reactor program. Equipment fabricators must be encouraged to keep step with advancements in the technology so that the real potentials can be realized without requiring major extrapolation of existing technology (e.g., if the goal of the gas-cooled reactor program is to handle 1500°F gas, equipment suppliers must begin now to develop components which can be utilized in such a system). Without such a program, there will be limited value in the design and evaluation of such systems since, at best, the cost evaluation will be based on inexperienced guesses.

The important items of equipment which should be considered are pressure vessels, piping, valves, heat exchangers, blowers or compressors, and fuel-handling systems.

Vessels. The dependence of the vessel design on coolant composition is mainly one of material compatibility. At some temperature, all of the possible coolants except helium attack the more common vessel materials such as carbon steel, low-alloy chrome-molybdenum steel, high-nickel-alloy steels, and chrome-nickel alloys. One of the foremost problems in reactor vessel design is the determination of temperature limits for each combination of materials and coolants with regard to material compatibility. Information is so limited at this time that it is not possible to determine what limits are imposed by various coolants on the vessel materials.

When compatibility problems do not exist, temperature-pressure considerations become the vessel-design control factors. Each material has a temperature limit beyond which its allowable stress falls off rapidly. These are roughly the following: low-carbon steel to 650°F; low-alloy steel to 1000°F; high-nickel-chrome alloy steels to 1500°F; nickel-chrome alloys to 1500°F; and, ceramics above 1500°F. These temperatures, however, can be only a rough guide in material selection. Many materials have overlapping temperature ranges so that in some cases a design may fit more than one metal. Furthermore, a small adjustment in temperature may shift a vessel design from

one material to another. Since some materials are more easily fabricable than others, the selection will have a major effect on the cost of the vessel. Thus, an SA-301, Grade B, chrome-molybdenum steel vessel may cost one-half that of an SA-347 stainless steel vessel when designed for 950°F, but if the temperature is elevated to 1000°F, the allowable strength of the chrome-molybdenum alloy drops to such an extent that the use of SA-347 stainless steel material may result in a cheaper vessel than would be the case with type SA-301, Grade B, steel.

The characteristics of high-temperature alloys which make them suitable for high-temperature service complicates their fabrication requirements. Hot-forming of high-temperature alloys usually is carried out at temperatures beyond the limit of all except some specialized fabricating facilities. When evaluating vessel design, one generally finds metals for all of the temperature ranges up to 1500°F which can be cold-formed to about the same thickness limitation. Once the cold-forming thickness limit has been passed, however, and hot-forming is required, the high-temperature alloys are limited in thickness and dimension by the available fabrication facilities suitable for high-temperature forming. An urgent need in the entire reactor-vessel design field is to establish what the dimensional limits are for all classes of materials under consideration for nuclear reactors and to determine a cost index for various vessel materials as affected by the dimensional limits. The analysis should, perhaps, be extended one step further to determine whether a deliberate attempt to supplement fabrication facilities for high-temperature alloys would alter nuclear-reactor-design criteria significantly.

Heavy dependence on the ASME Unfired Pressure Vessel Code as a basis for the design of nuclear-reactor vessels is another area requiring critical evaluation. The use of the conservative factors of safety of the ASME Code sometimes results in fabrication monstrosities which defeat the purpose of the Code to establish a safe and conservative design basis for vessel design. To what extent could nuclear-reactor designs benefit by modification of vessel-design philosophy? In some cases, it would permit use of materials with known and reliable fabrication properties in preference to exotic high-strength materials of which there is limited fabrication knowledge. Perhaps more reliable and, at the same time, more economic designs would evolve.

The design of reactors for temperatures above 1500°F will lead into the field of ceramics and ceramic-lined vessels. These will be more nearly akin to furnaces, but with high-integrity requirements for containing gas under pressure. Ceramic-lined pressure vessels are not new to industry, but methods of fabrication and assembly are somewhat limited. If the fuel elements have substantial weight, there will be a problem in carrying the load through the ceramic liner while still insulating the metal vessel wall from the high temperature.

Even more complex is the matter of vessel penetrations (see also Section 14.4.5). If advantage is to be taken of the high temperature, the hot gas must be conducted to the energy extraction point. The design of ceramic liners at vessel penetration points will be extremely complex. In particular, one should not ignore the vessel assembly problem. It will be desirable that the vessels be small to simplify fabrication of pieces but a small vessel will have limited access. A tremendous amount of ingenuity will be required to make the vessels accessible for erection. Thus, reactor systems, which on paper appear to have excellent performance characteristics, may be limited in realizing their potential by fabrication problems.

The D_2O-moderated gas-cooled reactors introduce a reactor vessel problem which is peculiar only to that concept. In this case, hot gas must pass through the region at which D_2O is being maintained at a low temperature. An isolation system is required to keep the D_2O insulated from the gas. The vessel may be designed for low pressures and temperatures and the containment accomplished within the pressure tubes. These, in order to reduce the system poison, must be of a low-cross section material such as aluminum, zirconium, or beryllium. The first two materials must be cooled by the D_2O; the last could be maintained at the gas temperature. The principal vessel problem will be in the design of the pressure tube and headers. Proper treatment of thermal stresses in the arrangement will be important. The pressure-tube header would probably be of some other material than the pressure tubes in order to withstand the high temperature and pressure of the hot gas. The problem of joining the tubes to the header will involve some investigation of the metallurgical problems associated with joining these two dissimilar metals or development of a mechanical seal. A design study of the reactor arrangement will be necessary to evaluate the fabrication problems of this design.

Piping and Valves. In general, the problems of piping fabrication are comparable to those of vessel fabrication. Thickness requirements are basically lower and cold-forming is practical. This generalization, however, is not necessarily true since cold-forming and even hot-forming limitations are dependent upon minimum-bend radii. The minimum increases with thickness so that in some cases, where thickness-to-diameter ratios are high, forming may not be practical. Expensive machined forgings may be required in these cases, which would affect the capital cost significantly.

Thermal expansion is another complex problem. Piping must be designed with sufficient flexibility to relieve thermal-expansion stresses. In relatively low-pressure systems, expansion joints may be effective. As pressures increase, the inherent flexibility of the expansion joint must be sacrificed because of the heavier wall sections required to contain pressures. At some stage in the design, the expansion joint becomes impractical. Some designs then resort to large pipe bends, an extremely costly method of obtaining flexibility. As pipe diameters increase, even pipe bends become impractical and the stresses are relieved by allowing the connected equipment to move.

The complexity of the problem is increased by the existence of differential temperatures in the piping systems so that one pipe expands at a different rate from another. If, as might be the case in a very high-temperature system, more than one type of material is included in the system, such as a ceramic and a metal, these two materials will probably expand at different rates and thus apply a rotational motion to the attached equipment which must be considered in the design. The interaction exists even when two regions of the same material operate at different temperatures. Concentric pipe systems in which the inner pipe is relatively free to move with respect to the outer pipe are sometimes utilized to circumvent the problem. No attempt is made here to measure relative complexity of various reactor piping systems, but it can be seen that very misleading conclusions as to the practical value of a design can result from failure to include piping considerations in design studies.

Valves for gas-cooled reactors are of two types. One type is required for shutoff purposes and another for flow control. The two types of valves have different design considerations.

Absolute shutoff of gas valves is limited to very low-temperature applications where soft valve seats can be used. Where metal-seated valves

are required some leakage must be tolerated. This in turn makes such valves of limited use as isolation barriers in highly contaminated gas systems. Where some leakage can be tolerated, as in a CO_2-cooled Calder Hall-type reactor, a valve of this type might permit limited access to the system external to the valve. In other systems, the requirements of shutoff valves require critical examination. In many cases, their functional criteria are impractical so that the valves should be eliminated or, at least, the feasibility of the reactor system should not be dependent upon their successful operation.

Flow-control valves are an effective means of maintaining reactor load-following characteristics. In most cases, simple butterfly or vane-type valves are adequate. The major problem is an adequate drive linkage to operate the butterfly or vane.

In all cases the problem of stem leakage must be reconciled. In systems of low contamination, valves with packed valve stems may prove adequate. In contaminated systems, packless valves with bellows-seals, or packed valves with buffered valve packing will probably be necessary. Packless valves become increasingly more difficult to seal as design pressure increases because of the difficulty in reconciling pressure stresses with flexibility in bellows-seals. In very high-temperature systems, valve packing, if used, may require insulation and/or cooling in order to utilize conventional packing materials.

Fabrication of valve bodies is also a difficult problem in high-temperature gas systems. Stresses imposed by the seat thrust must be added to normal piping stresses. Designs in which the valve disk is lifted out of the gas stream require additional volume for storage of the disk and usually result in asymmetrical body structures with attendant stress-design problems. Rotary valves are frequently preferred to avoid this complication.

Fabrication techniques for valves may also be influenced by gas-tightness requirements of the body structure. In many cases castings are not considered reliable and welded or forged structures are required. Under these circumstances, valve fabrication can represent a substantial portion of reactor system costs.

Valve design problems are not normally an important factor in reactor systems analysis, but in some instances the success or failure of a design may hinge on availability of suitable valves. Design work on gas-cooled reactors

could be furthered substantially by a thorough survey of the available valve types and their design limitations. From this, the next natural step might be toward the development of improved designs for advanced reactor concepts.

Heat Exchangers. With the exception of the direct-cycle gas-turbine power-recovery systems (see Section 14.3.3), all of the various designs are dependent upon transfer of heat from the reactor coolant to a heat-extraction fluid, usually water or steam. There is a very close interconnection between heat exchanger performance and capital costs which requires thorough examination in each gas-cooled reactor design. How much of the available energy should be extracted from the coolant? What is the optimum temperature level for the heat-extraction fluid? To what extent should the turbulence of the reactor coolant be increased through the heat exchanger to improve heat-transfer performance? These are some of the questions to be answered. In gas-cooled reactor systems, the problem is more significant than in any other type because of the major capital investment which the heat exchanger equipment represents. In some cases, the heat exchanger equipment may cost as much as the remainder of the reactor system components. (A recent preliminary examination of one high-pressure gas-cooled reactor concept indicated that in a system requiring $16,000,000 worth of components, the steam generator represented $7,400,000 of the cost.)

In heat exchanger fabrication, the problems of the pressure vessel are compounded by the need for internal heat-transfer surface. Generally the following conditions hold:

1. At some point in the heat exchanger the hot gas must be fed to the heat exchanger surface. A high-temperature vessel penetration problem naturally is incurred.
2. The heat-extraction fluid is not compatible with the reactor system so that no leakage can be tolerated in the system for extended periods. The coolant system must therefore be leak-tight with respect to the steam system.
3. Reliability of the heat exchanger tubing can be best assured by bringing all welded connections outside the heat exchanger vessel, but the vessel fabrication is greatly complicated by this arrangement.

The evaluation of each new reactor concept involves consideration of these conditions for the heat exchangers: Size, temperature differentials, power consumption, pressures, and tube arrangements all must be factored into the design. For quick evaluation of a number of systems, it is desirable to determine the effects of these parameters on capital cost and reactor performance. Unfortunately, the design arrangement greatly affects the evaluation. There is a need to prepare a number of layout studies of heat exchanger arrangements, to carry this work far enough to show that the arrangements are fabricable, and to analyze the elements of cost in each design so that advanced reactor concepts can be measured in terms of their effects on these basic heat exchanger designs.

Fuel Handling. The methods of feeding and removing fuel from the reactor system are tied first to the form of fuel in the reactor and second to the type of coolant and its operating conditions.

When each fuel element is a separate entity, as in the case of the Kaiser-ACF design, the Calder Hall-type reactor, or the ORNL GCR-2 design, the fuel-handling problem is merely one of mechanical manipulation. There are a number of such arrangements which involve top and bottom loading and there are others, such as the French Marcoule Reactor, which utilize side loading. In general, systems of this sort must be measured in terms of mechanical reliability. For CO_2 systems of low contamination, continuous fueling under pressure can be considered without fear of costly or hazardous leaks. By comparison, highly contaminated systems or systems which must maintain low-coolant losses require a high degree of leakage integrity. Continuous loading in the latter case would involve considerable speculation on the reliability of loading connections. Some test work is needed to eliminate the speculative character of these designs.

Pellet and fluidized fuels, which are essentially uncontained and therefore involve a high level of activity and significant quantities of fission products, do not lend themselves to conventional mechanical-handling techniques. Gravity charge and discharge techniques or fluidized transport systems are implied in the design of these fuels. In these cases, the external fuel-handling systems must have the same containment integrity as the reactor system. Hermetically sealed systems will be required.

Homogenized cores and small core assemblies (see Section 14.1) involve installation and removal of major core assemblies during each loading operation. These, too, may be mechanical-handling systems comparable to the GCR-2 techniques; but because of the size of the pieces, the designs will probably require either removal of the reactor head assembly or large access openings. In either case, the large size of the core sections to be removed will require that the reactor be shut down for loading. Access openings might necessarily be bolted because of the need for frequent access. If hermetic conditions are required, flanges may be aided by supplemental light seal welds.

An evaluation of each reactor system will require a study of one of these methods of fuel handling. Reference design studies of the various methods are necessary to give an insight into the influence of each fuel-handling system on the over-all reactor design.

Blowers and Drives. The mechanical problems of coolant blowers and drives are sufficiently complex so that an attempt must be made wherever possible to segregate these components from the more severe design conditions of the reactor. Such an approach can be only partially successful. The following are significant design conditions which might affect the design of the blowers and drives: (1) increase in pressure differential in the reactor system in order to obtain higher throughput of coolant in a given reactor configuration, (2) variation in the composition of the coolant, (3) increase in temperature level of the coolant, (4) increase in coolant-system pressure, and (5) isolation of the blower drive from the coolant system.

Increase of pressure difference across blowers will give a proportionate increase in axial thrust load. This must be accommodated by a larger thrust bearing or by a revision of the "balance piston".

Increased coolant density means increased windage losses for "canned" motors. Helium density in GCR-2 was approximately the same as atmospheric air and, consequently, gave 2 to 3% windage losses. If carbon dioxide were used (all other things being equal), losses would be approximately ten times this figure and would be obviously impractical. It is probably safe to say that for helium or hydrogen coolants one could use canned motors at pressures up to 500 psia, but external motors with shaft seals should be used for higher densities.

Increased temperature at the blower, resulting either from increased system temperature or from placing the blower in a hotter part of the system, aggravates several problems. The cooling problem may make a canned motor impractical even where windage losses are acceptable. Heat conducted along the shaft must be removed, probably through the bearings, or, in the case of a wet sleeve seal, through the sealant liquid. This means the vapor pressure of the sealant will be increased with consequent increase in possibility of back diffusion of sealant vapors into the system. None of the temperatures contemplated, even at the reactor discharge, would result in undue difficulty in design of the blower impeller. Both stresses and temperatures are well within the range of gas-turbine or even steam-turbine technology. The difficult problems will be connected with seal and bearing temperatures and differential thermal expansion between different parts of the blower. The situation is comparable to that in a steam turbine where arrangements for pressure and temperature reduction along the shaft to protect bearings and seals are long and complicated.

If wet shaft seals are used, the solubility of coolant in the sealing liquid will affect the seal system. If the coolant is valuable, that which is dissolved in the sealant must be recovered and recycled to the coolant system, and the greater the solubility the larger the recovery equipment must be. If the coolant carries activity, its solubility in the sealing liquid will influence the amount of shielding which must be placed around components of the seal system.

As system pressures are increased, the solubility of coolant in the sealing liquid will increase. Solubility of gases in liquids is decreased as temperature is increased, but it is not expected that increased coolant temperature at the blower will have a full effect on coolant solubility in the sealing liquid. The blower bearings will have to be kept within certain temperature limits, and since the seal is located close to the bearings, it is expected that the bulk of the sealant will be at essentially constant temperature regardless of the coolant temperature.

14.4 Instrumentation and Control

This section gives a brief outline of some of the more important instrumentation and control problems anticipated in the development of operational reactors utilizing the gas-cooling concept, along with discussions of suitable methods and techniques which might be applied in achieving solutions.

14.4.1 Analog Simulator Studies: A major objective of analog simulation is the determination of interactions between the various component systems of the complete power plant. In the present case, these components are the reactor, the coolant-gas circuit, the steam generators, the turbines, and the feed-water supply. In order to achieve a complete understanding of the dynamic characteristics of these systems, it is necessary to determine first the detailed characteristics of the individual constituents of each system or circuit. Of particular interest in this connection, there is, for example, the investigation of the behavior of the once-through steam generators proposed in the GCR-2 design.

The analog simulator will be especially helpful in determining the effects of failures of various power-plant components, of the corrective actions required to maintain temperature excursions within design limits, and the general problem of controlling the plant under normal operating conditions.

Component failures of interest should include the effect of loss in gas flow, stoppage of steam flow to the turbines, and the loss of feed water.

The study of corrective actions should include operating conditions involving component failure, as well as steady-state behavior without failure.

Plant control under normal conditions should include studies of the effect of change in gas flow rate, of reactor reactivity, and of feed-water flow rate required for adequate control, together with the necessary sensing and stabilizing signals.

Finally, it should be pointed out that detailed analyses of this nature can be carried out only if complete engineering and nuclear information is available. Thus, detailed engineering studies of the type prepared for the GCR-2 are essential if meaningful simulator studies are to be obtained.

14.4.2 Control-Rod Mechanism: The operational requirements of a large gas-cooled power reactor, such as the ORNL GCR-2, may place several unique specifications on the control-rod design. These are:

1. Simplicity and ruggedness of design to achieve a high order of reliability for extended periods of operation.
2. Individual rod drives in order to minimize the reactivity made unavailable due to mechanism failure.
3. Simple but accurate position indicators for each rod coupled with graphic annunciators to display the configuration of complete control-rod array and deviations from desired pattern.

4. Simple actuator mechanisms for operation within the hot, high-pressure gaseous environment of reactor.
5. Development of gas and electrical seals for reactor-pressure-vessel penetrations.

14.4.3 Neutron Detection: The need for accurate measurement of the neutron flux is apparent in the case of the GCR-2. Because of the large size of the reactor, the thermal flux at the chamber locations at a power level of 700 Mw, is estimated to be approximately 10^9 n/cm^2·sec, which is between one and two decades lower than that desired for presently available chambers. This low chamber flux at full power requires that the useful lower limit of the reactor power be a relatively high value, in order that the nuclear power be indicated over the full range of operation. The shutdown neutron-flux level will fall to values too low for adequate detection using the presently available chambers unless a very large neutron source is provided. Several possible solutions to this problem should be examined: (1) more sensitive neutron chambers, (2) the use of beryllium as a photo-neutron source, (3) neutron chambers that can be placed in the core, with the attendant problems of temperature and pressure, and (4) modifications of the reactor-pressure-vessel design that will permit the present chambers to be placed in a higher neutron flux. It will be necessary to obtain data concerning the spatial distribution of the neutron and gamma flux as well as data describing the decay of gamma flux following shutdown of the reactor.

A second reactor control problem of primary importance is the development of suitable techniques for obtaining neutron flux plots. The possibility of a rod position modulating scheme should be examined for determining the radial distribution of the reactor flux. This might be accomplished by observing the change in neutron flux at the chamber locations when the position of a control rod is varied in a certain prescribed manner. This particular problem requires considerable analysis, and the solution, if feasible by the method proposed, is closely associated with the design of the rod drive mechanisms and the neutron sensing chambers.

14.4.4 Fuel Element Operating Conditions: A close surveillance of fuel element operating temperatures throughout the reactor provides an indication of possible fuel element burnout. It is obviously impractical to monitor the

temperature of every fuel element for hot spots. Also, it is meaningless to measure the temperature of only a few elements. However, in the case of a prototype gas-cooled reactor, it may be desirable to provide temperature measurements to locate the hottest elements and to evaluate the heat-transfer performance of the elements throughout a given time of operation. Various techniques should be investigated for use in such sensing devices and instrumentation.

In the event that a leakage of fission products from a fuel element occurs, it would be desirable to know when and where an element has failed because the consequent deposition of fission products in the components of the coolant system would be hazardous to maintenance and operating personnel. The instrumentation to provide such information represents the greater part of the over-all cost for plant instrumentation and controls. This high cost, together with the extreme complexity of the system, has stimulated the desire to investigate the possibility of simplification within the limits of safety requirements. An evaluation of these considerations requires answers to the following questions: (1) How serious is the failure of a fuel element? (2) What operational action will be taken in case of a failure? (3) What failure rate can be expected? The answers to these questions will determine whether or not continuous monitoring of each fuel channel is essential and whether the determination of the activity of the coolant in the main ducts is adequate as an indication of a fuel element failure. If continuous monitoring of each fuel channel is unnecessary, identification of a faulty channel could possibly be accomplished with local instrumentation together with the use of the fuel-charge machine as a scanner under load conditions. Such a scheme would completely eliminate tubing penetrations and considerable equipment.

14.4.5 Pressure-Vessel Penetrations: Penetrations of the pressure cell are required for gas-sampling tubings, thermocouple leads, and control wires for the rod drives. Tubings and thermocouple leads enter and leave the pressure system through the vessel wall proper; whereas, the control wires for the rod drives penetrate the control-rod-drive housings.

All penetrations must be designed to satisfy ambient operational conditions, such as pressure, temperature, material compatibility, and above all, must be completely leak-proof and simple in construction. Also, the design should be such as to permit increase in ambient operating pressure without modifications.

SECTION 15

MATERIALS RESEARCH AND DEVELOPMENT

15. MATERIALS RESEARCH AND DEVELOPMENT

Contents

15.1 Introduction 15.1

15.2 Investigation of Alternate Coolants 15.2

15.2.1 The Relationship Between the Heat-Transfer Rate and the Pumping Power as an Index of Coolant Suitability . . . 15.2

15.2.2 Induced Radioactivity in Coolants 15.6

15.2.3 Materials Compatibility Problems 15.7

15.3 Advanced Fuels 15.14

15.3.1 Fuels with Lower Fabrication Cost 15.16

15.3.2 Fuels with Improved Nuclear Properties 15.16

15.3.3 High-Temperature Fuels 15.17

15.4 Alternate Moderator Materials 15.17

15.4.1 Graphite 15.19

15.4.2 Coated or Canned Graphite 15.19

15.4.3 Heavy Water 15.20

15.4.4 Beryllium Oxide 15.20

15.5 Capsule Materials 15.21

15.5.1 Beryllium 15.21

15.5.2 Aluminum and Magnesium 15.23

15.5.3 Zirconium 15.24

15.5.4 Niobium and Molybdenum 15.25

15.5.5 Stainless Steel 15.26

15.5.6 Graphite and Coated Graphite 15.26

15.5.7 Canned Graphite 15.27

15. MATERIALS RESEARCH AND DEVELOPMENT

15.1 Introduction

The materials research and development program for the GCR-2 was discussed in Section 12 of ORNL-2500. In this report some speculations will be made on advanced concepts of the gas-cooled-reactor materials, the use of which would result in cost advantages over the GCR-2, provided further research and development should show that the compatibility requirements are met. Costs would be reduced if new materials were used in the reactor which (1) would operate at a higher temperature, (2) would absorb fewer neutrons at the same temperature, and (3) are cheaper but perform equally well. Costs would also be reduced if a coolant such as hydrogen could be used which has better heat-transfer characteristics than helium. The primary limitation on many of the proposed materials is that very little information is available about their behavior in a reactor environment. Some of the known limitations will be discussed and areas most needful of information will be outlined.

For the optimum utilization of new materials, entirely new classes of gas-cooled reactors must be considered. For instance, materials which could operate at much higher temperatures than stainless steel, such as ceramics or graphite, may be feasible only if a portion of the fission products can be tolerated in the gas stream or if vented capsules are used. The increased efficiency associated with the higher temperatures might far outweigh the increased maintenance costs of a contaminated system. Other factors being equal, however, contamination-free systems are much to be preferred. Thus both contaminated and contamination-free gas systems will be considered for future reactors. Operation at much higher temperatures might make it more advantageous to use gas turbines instead of steam to drive the generators. For this reason, both gas-turbine and steam-generating types of systems will be treated. The reactor types may be further divided, with respect to the fuel element, into heterogeneous types using bulk oxide fuel as in the GCR-2 and homogeneous types using fuel-moderator mixtures

such as $BeO-UO_2$ or UC-C. Those systems which use graphite as a moderator may be subdivided as to coated, uncoated, or canned graphite types. To assist future design studies, the more likely possibilities for the coolant, moderator, fuel, and capsule materials will be discussed with their attendant advantages and disadvantages.

15.2 Investigation of Alternate Coolants

Since the United States is the only nation with a known adequate supply of helium, other countries which represent markets for gas-cooled reactors would prefer other gases as coolants. Other gases might be used if changes were made on the radiation levels in the coolant which could be tolerated if the graphite were canned or coated, or if new fuels or moderators were developed. Any one of these modifications would call for a re-evaluation of the coolant gas. In addition, a different coolant might be desirable for a gas-turbine reactor. Thus, further consideration must be given to alternate coolants to keep pace with the development of other components of the reactor.

In considering different coolants the following factors must be taken into account: (1) the relationship between the heat-transfer rate and the pumping power, (2) radiation levels, and (3) materials compatibility. The order of listing of these factors is not directly related to the order of importance, since an unfavorable materials compatibility is the only factor that can completely negate a given system from consideration. The other factors are related more to the question of costs. All of these factors will be discussed in the following sections in order to point out the relative merits of possible coolants.

15.2.1 The Relationship Between the Heat-Transfer Rate and the Pumping Power as an Index of Coolant Suitability: One of the criteria of suitability of a given gas as a nuclear reactor coolant is its ability to provide high rates of heat transfer from the fuel to the coolant stream and high rates of transport of heat from the reactor to the boiler. A distinction must be made between these two processes since they are functions of different properties of the gas. The former is dependent on the conductivity and the Reynolds and Prandtl numbers; the latter is a function of the molar heat capacity. Carbon dioxide, for instance, has excellent heat-transport properties, but is

relatively poor from a standpoint of heat transfer, while helium is just the reverse. The pumping power required to remove a given quantity of heat from a reactor operating at a given set of pressure and temperature conditions is an index of its quality as a coolant.

Previous studies[1,2] have indicated that for a given rate of heat removal the ratio of the pumping power required to circulate one coolant to that required for a different coolant, $\frac{W_2}{W_1}$, is given by the expression

$$\frac{W_2}{W_1} = \left(\frac{C_{p_1}}{C_{p_2}}\right)^3 \left(\frac{m_1}{m_2}\right)^2 \left(\frac{Pr_2}{Pr_1}\right)^{2/3}$$

where C_p, m, and Pr refer to the specific heat, density, and Prandtl number of a given coolant. However, this expression is based on the assumption that the total rate of heat transfer, q, the reactor heat-transfer area, A, and the effective temperature driving force for heat transfer, ΔT, are the same for both coolants. Since these quantities are related to the coefficient of heat transfer, h, by the heat-transfer rate equation,

$$q = h\ A\ \Delta\ T$$

it can be seen that h must be the same for both coolants in order to maintain equality of the other factors. The heat-transfer coefficient depends on coolant-channel dimensions, physical properties, and flow rate. Since the flow rate is already fixed by the energy balance of the system, it will be necessary to vary the channel dimensions in order to maintain h constant for coolants of different physical properties. In the previous studies, the channel dimensions had been erroneously assumed to remain constant.

The ratio of channel diameters necessary to maintain constant h for coolants of different properties can be found by setting

---

[1] Preliminary Design Study of a Closed-Cycle Gas-Turbine Nuclear Power Plant, (New York: Sanderson and Porter Engineers and Constructors, 1955) Section 1.2.

[2] J. Diamond, "Heat Removal from Nuclear Power Reactors," J. British Nuclear Energy Conference, 1, No. 3, 227 (October 1956).

$$\frac{h_2}{h_1} = \frac{\frac{k_2}{D_2}\ (Re_2)^{0.8}(Pr_2)^{0.4}}{\frac{k_1}{D_1}\ (Re_1)^{0.8}(Pr_1)^{0.4}} = 1$$

where k is the fluid thermal conductivity, D is the equivalent channel diameter and Re is the Reynolds number. Solution for $\frac{D_2}{D_1}$ yields

$$\frac{D_2}{D_1} = \left(\frac{k_2}{k_1}\right)^{0.6} \left(\frac{C_{p_1}}{C_{p_2}}\right)^{0.4} \left(\frac{\mu_1}{\mu_2}\right)^{0.4}$$

where μ is the fluid viscosity. In arriving at this relationship it is assumed that the channel geometry is such that its flow-cross-sectioned area is proportional to the square of the equivalent diameter.

The pumping power required to circulate a coolant at mass velocity, G, through a channel of length, L, and equivalent diameter, D, is given by the Fanning equation

$$W = \frac{2\ f\ D\ G^3\ L}{g_c m^2}$$

where f is the friction factor and g_c is a unit conversion constant. From the heat-balance relation, with heat-transfer rate and coolant-temperature rise held constant, it can be shown that

$$G \propto \frac{1}{D^2 C_p}$$

Also, if the flow through the coolant channel is assumed turbulent,

$$f \propto (Re)^{-0.2}$$

and hence

$$f \propto (\mu C_p D)^{0.2}$$

Combination of the above relationships yields the desired pumping-power ratio

$$\frac{W_2}{W_1} = \left(\frac{C_{p_1}}{C_{p_2}}\right)^{1.73} \left(\frac{k_1}{k_2}\right)^{1.6} \left(\frac{m_1}{m_2}\right)^{2} \left(\frac{\mu_2}{\mu_1}\right)^{1.27}$$

One may eliminate the viscosity in terms of the Prandtl number, $\frac{C_p \mu}{k}$, and obtain

$$\frac{W_2}{W_1} = \left(\frac{C_{p_1}}{C_{p_2}}\right)^3 \left(\frac{m_1}{m_2}\right)^2 \left(\frac{k_1}{k_2}\right)^{0.33} \left(\frac{Npr_2}{Npr_1}\right)^{1.27}$$

It is then seen that the correct pumping-power ratio differs from that previously given by a simple multiplication factor involving the fluid thermal conductivity and Prandtl number.

TABLE 15.1

PUMPING POWER AND EFFECTIVE CHANNEL DIAMETERS FOR VARIOUS COOLANTS

	Helium	Hydrogen	Carbon Dioxide	Air Nitrogen Carbon monoxide	Steam	Ethane	Ethylene	Methane
D/D_h	1.0	1.02	0.82	0.83	0.84	0.92	0.91	0.94
W/W_h	1.0	0.118	1.02	3.3	2.0	0.12	0.17	0.23

The above equations are based on the assumption of a fuel element configuration in which the cross sectional area for flow is directly proportional to the square of the equivalent diameter. Configurations such as the GCR-2 "septofoil", require a trial and error solution, but the results are similar.

The diameter ratio and pumping-power ratio calculated for several gases relative to helium at 1000°F are given in Table 15.1.

The diameter ratio is important if the use of two different gases is contemplated for the same system. Note that the above pumping-power ratios represent only the power required to overcome pressure losses incurred in the heat-transfer process. When parasitic losses are added the spread will be reduced. Also, the equations given are derived for a certain channel configuration. A separate relationship must be derived for each channel type, and this will affect the relationships to some extent. The data show that on the basis of heat-transfer efficiency alone hydrogen is probably the best coolant, followed in order by the hydrocarbon gases and helium, carbon dioxide, and water vapor, with the common diatomic gases - air, nitrogen, and carbon monoxide - being definitely inferior.

15.2.2 Induced Radioactivity in Coolants: A high radioactivity in the coolant gas is undesirable, because of the cost of the increased weight of gamma shielding and because of delays associated with maintenance when long-lived isotopes are present. Radioactivity in the coolant gas may arise from several sources: (1) neutron reactions with the gas, (2) neutron reactions with impurities in the gas, (3) neutron reactions with particulate matter in the gas such as that resulting from erosion, and (4) gaseous fission products released from the fuel. For the GCR-2 Reference Design, the activity in the gas stream from these sources is expected to be quite low; but, since a safe operation might be achieved at a lower total cost through the use of unprotected fuel elements or through the use of coolants with a high induced radioactivity, future concepts do not exclude these systems.

The primary source of induced activity in helium arises from the production of tritium, which emits only beta radiation. Since the tritium may be easily removed, it does not represent a major hazard. The hydrocarbons and CO_2 are also quite satisfactory from the standpoint of induced radioactivity because the only long-lived isotope produced in these gases is 5570-year C^{14}, which is a weak beta emitter. This isotope arises from C^{13} by a (n,γ) reaction and is also produced from N^{14} by a (n,p) reaction. Nitrogen-16, which with its daughters emits 6.3, 7.2, and 2.7 mev gamma, is also formed by a (n,γ) reaction from N^{15} or a (n,p) reaction from O^{16}, but its half-life is only seven seconds. Thus, if no impurities were present, all of the gases in Table 15.1 would show low radiation levels in the cooling circuit.

When impurities are considered, the induced activity problem becomes more serious. The most undesirable impurity is argon because A^{41}, which has a 1.8-hr half-life and emits 1.7 mev gamma, is produced from A^{40} by a (n,γ) reaction. Commercial high-purity nitrogen containing about 50 ppm argon would have a low activity. However, air, which contains 1% argon, would have an equilibrium gamma activity sufficiently high to make routine servicing difficult in systems using air as a coolant. The relative total and gamma activities of the more promising coolants were given in detail in Table 6.1 of Section 6.

The effects of induced radioactivity in the coolant gas would be less important in systems where a portion of the fission products are released to the gas stream, i.e., those systems without an impervious fuel capsule. Volatile fission products which would be released include cesium, rubidium, krypton, xenon, bromine, and iodine. A study by a 1954 ORSORT group[3] on a gas-cooled ceramic power breeder showed that the resulting activity was one curie for each ppm of fission gas released. In a given system the radiation levels must be calculated because they depend on the amount released and the way that the fission products are distributed.

In view of the above considerations, air must be excluded as a coolant from those systems in which low activity levels are desired; otherwise, with respect to induced radioactivity, any of the coolants may be considered.

15.2.3 Materials Compatibility Problems:

A. Hydrocarbons - The data given in Table 15.1 indicate that the lighter saturated hydrocarbons possess desirable heat-transfer characteristics and, therefore, might be used advantageously as coolants. In addition to being highly inflammable, these gases may possess certain other undesirable properties. The effect of intense radiation is not well known, although it appears likely that these gases would be decomposed to a significant degree if subject to the radiation flux prevailing in a high-power reactor. These gases also may act as carburizing agents, thereby eliminating the use of metals or alloys adversely affected by carburization. If used in conjunction with graphite, carbon mass transfer also could be a serious problem. Considerable development work would be required to evaluate the stability and carburizing power of these gases under radiation before they could be considered as coolants.

B. Carbon Dioxide - Carbon dioxide has found practical application as the coolant in the Calder Hall Reactor. As a heat-transfer medium it is considerably better than air or nitrogen and comparable to helium.

---

[3] J. F. Smith et al., Gas-Cooled Ceramic Power Breeder Reactor, ORNL CF-54-8-235 (August 1954).

Carbon dioxide is relatively inert at temperatures below 400°C and possesses acceptable nuclear properties. In the absence of ionizing radiation and of graphite, carbon dioxide is a stable gas; for example, K_p for the reaction $CO_2 \leftrightharpoons CO + \frac{1}{2} O_2$ is 1.3×10^{-16} at 450°C. Slight decomposition does occur, however, under irradiation. Davidge and Marsh[4] reported values for the carbon monoxide concentration ranging from 10^{-3} to 6×10^{-4} % decomposition of CO_2 at 325°C when carbon dioxide was exposed in the British Experimental Pile 0 in the absence of graphite. Limited data obtained from the exposure of carbon dioxide to the radiation of a Hanford pile indicate that about 1% of the CO_2 was decomposed at 30°C.[5] The data, while not precise, show rather conclusively that the decomposition is promoted by radiation. This limited decomposition of carbon dioxide probably should not prejudice its use as a coolant.

As previously mentioned, carbon dioxide does not react readily with the reactor components at low temperatures. At temperatures above 500°C carbon dioxide does undergo reactions with graphite and a number of metals. With graphite the reaction $C + CO_2 \leftrightharpoons 2\ CO$ becomes of increasing importance as the temperature rises. Oxidation by carbon dioxide results in significant changes in the mechanical properties of the graphite, and a reversal of the reaction results in the deposition of carbon in the cooler regions. This deposition resulting from the decomposition of carbon monoxide is catalyzed by radiation and active metal surfaces. Carburization of structural metals very likely would occur in the system as the concentration of carbon monoxide increased with increasing temperature.

In order to use carbon dioxide at temperatures above 1000°F, the graphite would have to be canned, coated, or another moderator (BeO) employed. One coating which shows promise for graphite is siliconized silicon carbide. Oxidation of metals by carbon dioxide at elevated temperatures would make the selection of a suitable canning material for the fuel elements more difficult and also complicate the design of the heat exchangers. Assuming that these

---

[4] P. C. Davidge and W. R. Marsh, The Effect of Pile Radiation on the Carbon-Dioxide-Graphite Reaction, AERE C/R 1374 (1955).

[5] R. E. Woodley, The Promotion of Chemical Reaction by Pile Radiation, HW-40142 (November 22, 1955).

materials problems can be resolved satisfactorily, carbon dioxide could serve as a satisfactory coolant at high temperatures using either a steam-generation or gas-turbine system.

C. Hydrogen - The utilization of hydrogen would be highly advantageous from the standpoint of costs, provided that no other materials changes were required for a steam-cycle reactor. However, the use of hydrogen in a gas-turbine system would necessitate extensive changes in design, and the materials problems associated with hydrogen are severe. In addition, the potential danger from an explosion of hydrogen-air mixtures is well known. Such an explosion could occur in this system if the system were not properly purged at the start-up, or if a quantity of hydrogen escaped from the reactor through a leak and were trapped in a confined volume between the shielding and the reactor, or due to a component failure. The elimination of this danger is primarily a design problem, because with proper design the rate of effusion of hydrogen from potential leaks could be reduced to a negligible value, the probability of a failure would be very small, and provisions for venting the shielding could be incorporated.

The use of hydrogen as a reactor coolant may also introduce other materials problems: (1) embrittlement of the pressure vessel steels, (2) weakening of the fuel capsule due to decarburization or preferential oxidation of the alloying constituents by gaseous impurities, and (3) mass transfer of carbon. Initially it was considered that the loss of hydrogen by diffusion through the pressure vessel would also be significant, but calculations based on existing data on the rate of permeation of hydrogen through iron[6] showed that with a 50-ft pressure vessel of 3-in. wall thickness containing hydrogen at 300 psia, only 9 g of hydrogen would be lost per day at 1250°F. At the reference-design pressure-vessel temperature of 450°F, only 0.1 g of hydrogen would diffuse out of the pressure vessel per day.

---

[6] C. J. Smithells, "Mechanical Properties of Metals and Alloys," Metals Reference Book, (New York: Interscience Publishers, 1955) p 543.

Hydrogen Embrittlement of Pressure Vessel Steels. The extent to which the embrittlement of the pressure vessel by hydrogen would occur is not known. That the possibility of embrittlement exists is shown by the fact that a piece of ductile Armco iron, which has been heated to a temperature of 750°C and slowly cooled in hydrogen, will shatter when it is dropped on the floor.[7] It has been postulated that the embrittlement occurring at high temperatures is produced by the action of free protons formed at the surface of the steel by the thermal decomposition of H_2. These protons diffuse into the metal lattice and interact with carbon at the surface of lattice defects to produce methane. The large methane molecule is unable to diffuse through the lattice, so the pressure increases as the amount of methane increases until eventually the fracture stress of the metal is exceeded in the localized area and a crack is initiated. Since there are literally thousands of such defects in the metal lattice, the resulting structure may be quite brittle. This effect is sometimes observed in ordinary steels in a hydrogen atmosphere under certain conditions but is not a problem with austenitic stainless steels. It is possible that this type of embrittlement will not occur in a reactor under any conditions, because it has been shown that gamma irradiation will decompose methane.[8]

A similar type of embrittlement has been observed at low temperatures.[9] Protons such as those produced by reaction of the steel with steam diffuse into the lattice and recombine at flaws to form H_2, which then behaves in a similar fashion to methane. There is the possibility that protons produced by the radiation-induced dissociation of hydrogen will embrittle the pressure-vessel steel by this mechanism.

---

[7] R. M. Brick and A. Phillips, Structure and Properties of Alloys, (New York: McGraw-Hill, 1949) p 51.

[8] ANL Chemistry Division, Summary Report for July, August, and September, 1952, ANL-5000, (January 21, 1953) Section C-II. Declassified with deletions February 12, 1957.

[9] C. A. Zapffe and C. E. Simms, "Hydrogen Embrittlement, Internal Stress, and Defects in Steel," Trans. AIME, 145, p 225 (1941).

In order to resolve the embrittlement questions associated with the fuel capsule and the pressure vessel, an experimental investigation should be made to determine the tendencies for embrittlement of the most promising materials suggested for use in a hydrogen-cooled reactor. Both in-pile and out-of-pile experiments would be run with samples in contact with high-pressure hydrogen and under the temperature conditions envisioned in the reactor. Measurements of the mechanical properties and metallographic analyses of samples so treated would reveal the tendencies for embrittlement.

Weakening of the Fuel Capsule Material. The high-temperature strengths of austenitic stainless steels may be lowered by hydrogen by means of two processes. The first is the decarburization of the stainless steel by hydrogen with the formation of methane. This reaction occurs rapidly above 700°C, but only a moderate reduction of strength would be anticipated from the loss of carbon. The second mechanism is the preferential oxidation of the alloying constituents in the stainless steel which occurs if the hydrogen is slightly contaminated with oxygen in the form of water vapor. Such a contaminated atmosphere of hydrogen may be reducing relative to iron but oxidizing relative to chromium. The preferential oxidation of chromium in this type of atmosphere could lead to a significant loss of strength of the fuel capsule material. This problem would be minimized by reducing the water vapor content in the hydrogen, a relatively easy task.

The above discussion of the possible interactions between hydrogen and the fuel capsule was based on the assumption that the capsule would be stainless steel. However, development work on an impervious graphite fuel element may eliminate stainless steel from consideration. In this event, the problems associated with the interactions between the fuel element and hydrogen will be the same as those associated with the moderator. In the definition of the compatibility problems between hydrogen and the fuel elements, only the capsule need be considered, because UO_2 is compatible with hydrogen at all temperatures. Other capsule materials will be discussed in a subsequent section.

Mass Transfer of Carbon. The final material problem associated with hydrogen is the possible mass transfer of carbon or carburization of the stainless steels due to methane produced by the reaction between hydrogen and graphite. The reaction which occurs is:

$$2\ H_2 + C\ (\text{graphite}) \rightleftarrows CH_4$$

In the absence of radiation and any metallic catalyst, the reaction of hydrogen with graphite does not occur to any measurable extent at temperatures below 1000°C. If the carbon were present in a more reactive form, such as soot, and/or if metal surfaces present were to act as catalysts, the reaction $C + 2\ H_2 \rightleftarrows CH_4$ could occur at appreciable rates at much lower temperatures. The reaction of protons produced by H_2 irradiation would also be expected to greatly increase the reaction rate. Even if an appreciable reaction were to occur, the equilibrium concentration of methane probably would not be attained, and the deposition of soot by this mechanism would not be a problem. It also appears unlikely that the graphite would react sufficiently to have any detrimental effects on its mechanical properties.

On the other hand, if even a very low concentration of methane is present its decomposition may be catalyzed by radiation or by active metal surfaces, leading to carburization of fuel capsules or other structural metals. Therefore, the effects of radiation on the rate of reaction of hydrogen with graphite and on the rate of decomposition of methane would have to be established before the role of hydrogen as a coolant for a graphite-moderated reactor could be evaluated. Such a program would also include studies of the carburizing effect of methane in high-pressure hydrogen and the catalytic activity of metals in the decomposition of methane in H_2-CH_4 mixtures. The advantages of coating or canning of the graphite are discussed in 15.4.

D. Nitrogen - If nitrogen is used as a coolant, the compatibility of the coolant with graphite is no longer a problem, since all available information indicates that nitrogen is inert toward graphite at high temperatures. In addition, nitrogen is completely compatible with BeO.

Unfortunately, the compatibility of nitrogen with metals at elevated temperatures is open to question, for it is possible that metals such as the stainless steels may be nitrided by a nitrogen atmosphere in a radiation

field at the pressures to be employed. The brittle nitrides formed would produce a severe loss in ductility of the metals.

An investigation of the process of nitriding of structural materials by a nitrogen atmosphere at elevated temperatures has been made by Vetrano.[10] Results showed that nitriding of type 318 stainless steel could be prevented at 1650°F by the addition of 0.5 vol % oxygen. Small additions of water vapor were also found to be an effective preventive. Thus, in the absence of irradiation, nitriding does not appear to be a serious problem with steels; but for in-pile use investigation is needed. It would, of course, be a serious problem if reactive metals such as zirconium were to be used as a structural material. Nitrogen appears to be particularly favorable for use in a gas-turbine system, because conventional air turbines could be used without modification for this service.

E. Air - There is little justification for serious consideration of air as a reactor coolant except in those special cases where the coolant gas would be replaced with a high frequency in remote areas. Although air is cheaper than any other gas, it suffers from the same disadvantages as nitrogen and in addition, the oxidizing effects of oxygen preclude the use of bare graphite as moderator and impose additional restrictions on the choice of metals and alloys. As discussed previously, the argon content of air also would lead to a level of radioactivity that would complicate servicing of the coolant circuit.

In conclusion, it appears that CO_2, H_2, and N_2, in addition to helium, all have good potentials as gas coolants for future types of gas-cooled reactors provided that certain modifications of the reference design are made and provided that experimental studies of the compatibilities of the materials in question give favorable results.

F. Water Vapor - It can be seen from the data of Table 15.1 on the channel diameter and pumping power that water vapor is approximately on par with carbon dioxide as a reactor coolant. Aside from its inexpensiveness

---

[10] J. B. Vetrano, Battelle Memorial Institute, Private Communication, October 1957.

and availability, its principal advantage would be to permit use of a Loeffler boiler[11] or similar cycle in which the reactor functioned as a steam superheater. This would allow steam to enter the power turbine at full reactor discharge temperature and would eliminate the multi-tube steam generators. Such an arrangement would be applicable to small, high-pressure reactor systems.

Water vapor has the disadvantage of undergoing the following reactions with hot graphite:

$$H_2O + C = CO + H_2$$

$$2H_2O + C = CO_2 + 2H_2$$

Both reactions proceed somewhat more rapidly, perhaps by a factor of two, than the graphite carbon dioxide reaction. This would require suitable coating of the graphite, or, alternatively, a different moderator material compatible with water vapor could be used.

15.3 Advanced Fuels

Advanced nuclear fuels must have at least one of three definite advantages over present bulk UO_2 fuels: (1) lower fabrication costs, (2) better nuclear properties, or (3) higher operating temperatures. To be considered for use in a reactor, a fuel with one or more of the above properties must also be stable under irradiation, exhibit no phase transformations, and be compatible with the environment. A list of possible fuels is shown in Table 15.2 as well as the approximate maximum temperature at which each fuel could be considered with possible coolants. In view of the lack of information on many of these materials, the values given are for the most part educated guesses. If the maximum temperature is set by a reaction between the fuel and its environment, the value given in Table 15.2 represents the surface temperature. In those cases where the fuels are stable in a given coolant at all temperatures, the values given in Table 15.2 represent the maximum safe operating temperature of the core. These maxima are set by the melting points, the softening

---

[11] W. N. Barnard, F. O. Ellenwood, and C. F. Hirshfield, Heat-Power Engineering, Part 2, (New York: John Wiley and Sons, Inc., 1935) p 709.

TABLE 15.2

ADVANCED FUELS AND THEIR COMPATIBILITIES WITH PROPOSED COOLANTS

Fuel	Coating or Capsule	Maximum Temperature °C in				
		CO_2	H_2	He	N_2	Air
UO_2	(A)	(B)	(B)	(B)	(B)	(B)
UC	(A)	(B)	(B)	(B)	(B)	(B)
UC_2	(A)	(B)	(B)	(B)	(B)	(B)
U_3Si	(A)	(B)	(B)	(B)	(B)	(B)
UC-Graphite	None (C)	500	> 1000	> 3000	> 3000	300
UC-Graphite	Graphite	500	> 1000	> 3000	> 3000	300
UO_2-Graphite	None (C)	500	> 1000	2500	2500	300
UO_2-Graphite	Graphite	500	> 1000	2500	2500	300
UO_2-ThO_2	None (C)	> 2000	> 2000	> 2000	> 2000	> 2000
UO_2-BeO	None (C)	1600	1600	1600	1600	1600
UO_2-BeO	BeO(D) (D)	1600	1600	1600	1600	1600
UO_2-Al_2O_3	None (C)	1850	1850	1850	1850	1850
UO_2-Al_2O_3	Al_2O_3	1850	1850	1850	1850	1850
UO_2-MgO	None (C)	1400	1400	1400	1400	1400
UO_2-MgO	MgO	1400	1400	1400	1400	1400
UO_2-Si-SiC	None (C)	1300	1300	1300	1300	1300
UO_2-Si-SiC	Si-SiC	1300	1300	1300	1300	1300
UO_2-Cr-Al_2-O_3	None (C)	1500	1500	1500	1500	1500
UO_2-Cr-Al_2-O_3	Cr-Al_2O_3	1500	1500	1500	1500	1500

(A) Capsule required due to reactivity or for structural reasons.
(B) Determined by the capsule material.
(C) Some fission-gas release expected.
(D) Coolants must be dry. BeO is volatile in the presence of moisture above 1200°C.

points, and the temperatures at which the vapor pressure of one of the components becomes excessive. Some of the advantages and disadvantages of these fuels will now be discussed.

15.3.1 Fuels with Lower Fabrication Cost: Due to the inherent chemical stability of UO_2 and the ease with which it may be prepared by various means, it is unlikely that other refractory uranium compounds will replace UO_2 for economic reasons alone. However, improvements in UO_2 as a fuel material could be made by developing more economical methods of fabrication of high-density fuel slugs or by developing new geometries. Other methods of fabricating UO_2, which should be investigated, include hydrostatic pressing, swaging, and extrusion. New geometries of UO_2, such as plates or tubes, would result in more efficient cooling which would lower temperature gradients and thus permit higher operating temperatures or heat fluxes without melting. Such geometries were ruled out for the GCR-2 because of the higher fabrication costs, but further development work to reduce these costs is in order.

15.3.2 Fuels with Improved Nuclear Properties: There have been many fuels proposed with better nuclear properties than UO_2 which are as yet undeveloped. In certain of these fuels the better nuclear properties are a result of a higher uranium density. The most promising of these are UC, UC_2, and U_3Si. However, these compounds are more difficult to fabricate than UO_2, have lower melting points, and some type of capsule would be required due to their chemical reactivity. Also, preliminary results indicate that the radiation stability of U_3Si is poorer than UO_2, while the radiation stability of the carbides is unknown.

Improvements in the nuclear properties could be achieved also through the use of a breeder fuel such as a mixture of UO_2 and ThO_2, or by incorporating the moderator into the fuel by the use of UO_2-BeO or UC-C compositions. Preliminary tests on compacts of UO_2 and graphite also have shown promise. These fuels have the disadvantage that the uranium density would be lower and that some fission-product release would be expected unless a capsule material were used. These disadvantages might be offset by the improved nuclear properties.

15.3.3 High-Temperature Fuels: Some high-temperature fuels have already been mentioned. Others which could be operated at very high temperatures without a capsule material and which would maintain their structural integrity include homogeneous mixtures of UO_2-Al_2O_3, UO_2-MgO, UO_2-Si-SiC, and UO_2-Cr-Al_2O_3. Preliminary studies on these materials indicate that in addition to their very high melting points they have good oxidation resistance and are stable under irradiation. Like the compositions mentioned above, these fuels would require more highly enriched uranium and would probably release some of their fission products. If this latter effect were undesirable it is possible that these fuels could be coated with a ceramic or cermet which would contain the fission products. The most promising coating or capsule material for each fuel is also given in Table 15.3.

Since very little experimental work has been done on these fuels, an extensive program would have to be carried out to determine the possibilities of damage from radiation, fission products, and thermal cycling. Compatibility and fission-gas-release studies would be required also.

15.4 Alternate Moderator Materials

Three materials - graphite, heavy water, and beryllium oxide - appear attractive for use as moderators in the advanced reactor designs. Of the limited number of materials that could serve as moderators, ordinary water possesses no advantage over heavy water other than its much lower cost, and has decidedly inferior nuclear properties. Similarly, beryllium metal and beryllium carbide possess moderating properties comparable to those of beryllium oxide, but the former are basically incompatible with nearly all structural materials at the temperatures of interest. Some estimated maximum service temperatures are listed in Table 15.3 for bare graphite, coated and canned graphite, heavy water, and beryllium oxide. A brief discussion of each of these materials related to the limiting conditions and the potentialities of their use in advanced reactor concepts is given in the following sections.

TABLE 15.3

COMPATIBILITY OF MODERATOR MATERIALS WITH COOLANTS

Maximum Temperature for Continuous Service of Selected Moderator Materials in the Coolants of Interest, °C

Moderator Materials	Coolants				
	CO_2	H_2	He	N_2	Air
Graphite	500	> 1000	~ 3000	~ 3000	300
Coated Graphite*	1350	> 1000	1350	1350	1350
Canned Graphite**	~ 800	> 1000	> 800	~ 750	~ 800
D_2O†					
BeO‡	1600	1600	1600	1600	1600

* Si-SiC coating.
**Ten-mils stainless steel can.
† Insulating materials used would establish the temperature.
‡ Coolants must be dry. BeO is volatile in presence of moisture at temperatures above 1200°C.

15.4.1 Graphite: Graphite is readily available at moderate cost with adequate purity and density. Its moderating properties, while somewhat inferior to those of beryllium oxide and considerably poorer than those of heavy water, are quite adequate for use in various reactor designs. It also possesses good mechanical properties, thermal stability, and is a good conductor of heat. As seen from the limiting temperatures given in Table 15.3, graphite is oxidized by air at relatively low temperatures and by carbon dioxide at somewhat higher temperatures. Although little is known about the rate of reaction of hydrogen with graphite, out-of-pile information indicates that hydrogen may be used in contact with graphite to temperatures above 1000°C. In practice, adequate measures to eliminate such contaminants as water vapor, oxygen and carbon dioxide would have to be taken. It is evident that if bare graphite is to serve as a moderator in a high-temperature reactor (~ 800°C), the coolant choice would be limited to pure helium, hydrogen, or nitrogen.

15.4.2 Coated or Canned Graphite: In the preceding section, the limitations imposed by the use of bare graphite as a moderator were discussed. Coating of the graphite with siliconized silicon carbide or other suitable materials would permit operation at considerably higher temperatures without attack by the oxidizing gases. However, the coating must be impervious to protect the graphite over long periods of time, since a permeable coating would allow subsurface oxidation of the graphite. Such oxidation would result in mass transfer of carbon and would affect the mechanical properties of the graphite.

Coatings for structural graphite are being actively developed by several vendors with varying success. Silicon, zirconium, titanium, and niobium have been studied as the base material for carbide coatings. The most promising coating at present is siliconized silicon carbide, which when properly applied is adherent and nonpermeable. However, it has been successfully applied only on a laboratory scale and only on a special grade of isotopic graphite. Further development work will be necessary to perfect a reliable coating.

Another method of protecting the graphite from attack by the coolant entails enclosing the graphite in a suitable metal can. The metal selected must be resistant to attack by the coolant. An examination of the data given in Table 15.4 in the discussion of Capsule Materials, 15.5, suggests that stainless steel is probably the most suitable canning material available. Use of this metal would introduce an appreciable quantity of a neutron poison into the core but this disadvantage might be more than balanced by increased operating temperatures. These higher temperatures could be achieved with the reactive coolants as well as the inert gases. However, diffusion of hydrogen through the stainless steel can might lead to formation of methane within the can and subsequent carburization of the stainless steel at high temperatures.

15.4.3 Heavy Water: Until recent years the supply of heavy water was limited but with the development of improved processes and expanded production facilities, the supply is ample for use as a moderator in large reactors. The main advantage of the use of heavy water lies in its excellent moderating and reflecting properties. The resonance-escape probability and the thermal utilization are both larger for heavy water than for graphite when used as moderators under equivalent conditions. The high unit cost of heavy water is not a serious disadvantage in view of its superior nuclear properties. However, the relatively low boiling point is a serious drawback introducing fabrication and construction problems if used in high-temperature reactors. Fully pressurized systems would require a high degree of reliability in joints and seals to minimize leakage. On the other hand, the insulation required around coolant channels in an unpressurized system, plus the extra thickness of channel wall for containing the pressurized coolant, would introduce significant amounts of nuclear poison as well as complicating the design and fabrication. In addition, the deuterium and oxygen produced by decomposition under irradiation could augment the corrosion problems.

15.4.4 Beryllium Oxide: The nuclear properties of beryllium oxide, while not as good as those of heavy water, are adequate for its use as a moderator. As seen from the data in Table 15.3, beryllium oxide possesses

excellent resistance to various gases at high temperatures. The gases must be dry, however, to take full advantage of the stability of beryllium oxide. Fabrication technology for beryllium oxide is quite advanced, although the unit cost of the finished moderator would be relatively high. Beryllium oxide has other desirable properties; namely, radiation stability and a relatively high thermal conductivity of 0.10 cal/cm/sec/°C at 600°C. In spite of this, high-density grades of BeO have poor resistance to thermal stresses because of the high elastic modulus (45×10^6 psi) and relatively high coefficient of thermal expansion (10.8×10^{-6}/°C up to 1000°C). Thus, cracking from thermal stresses or thermal shock is a definite problem.

15.5 Capsule Materials

A survey of capsule materials for advanced reactor designs should necessarily include beryllium, aluminum, magnesium, zirconium, and niobium because of their attractive nuclear properties. Others which will be discussed also are molybdenum, stainless steel, graphite, and combinations of coated and canned graphite because of their excellent mechanical properties at high temperatures. Each of these has unique problems of compatibility with coolant materials or fuel, and in some cases are not readily available or have poor fabrication technology. The maximum service temperatures for the various capsule materials are summarized in Table 15.4. The limiting conditions for each will be discussed in the following sections and the scope and magnitude of the associated problems will be outlined. Finally, an attempt will be made to estimate the potentialities of each for use in future design studies.

15.5.1 Beryllium: Beryllium generally has poor mechanical properties, but results of recent work have shown that it has useful strength to temperatures as high as 1350°F. The principal problem associated with the use of beryllium is its brittleness, which causes difficulty in fabrication and low ductility in service conditions. Techniques have been developed for the fabrication of sheet and tubing with highly oriented structures which have useful ductility in one or two directions.[12] However, present fabrication

---

[12] D. W. White, Jr., and R. E. Burke, The Metal Beryllium, Am. Soc. Metals (1955).

TABLE 15.4

MAXIMUM TEMPERATURE AT WHICH CAPSULE MATERIALS ARE COMPATIBLE WITH VARIOUS COOLANTS, °C

Capsule Material	Coolants				
	CO_2	H_2	He	N_2	Air
Be*	600	600	600	600	~ 500
Al	600	600	600	600	600
Mg	550	600	600	400	400
Zr	400	200	> 800	500	400
Cb	~ 550	200	> 2000	~ 650	300
Mo	400	> 2000	> 2000	> 1500	300
Stainless steel	850	> 1000	> 1000	850	850
Graphite	500	> 1000	> 3000	> 3000	300
Coated graphite**	1350	> 1000	1350	1350	1350
Al and graphite	600	600	600	600	600
Zr and graphite	400	200	> 800	400	400
Stainless and Graphite†	~ 800	~ 800	~ 800	~ 750	~ 800

* Be is limited to a temperature of 600°C because of the volume expansion caused by n - α reactions.

** Coated graphite would be limited only by the serviceability of the coating.

† A stainless steel-graphite composite would be limited by the carburization reaction causing embrittlement of the stainless steel capsule.

technology of tubing is sketchy, and much remains to be done both on primary fabrication and joining.

Preliminary work has indicated that the use of beryllium would be limited to temperatures of 600°C and below, because of volume changes produced by the $n \rightarrow \alpha$ reaction.[13] All prospective coolants except air would be compatible at or somewhat above this temperature. Oxidation beginning around 500°C would limit its use in air.

If material with suitable mechanical properties can be developed to the stage where design studies are possible, considerable investigation of radiation effects will be necessary. Results of only one series of experiments on irradiation of beryllium showed that the normally low tensile elongation (ductility) was reduced by a factor of five.[14]

15.5.2 Aluminum and Magnesium: In general, aluminum and aluminum alloys have very poor mechanical properties above 700°F. The Sintered Aluminum Powder alloys, containing up to 16% Al_2O_3, have useful strength even above 1000°F, however.[15] Technology of these alloys is not well developed, particularly with regard to fabrication of tubing and joining by welding or brazing.

The present use of aluminum is limited by its incompatibility with fuel materials. Aluminum reacts with uranium at 350°C and with UO_2 beginning at 500°C to form intermetallic compounds. Use of barrier layers or coatings would be necessary for full utilization of aluminum as a capsule material. Its compatibility with coolant gases is good with the possible exception of hydrogen. Hydrogen from moisture or other vapors may be taken into solution in aluminum, causing brittleness and dimensional changes (blistering).

---

[13] H. M. Finniston, Private Communication (November 1957).

[14] M. H. Bartz, "Radiation Damage Observations at the MTR," Papers Prepared for Radiation Effects Review Meeting, Congress Hotel, Chicago, July 31 – Aug. 1, 1956, TID-7515, Part 1, 19 (August 1956).

[15] E. Gregory and N. J. Grant, "The High-Temperature Strength of Wrought Aluminum Power Products," J. Metals, 6, No. 2, 247 (February 1954).

The problems of utilizing magnesium as capsule material have been adequately discussed in previous reports.[16,17] Magnesium has the definite advantage of good compatibility with uranium metal. The extent of its reactions with UO_2 up to 600°C needs further investigation, however.

15.5.3 Zirconium: Zirconium alloys have considerable mechanical strength at temperatures up to 1000°F. It has fair corrosion resistance to high-temperature water, in which it has been used quite successfully, and its fabrication technology has been quite well developed.

The problems associated with the use of zirconium in gas-cooled systems result from its basic incompatibility with most coolants. The high chemical activity of zirconium causes absorption of all but the noble gases at moderate temperatures, with resultant embrittlement of the base metal. Exposure of zirconium to a number of coolants,[18] including air, CO_2 and hydrocarbons, at 600°C resulted in a weight gain of about 1 mg/cm^2 in 24 hr, while oxygen under the same conditions produced a weight gain of 9 mg/cm^2 in 24 hr. Nitrogen was absorbed at a much lower rate, even at 700°C, but at higher temperatures, absorption proceeds readily. Hydrogen, on the other hand, is absorbed rapidly at about 250°C to form the extremely brittle hydride.

Zirconium is also reactive with UO_2 at the temperatures of interest. The UO_2 is reduced and taken into solution, forming brittle, pyrophoric dispersed phases in the base metal. At temperatures above 1100°F, the reaction is significant,[19] while at 1300°F it is estimated that a 0.030-in. layer of zirconium will be penetrated in 300 days.

Thus it appears that zirconium would be useful only in a gas-cooled system cooled with helium, and then only if contaminants in the coolant are

---

[16] T. Bishop, "Metallurgical Aspects of Calder Hall," Metal Progress, 71, 65 (June 1957).

[17] J. G. Ball, "Metallurgical Research in Nuclear Power Production," J. Inst. of Metals, 84, 239 (1956).

[18] B. Lustman and F. Kerze, Jr., Metallurgy of Zirconium, National Nuclear Energy Series VII (New York: McGraw-Hill, 1955).

[19] M. W. Mallett et al., Kinetics of the Zr-UO_2 Reaction, BMI-1028, (August 15, 1955). Declassified without deletions TID-1147.

carefully controlled. The reaction between UO_2 and zirconium could be controlled with suitable coatings on the fuel slugs.

15.5.4 Niobium and Molybdenum: Niobium and molybdenum have very high strength at elevated temperatures. However, both are comparatively expensive and have not been fully developed or utilized on a large scale. Both are completely compatible with UO_2, and do not react with uranium metal up to its maximum usable temperature in a reactor.

Niobium has much the same disadvantages as zirconium, and is somewhat more expensive. It is a strong hydride former and oxidizes rapidly in air at temperatures above 400°C. Niobium is somewhat less reactive with nitrogen or CO_2, but at higher temperatures (~ 800°C) these gases are absorbed and produce an embrittlement from the nitrides, carbides, and oxides which are formed. Thus the use of niobium in advanced studies is not encouraging, because of its high chemical activity and high cost. However, oxidation-resistant alloys of niobium with zirconium, aluminum, and other additions show some promise for future use.

Molybdenum, on the other hand, is completely compatible with hydrogen and has been found to be unreactive with nitrogen at high temperatures.[20] However, it is subject to catastrophic oxidation in air or CO_2 even at moderate temperatures. Other disadvantages of molybdenum are its lack of ductility and high transition temperature. Successful joining is also very difficult. Thus the use of molybdenum appears unlikely except for possible designs operating at very high temperatures to utilize its high-temperature strength.

Considerable study of the radiation effects of both niobium and molybdenum would be necessary, since the ductility of body-centered cubic metals is extremely sensitive to impurities. Neutron irradiation of steels[21] and molybdenum[22] invariably results in embrittlement as manifested by

---

[20] R. M. Parke et al., "Molybdenum and Its Alloys," The Reactor Handbook, 3, Section 1, 210 (1955).

[21] R. A. Berggren and J. C. Wilson, Recent Data on the Effects of Neutron Irradiation on Structural Metals and Alloys, ORNL CF 56-11-1 (January 30, 1957).

[22] C. A. Bruck et al., "Variations in Radiation Damage of Metals," Trans, AIME, 206, 1362 (1956)

increased transition temperature and reduced fracture energy. Annealing effects in steels are significant at 500°F, but will probably not be noticeable with niobium or molybdenum except at much higher temperatures.

15.5.5 Stainless Steel: As discussed previously in Section 3, stainless steels have more than sufficient mechanical properties, and would be quite compatible with the coolants of interest except under carburizing conditions. There is evidence from experience in the petroleum industry that the high-chromium stainless steels (types 309, 310, and 446) resist carburization quite effectively, and that type 304 with 18% Cr may be satisfactory for long periods. Stainless steels have a highly developed fabrication technology as well. Advanced designs should be concerned with stainless steel capsule configurations which possess greater thermal stability than the seven-rod cluster, and which have better heat-transfer characteristics. Designs utilizing finned tubes or slabs would present much more heat-transfer surface and, in the case of the slab geometry, a much shorter thermal path.

15.5.6 Graphite and Coated Graphite: The mechanical properties of graphite are quite favorable. In the temperature range from 1000 – 3000°F, reactor-grade graphite has tensile strengths varying from 700 – 1200 psi in the direction transverse to extrusion and up to about 3000 psi in the extrusion direction.[23] Furthermore, the strengths of all commercial grades have been found to increase steadily with increasing temperature up to 4000 – 4500°F. Coated material or low-permeability grades should have quite similar properties. The fabrication technology of low-permeability graphite is under development, and any future designs utilizing it would necessitate allowance for its inherent lack of ductility.

One of the serious problems in use of graphite as a capsule material is the release of fission gases. It is realized that even the most impervious grades of graphite would contain only a part of the fission gases. Thus the coolant cycle would be highly active unless vented capsules are used.

---

[23] M. C. Udy and F. W. Boulger, The Properties of Graphite, BMI-T-35, (June 20, 1950).

An additional problem is the compatibility of graphite with the proposed coolants. Graphite is quite unreactive with nitrogen, but is limited to 500°C in CO_2, and is useless in air. Oxidation-resistant coatings under development should protect graphite up to 1350°C in CO_2 or air. Thus the use of graphite as a fuel capsule material is quite attractive, and future designs for higher temperatures should be able to utilize its favorable properties.

15.5.7 Canned Graphite: A double-walled capsule utilizing graphite for support and a metal barrier or sheath for containment of fission gases offers unique possibilities in design. Such a composite capsule should comprise an ideal fuel element for certain designs. The materials and configurations would be selected on the basis of compatibility with fuels and proposed coolants, as listed in Table 15.4. The use of CO_2 as a coolant would require that the graphite moderator be at least partially coated or canned for protection.