MASTER

CIVILIAN POWER REACTOR PROGRAM

PART III

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Status Report on Gas-Cooled Reactors as of 1959

UNITED STATES ATOMIC ENERGY COMMISSION

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Published: 1960

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1. INTRODUCTION

Gas cooling embraces a variety of reactor concepts which have been carried to different stages of technological development. It is necessary to consider the following factors in describing the current status of gas-cooled reactors for civilian power generating applications:

- (1) The major United States development effort has been concentrated on reactors which require partially enriched fuel.
- (2) No major United States civilian gas-cooled reactor development projects were initiated until 1957 although the history of United States studies in this field goes back to the Manhattan District era prior to 1946.
- (3) The only concept which has now reached the stage of full scale plant construction and operation is the United Kingdom natural uranium fueled, CO_2 -cooled, graphite-moderated version. No plants of this kind are under construction or planned in the United States. Evaluation studies have indicated that plants of this type are not economically attractive in the United States.
- (4) A development project toward possible construction of a 50 MWE low enrichment D_2O -moderated, CO_2 -cooled prototype for a large-scale natural uranium fueled plant was initiated in this country in 1958.
- (5) The technology of enriched fuel civilian gas-cooled reactors has not been sufficiently developed to permit the construction of a prototype or full-scale plant without concurrent (preoperating) research and development.
- (6) Two development projects leading to the construction of experimental or prototype, enriched fuel, civilian gas-cooled reactors were initiated in 1959 based on the following alternatives:

- (a) Heterogeneous-clad fuel core (UO_2/SS) with complete fission product retention within the fuel elements, using a U 235-U 238-Pu 239 fuel cycle.
- (b) Semihomogeneous graphite-clad fuel core (UC-ThC in graphite) with partial fission product retention within the fuel elements, using a U 235-Th 232-U 233 fuel cycle.

Current status has been defined for the purpose of the long-range program in terms of the technical performance parameters and costs attainable today in full-scale plant construction without preoperating research and development. It is therefore evident, in view of the above facts, that any current status report on the United States gas-cooled reactor program must consist primarily of development work completed and under way on the technical problems of enriched fuel, gas-cooled reactor power plants. No economic status determination is made in this report for plants of this type.

The more highly developed technology of natural uranium fueled, graphite-moderated gas-cooled reactor power plants is summarized primarily for its relevance to the technology of enriched fuel, graphite-moderated systems. Capital and generating costs for the natural uranium fueled plants are included in Sargent and Lundy Report SL-1674, "Power Cost Normalization Studies—Civilian Power Reactor Program—1959."

The technology of D_2O -moderated, gas-cooled reactors is also summarized. Estimated technical performance parameters are given for the enriched fuel prototype and for a large natural uranium fuel plant. No economic status determination has been made in this report for plants of this type.

2. CURRENT TECHNICAL STATUS

2.1. REACTOR PHYSICS

The optimization of a gas-cooled reactor plant with respect to the cost of producing electricity reduces to the selection of a proper balance between capital costs and fuel cycle costs. Once the heat transfer characteristics of various fuel element configurations are established, it becomes a problem of determining how much fuel or heat transfer surface can be put into the reactor and how much energy can be extracted from the fuel elements before they must be removed from the reactor. The main questions then, from the nuclear analysis point of view, have to do with the multiplication factor and the reactivity lifetime of the core as functions of fuel element size, spacing, and enrichment. Other parameters whose effects must be evaluated are cladding thickness, cooling-channel diameter, moderator quality and density, etc. In addition to these major problems, design of the plant requires information on the temperature coefficient of reactivity, requirements for control rods and the effectiveness of various control rod configurations, radiation shielding requirements, radiation heating in various reactor components, and finally, the possible means for obtaining a reasonably uniform power density distribution. The following is a brief status report on these various aspects of gas-cooled reactor physics. The results of all calculations reported herein have been normalized to the available experimental data. All calculations are for heterogeneous-clad systems, although the same techniques are directly applicable to semihomogeneous systems.¹

¹For information on special physics problems in semihomogeneous systems, see ORNL-2900, Sec. 4.3.

Multiplication Factor

The multiplication, or reproduction, factor of the fuel lattice is expressed by the four-factor formula

$$k = \eta \epsilon p f,$$

where

η = direct yield of fission neutrons per neutron absorbed in fuel,

ϵ = fast fission factor,

p = resonance escape probability,

f = thermal utilization, i.e., the ratio of the number of thermal neutrons absorbed in the fuel (excluding those absorbed in U 238 resonances) to the total number absorbed in the lattice (excluding those absorbed in the resonance of U 238 and stainless steel).

Neutron Yield. The yield of neutrons per neutron absorbed in the fuel, η , has been obtained directly from the measured neutron cross sections. The values of η used for gas-cooled reactor design are given in Figure 2.1 as a function of enrichment.

Fast Fission Effect. The fast fission factor, ϵ , expresses the increase in the net number of fission neutrons available to the chain reaction as a result of fissions produced in U 238 and U 235 by fast neutrons ($E_n > 1$ Mev). Values of the fast fission factor for single UO₂ rods are given in Figure 2.2. For a cluster of rods, the value of ϵ is expected to be between the value for a single rod of the cluster and the value for a single rod of the same total cross sectional area as the cluster, and it will be somewhat closer to the latter value. The values for clusters of seven rods of various sizes are given in Table 2.1.

Resonance-Escape Probability. The resonance-escape probability, p , for a lattice of infinite extent is here defined as the proba-

TABLE 2.1.—FAST FISSION FACTOR FOR CLUSTERS OF ${}^1_0\text{UO}_2$ RODS

Rod diameter (in.)	ϵ	
	For cluster	For equivalent single rod
0.5.....	1. 016	1. 018
0.75.....	1. 024	1. 026
1.00.....	1. 028	1. 034

bility that a neutron produced in fission will slow down to an energy below the lowest resonance of U 238, i.e., below about 6 ev, without capture in the resonances of U 238 or of stainless steel. Epithermal captures in U 235 or in any of the plutonium isotopes are disregarded in calculating the resonance-absorption rate, because the effect of such captures on k is taken into account in calculating the thermal utilization. The same argument applies to captures in U 238 and in stainless steel that are attributable to the $1/v$ portion of their epithermal capture cross sections.

FIGURE 2.1.—Fission neutrons produced per neutron absorbed in uranium.

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FIGURE 2.2.—Fast fission factor for single UO_2 rods.

The problem of computing p reduces essentially to that of determining the effective resonance integral of U 238. The theoretical treatment² of the subject is quite complex, and there are rather serious potential ambiguities associated both with the use of UO_2 , instead of the metal, and with the use of the cluster geometry. It is fortunate that the results of measurements made at Stockholm, by Hellstrand,³ on the resonance absorption of U 238 in metallic and oxide rods in various geometries, including seven-rod clusters, have recently become available. Hellstrand has found empirically that the experimental resonance integrals, σ_r , for lumps of both metal and oxide, correlate with the expression,

$$\sigma_r = A + B \left(\frac{S}{M} \right)^{1/2},$$

² E. P. Wigner *et al.*, *J. Appl. Phys.* 26, 260 (1955).³ E. Hellstrand, "Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries," AEF-S4, *AB Atomenergi*, Stockholm (June 1957).

over a wide range of S/M (the surface-to-mass ratio of the fuel element). With a $1/v$ contribution of 1.2 barns deducted from the absorption term, the resonance integral for lumps of UO_2 is found by Hellstrand to be

$$\sigma_r = 4.05 + 26.6 \left(\frac{S}{M} \right)^{1/2} \quad 0.08 \leq S/M \leq 0.7.$$

The experimental correlations of Hellstrand confirm a proposal by E. R. Cohen (presented in ref. 3) that the surface to be used for a cluster of rods is the "rubber band" surface, i.e., the surface of a membrane stretched tightly around the cluster. The validity of the membrane analogy has been established for relatively heavy cladding materials (e.g., stainless steels), but not for lighter materials (e.g., beryllium) having a considerable moderating effect.

The temperature dependence of the resonance-escape probability is quite important because it influences the change in reactivity of the reactor between room temperature and operating temperature, and because it contributes a significant negative component to the temperature coefficient of reactivity at operating temperature. The problem consists of two parts: (1) determining the temperature dependence of the effective resonance integral and (2) determining the effective temperature of the fuel.

The temperature coefficient of the effective resonance integral of uranium metal is taken from the work of Davis.⁴ Between 40 and 300° C, Davis finds that

$$\frac{1}{\sigma_r} \frac{d\sigma_r}{dT} \sim (1.56 \pm 0.12) \times 10^{-4}/^\circ\text{C};$$

the temperature coefficient is constant over this range.

The mean temperature of the fuel is usually determined rather approximately for an average heat flux. An average temperature is calculated for a single rod with a resonance-absorption distribution $[b(r)]$ taken from Hellstrand's

measurements.³ The temperature distribution is considered to be parabolic,

$$T(r) = T_0 - ar^2,$$

where T_0 is the central temperature of the fuel rod, and the constant (a) contains information about the conductivity and the power density in the rod. With

$$\bar{T} = \frac{\int_0^R T(r) b(r) r dr}{\int_0^R b(r) r dr},$$

where R is the rod radius, it can be shown that

$$\sigma_r(T) = \sigma_r(20^\circ \text{C}) \exp [+ 1.56 \times 10^{-4} (\bar{T} - 20^\circ \text{C})].$$

Thermal Utilization. The thermal utilization of a lattice cell is defined as the ratio of the number of thermal neutrons absorbed in the fuel of the cell, including those absorbed in the $1/v$ part of the U 238 captures, to the total number of thermal neutrons absorbed in the cell, i.e., those absorbed in the $1/v$ part of the U 238 captures and in the cladding. In terms of the appropriate cross sections, the thermal utilization is written as

$$f = \frac{V_f \sum_a' \Phi_f}{V_f \sum_a' \Phi_f + V_m \sum_a^m \Phi_m + V_p \sum_a^p \Phi_p},$$

where $\sum_a'$, $\sum_a^m$, and $\sum_a^p$ are, respectively, the macroscopic effective absorption cross sections in the fuel, moderator, and poisons (e.g., cladding material). The Φ_i terms are the respective average effective fluxes, and the V_i terms are the volumes of the respective regions. The effective cross sections used in room temperature calculations must be averaged over a Maxwellian flux spectrum, with $T_n = 20^\circ \text{C}$. High temperature calculations are done with suitably averaged high temperature cross sections.

Cross Sections. Cross sections at operating temperature are needed for calculations of initial reactivity and for studies of reactivity vs. exposure. For most purposes, it is convenient to use effective one-group cross sections. Thus the resonance-absorption effects of all materials other than U 238 and stainless steel are included

⁴M. V. Davis, *J. Appl. Phys.* **2**, 250 (1957).

in effective cross sections, $\hat{\sigma}$, as defined, for example, by Westcott:⁵

$$\hat{\sigma} = (g + rs)\sigma_{2200}.$$

The flux spectrum is assumed to consist of a Maxwellian component at temperature T_n , plus a dE/E tail cut-off at a lower limit of $5kT_n$. The factor g takes into account the averaging of the cross section over the Maxwellian part of the neutron distribution; r is $(5\pi)^{1/2}/4$ times that fraction of neutrons which lies in the dE/E tail; and s depends on the resonance integral

$$\int_{sKT_n}^{\infty} \sigma dE/E.$$

For a $1/v$ absorber, $g=1$ and $s=0$. (The factor is always positive; but s is negative when the resonance integral is less than the corresponding integral obtained with σ set equal to $g\sigma_{2200}v_{2200}/v$.)

The assumption of this simplified form for the spectrum bypasses such questions as how the resonance absorptions distort the dE/E tail, how this tail fits on to the Maxwellian component, and how the Maxwellian component is distorted inside the fuel element both by preferential absorption of lower energy neutrons and by scattering of neutrons with energetic atoms in the high temperature fuel rods. However, even the determination of the parameters T_n and r for the simplified spectrum poses nontrivial problems.

The temperature T_n which best describes the thermal neutron distribution may be significantly different from the temperature T_m of the moderator atoms. For an infinite homogeneous medium with $1/v$ capture cross section, Coveyou *et al.*⁶ found that

$$T_n \sim T_m(1 + 1.11 AK).$$

Here A is the mass number of the scatterer, and

$$K = \left(\frac{2}{3}\right)^{1/2} \left(\frac{293}{T_m}\right)^{1/2} \frac{\sum_a}{\sum_s},$$

$\sum_a$ (2200 m/sec) and $\sum_s$ are the macroscopic cross sections of the absorbing and scattering species. The reliability of this formula for com-

puting T_n has been confirmed experimentally for homogeneous mixtures.⁷ Although its usefulness for heterogeneous reactor calculations is uncertain,⁸ it is considered the best guide available at the present time. The formula is applied to the heterogeneous reactor under consideration by replacing its lattice cell with one in which all materials are uniformly distributed over the cell, and K is then computed with flux-weighted cross sections. Typical results are given in Table 2.2.

Graphs of $\hat{\sigma}$ vs. neutron temperature and r for U 235, Pu 239, Pu 240, and Pu 241 are given in Figure 2.3a-d. These graphs are based on Westcott's results,⁵ with the cross sections from 400 to 600° C calculated from extrapolations of Westcott's g and s factors. In the case of Pu 239, where the extrapolations seem questionable, the point at 600° C was independently calculated by numerical integration of the cross section⁹ over the Maxwellian spectrum.

TABLE 2.2.—NEUTRON TEMPERATURE CALCULATIONS FOR AN 8-IN. LATTICE PITCH, SEVEN 3/4-IN.-DIA. UO₂ RODS, A 3 1/4-IN.-DIA. COOLING CHANNEL, AND A MODERATOR TEMPERATURE, T_m , OF 750° F

U 235 (at. %)	T_n/T_m	$T_n(^{\circ}\text{K})$
1.2-----	1.164	782
1.5-----	1.179	792
2.0-----	1.204	809
2.5-----	1.221	821

Lattice Calculations. The methods outlined above have been used to analyze a variety of lattices fueled with clusters of UO₂ elements that would be suitable for partially enriched gas-cooled reactors. The detailed results of

⁷ M. J. Poole, *A Measurement of the Neutron Spectra in Aqueous Moderators and Reactor Lattices by Time-of-Flight Methods Using a Pulsed Linear Accelerator*, NRDC-95 (1956).

⁸ Recent information received from Harwell, by private communication, suggests that a slightly better fit is obtained experimentally for heterogeneous graphite lattices with $T_n \sim T_m (1 + 1.08 AK)$.

⁹ D. J. Hughes and J. A. Harvey, *Neutron Cross Sections*, BNL-325 (July 1, 1955).

⁵ C. H. Westcott, *The Specification of Neutron Flux and Effective Cross Sections in Reactor Calculations*, ORRP-662 (Aug. 15, 1956).

⁶ R. R. Coveyou, R. R. Bate, and R. K. Osborn, *J. Nuc. Eng.* 2, 153 (1956).

FIGURE 23.—Effective cross sections of uranium and plutonium vs. neutron temperature. (a) U^{235} . (b) Pu^{239} . (c) Pu^{240} . (d) Pu^{241}

these lattice calculations, which are presented in Appendix A, check very well with the available experimental data. In view of the extensive experience with single-rod lattices, it is useful to summarize the principal features of these calculations, because the lattices with clusters exhibit the same general characteristics as those with single-rod fuel elements. However, certain interesting differences do appear. In particular, the balance between thermal utilization and resonance-escape probability is different. On the one hand, f is greater for clusters than for single rods having the same volumes of fuel and cladding

material. For clusters, the cladding is, in effect, uniformly mixed with the fuel. For single rods the cladding lies entirely on the outer surface. On the other hand, p is smaller for clusters, because the surface-to-mass ratio is greater (at least for the rod spacings of interest in gas-cooled reactors). Consequently, for large lattice spacings, clusters have a higher k than equivalent single rods, whereas, for small lattice spacings, single rods have the higher k . A comparison of k for rods and clusters is given in Figure 2.4, and, in Figure 2.5 it is shown that for equal k values the cluster has the higher conversion ratio.

FIGURE 2.4.—Comparison k values for clusters of rods and single rods of equal volume.

FIGURE 2.5.—Comparison of conversion ratios for single rods and clusters of rods.

The point that enrichment of the fuel to about 1.0 percent U 235 is required to accommodate 0.020-inch-wall stainless steel cladding is illustrated in Figure 2.6. Enrichment above 1.0 percent is therefore chargeable to the production of a higher power density and to increasing fuel lifetime. Curves that correlate the conversion ratio and the multiplication factor as functions of lattice pitch and enrichment are presented in Figure 2.7 for 7-rod

clusters with 0.020-inch-thick cladding and an average neutron temperature of 540° C.

The need for a volume in which fission product gases can accumulate has led United States designers to utilize annular fuel pellets in these reactors. The effect on reactivity of providing a central hole in each of the fuel rods has been studied, and the results for 0.75-inch-diameter rods are shown in Figure 2.8. It is apparent that small holes make practi-

FIGURE 2.6.—Effect of cladding thickness on multiplication factor.

FIGURE 2.7.—Multiplication factor vs. lattice pitch and enrichment, $T_a = 814^\circ\text{K}$.

cally no difference in multiplication factor, but that a significant reduction in k occurs for holes with diameters between 0.3 and 0.4 inch. A $\frac{5}{16}$ -inch diameter hole in each rod is a reasonable compromise between the requirements for gas accumulation volume and the need to maintain reactivity. The loss in k is 0.9 percent.

Reactivity Limitations on Fuel Exposure

The values of the multiplication factor discussed above are of interest to the extent that they help to determine the amount of reactivity that the control rods must override, for example, but the reactivity of the system is significant chiefly through its affect on the attainable fuel exposure. The resistance of UO_2 fuel elements to unfavorable changes under irradiation makes possible very much longer fuel exposures than could be achieved with uranium metal fuel. Indeed, the relatively high fabrication cost expected for UO_2

fuel, as compared, for example, with the cost of Magnox-clad uranium metal elements, makes long fuel irradiation a very urgent objective. One of the major problems in the nuclear analysis, therefore, is to obtain reasonably reliable estimates of the reactivity lifetime of the various fuel configurations and to determine the isotopic composition of the fuel as a function of exposure. The fuel exposure is expressed as the total thermal energy in megawatt days per metric ton of uranium fuel (Mwd/MT) extracted from the fuel before discharge from the reactor. It is sometimes expressed as the energy obtained from the fuel per gram of $\text{U } 235$ in fresh fuel, Mwd/g of $\text{U } 235$. This may exceed the amount of energy released by the fissioning of all the $\text{U } 235$ in the fuel, since $\text{U } 238$, via plutonium, is also being consumed. Details and results of such calculations are given in Appendix A.

Power Flattening

One of the possible uses of fuel enrichment is to permit distributions of fissionable material in the reactor that will tend to produce a uniform power density in all fuel elements. The potential advantages of such an arrangement are great, since all heat-transfer surfaces would operate at the maximum allowable heat flux and all fuel elements would operate at the maximum allowable power density. Unfortunately, as implied by the need for fuel enrichment, such uniform power arrangements are always less reactive than configurations having uniform fuel distribution, and it becomes necessary to find an economical compromise between requirements for maximum power density and the need to maintain reactivity. Some of the benefits that accrue from power flattening can be realized, at least in part, by properly matching the flow of coolant in each channel to the power produced in that channel. The degree to which this procedure can approximate the benefits of power flattening depends very much on what the limitations on heat transfer are in the particular system

FIGURE 2.8.—Effect of central hole in fuel rods. (a) Effect on f , p , and ϵ . (b) Effect on multiplication factor.

under study and on the relative magnitudes of certain characteristic temperature differences in the core, e.g., the coolant temperature rise through the core, the film drop between fuel surface and mean coolant temperature, and the internal temperature rise within the fuel element. Broad generalities about the effectiveness of orificing the flow should be interpreted with caution; each system must be studied independently. In the case of the partially enriched thermal converters, the fuel element surface temperature is the principal limitation on heat transfer; internal temperature rise appears to be less crucial (especially if the gap between cladding and UO_2 is very small) since the central oxide temperature is, in any case, well below the UO_2 melting temperature. (Larger fuel-rod diameter, or higher surface-film drop, could change this situation.) Under these conditions, it has been found that the core can be made to produce nearly as much

power with a radial peak-to-average power ratio of 1.3 to 1.5 as with a uniform radial power distribution, provided the coolant flow is properly matched to the power distribution. This matching is best achieved by varying the cooling-channel diameters but can be accomplished almost as well with orifices. It appears, therefore, that the incentive for power flattening is not great, and the criterion has been adopted that as much power flattening should be provided as can be accomplished without significantly sacrificing the effective use of fissionable material in other ways.

The general method for flattening the power distribution is to reduce the infinite multiplication factor at the center of the reactor core and increase it near the boundaries of the core. This may be done in several ways. For a homogeneous system, it may be accomplished by increasing the concentration of the fissionable material toward the outer regions of the

core by poisoning the central region of the core, as with control rods. In a heterogeneous system, it may also be accomplished by varying the lattice pitch so as to modify the balance between thermal utilization and resonance-escape probability. A third approach, for constant lattice pitch and constant fuel enrichment, is simply to allow the fuel near the center to reach a higher exposure level, and hence, lower reactivity than the fuel near the core boundaries.

2.2. HEAT TRANSFER AND FLUID FLOW

It is crucial in the design and prediction of performance of gas-cooled reactors to have an adequate detailed understanding of the heat-transfer and gas-flow mechanisms which limit fuel element performance. The present status of this field is summarized here.

Pressure Drop

It is desirable to minimize the pressure drop in the gas circuit of a gas-cooled reactor because the primary compressors are large-volume, high-head machines which consume significant amounts of power. There are not adequate correlations in the literature for reliable prediction of the pressure drops through flow channels which are packed with complex structures such as fuel elements, hangers, spacers, and flow-shaping devices. Accordingly, full-scale experiments on the proposed heat-transfer design of each gas-cooled reactor have been traditionally made in order to establish the pressure drop precisely. As a result of these experiments, a number of correlations have been partially established which can be used in future reactor designs.

Mechanical Stability

It is desirable from the points of view of cost and neutron economy to make the fuel elements as long as possible consistent with obtaining the required heat-transfer and heat-transport performance of the system. When

the elements are lengthened, the individual capsules tend to bow toward other capsules in the cluster or toward the channel wall. If the capsules should touch one another or touch the channel wall, burnout would occur. Accordingly, in long capsules it is necessary to supply mechanical spacers within the fuel assemblies to avoid burnout due to capsule bowing.

The information on the design requirements and performance of fuel element spacers is wholly experimental at this time. Successful preliminary attempts have been made to correlate the results of experiments on mechanical stability of fuel clusters with the results of calculations. Thus far, however, there is not enough experience in this field to be certain that such correlations are sufficiently reliable that experiments are not necessary.

Hot Spots

The basic limitation of gas-cooled reactors, which arises from the poor heat-transfer characteristics of gases as a whole, is that the hot-spot temperatures which are developed on the surface of the fuel element are very high when compared with the average surface temperature. Up to the present time, much of the heat-transfer work has been aimed at finding a ready solution to the problem of maintaining the hot-spot temperature as close to the nominal maximum surface temperature as possible. In reactors with Magnox-clad fuel elements, this is accomplished by using extended-surface fuel elements. This solution is not possible when the cladding material of the fuel element has a high neutron-absorption cross section, as in the case of stainless steel. It must be recognized in this connection that a large portion of the hot-spot temperature difference in cluster-type elements arises from the geometrical asymmetry of the heat source.

2.3. CORE MATERIALS

The design of any gas-cooled reactor plant must start, as a practical matter, from a con-

cept of a system built from materials which are chemically compatible and which offer the possibility of high power densities and long irradiation lifetimes. It is accordingly necessary to survey the present status of technology of the important materials which enter into gas-cooled reactor designs.

Moderator—Graphite

In all gas-cooled reactors built to date, graphite has been selected as the moderator because of its excellent high-temperature strength, good nuclear properties, and low cost. One important limitation on graphite as a moderator material with CO₂ as the coolant at the present time arises from its rapid oxidation in the presence of CO₂ or free oxygen at surface temperatures above 500 to 550° C. There are numerous attempts under way to develop satisfactory surface coatings for graphite to inhibit this reaction and permit higher temperature operation. The use of helium as the coolant gas avoids this problem. However, there is an additional problem in the differential shrinkage of graphite moderators in power reactors of this type. At moderator temperatures in excess of about 350° C (660° F) graphite shrinks on exposure to fast neutron flux. The rate of shrinkage is a function of irradiation temperature, type of graphite coke, graphitization temperature, and degree of orientation of the particles in the bulk graphite. The major consequences of this differential shrinkage are induced stresses and bowing, which may be alleviated to some extent by relaxation due to graphite creep, and by structural restraint. Remedial methods are being investigated in the Experimental Gas-Cooled Reactor (EG-CR) development program. The service life of gas-cooled reactor cores may be significantly reduced by differential shrinkage. The High Temperature Gas-Cooled Reactor (HTGR) semihomogeneous reactor design avoids this problem as most of the graphite is removed with the fuel and is not subjected to long irradiation flux-times.

Fuel Cladding—Magnox

At the start of the gas-cooled reactor program in England, aluminum, magnesium, beryllium, and zirconium were considered as possible cladding material. The British chose to concentrate on magnesium and developed an alloy containing minor amounts of aluminum, beryllium, calcium, and iron. This alloy was called "Magnox" and its original composition was as follows:

	Content (wt %)	
	Minimum	Maximum
Aluminum.....	0. 7	0. 9
Beryllium.....	0. 002	0. 03
Calcium.....	-----	0. 008
Iron.....	-----	0. 006

The aluminum was added for strength, the beryllium for oxidation resistance, and the calcium for grain refinement. Over the years, the calcium content has been lowered appreciably because it promotes weld cracking. The beryllium content has been lowered to reduce inclusions (which have been quite troublesome in the procurement of acceptable Magnox for the fabrication of capsules).

The single most important variable that has affected the use of Magnox as a fuel-capsule material has been the grain size. The ductility and creep behavior and, to a lesser extent, the corrosion resistance of the capsule depend on the grain size.

In the routine production of fuel capsules, extreme care is used to prevent the following materials from contacting the Magnox: mercury, thorium, tin, bismuth, cadmium, zinc, and solder. All these materials form low-melting eutectics with Magnox. Aluminum, calcium, silver, copper, Monel, and bronzes are also detrimental to Magnox in that they accelerate the CO₂-Mg reaction.

It has been found that Magnox is compatible with CO₂ to temperatures in excess of its

melting point (645°C). Thus its usefulness as a fuel-cladding material does not depend principally upon its corrosion resistance, but rather on its strength characteristics. Typical capsule deformation involves grain boundary migration and vacancy movement. In initial operation of fuel elements in the Calder Hall reactor, 15 to 20 failures occurred among the 30,000 to 40,000 elements in use. Of these failures, roughly 12 could be attributed to deformation of the Magnox cladding. The location of the failed elements was in the low temperature zones of the reactor. This phenomenon has been called "grain boundary cavitation" by the British. Its occurrence has been minimized in present-day Magnox by further refining the grain size.

Welding difficulties have accounted for four to six of the failures of Calder Hall fuel elements. The main welding difficulty has been porosity, which can be corrected by maintaining proper gas coverage and minimizing the possibility of "cratering" during the striking and snuffing of the welding arc.

In summary, Magnox is a good cladding for natural uranium in reactors cooled with CO_2 . The usefulness of the Magnox-clad fuel elements in such reactors is not limited by the Magnox capsule.

Fuel Cladding—Stainless Steel

The selection of stainless steel for encapsulating ceramic fuels represents an intermediate step in the development of gas-cooled reactors. The transition from low-temperature to high-temperature operation is not a simple one, even for relatively uncomplicated designs, because the detailed principles of fuel element design for gas-cooled reactors and metal behavior in gas-cooled reactor environments are not well understood.

Stainless steel possesses many characteristics which are useful in any high-temperature reactor. It is relatively cheap and easy to form and weld. The practices developed through years of experience enable the producer and the fabricator to guarantee a consistent high-

quality product. The various stainless steel alloys have good high-temperature strength. They form a protective oxide film in air at temperatures up to approximately $1,800^{\circ}\text{F}$. Their inherent ductility renders them resistant to thermal and mechanical fatigue at elevated temperatures. Likewise, they can be treated roughly at ambient temperatures without causing brittle fracture. Of specific interest for gas-cooled reactors is the fact that they are compatible with uranium oxide fuel up to at least $2,000^{\circ}\text{F}$. They have the disadvantage of a high thermal-neutron absorption cross section.

Little information is available on the effect of irradiation on the creep and rupture properties of stainless steel. Tests in pure gases such as CO , CO_2 , A , H_2 , and O_2 have shown that wide variations in creep and rupture can occur as a function of the chemical character of the environment.

There are certain problems bearing on the limitation of the fuel element temperature and fuel element life which are interrelated with factors other than alloy composition. For instance, in a gas-cooled reactor the metal capsule cannot be considered as a rigid or elastic member, as it is in water-cooled fuel elements. The temperatures at which gas-cooled elements operate are such that the metal is subject to time-dependent flow under static loads. Thus the magnitude of the initial gap between the fuel pellet and the wall of the capsule is of concern. In service the capsule is subjected to the external pressure of the coolant. This force causes the capsule to become smaller in diameter by increasing the thickness of the wall as well as the length. The process continues until the wall is supported by the fuel pellet. If the gap between the pellet and the capsule is too great, nonuniform deformation will occur and wrinkles will form along the length of the capsule.

An ORNL experimental program to study creep buckling has shown that the gap required for uniform collapse is a function of the external pressure, the temperature, and the

wall thickness of the capsule. The data given below present some of the results obtained using a 300-psig external pressure on $\frac{3}{4}$ -inch-diameter tubing:

Wall thickness (in.)	Temperature (°F)	Maximum diametral gap for uniform collapse (in.)
0.020-----	1500	0.007-0.010
0.010-----	1500	0.002-0.004
0.020-----	1800	0.010-0.013

It is thus evident that the production of tubing and fuel pellets to the required tolerances to prevent wrinkling is important. Achieving tight tolerances is ordinarily expensive. Since the limiting diametral gap between pellet and capsule is a function of both the capsule diameter and its wall thickness, there are limits in cost as well as reliability in the utilization of thin-walled tubing.

Thermal ratcheting and distortion are also problems. Asymmetrical temperature distributions as a result of channeling effects or wrinkles in the capsule can produce bowing in the elements. This effect is progressive and can readily lead to melting of the metal capsule in localized regions. These problems do not have metallurgical solutions but must be solved in terms of mechanical support and fuel element design; they are however, large factors in determining the time to rupture of the metal capsule.

The problem of establishing limiting conditions for fuel element operation in order to predict maximum burnup is an extremely complex one. Conditions in a gas-cooled reactor are not static. The fuel elements operate at a wide variety of temperatures and flux levels depending on their positions. Fuel elements will be moved periodically in these reactors in order to obtain the required burnup. The ultimate objective in this maneuvering is to reach the reactivity limited burnup before the metal capsule fails.

The probable life of a metal under load at elevated temperatures is determined by the stress and temperature. When these conditions are held constant, it is possible to obtain a fairly reliable prediction of deformation and rupture from laboratory experiments. No such constant conditions exist over the lifetime of an element in a gas-cooled reactor. Initially, elements with stainless steel capsules can operate at 1,800 or 1,900° F without danger of failure, because the only significant stress is that induced by the pressure of the coolant. The metal wall is adequately supported by the pellet. However, as fission gas products are produced and released from the fuel, a gradual buildup of pressure inside the capsule begins. Once the internal pressure exceeds the external pressure, the additional life expectancy of fuel elements operating at these temperature extremes is very short.

Experiments have been conducted at ORNL to simulate the effect of fission gas buildup to pressures in excess of the reactor coolant pressure. Based on the results of these experiments, the effect of pressure on the time to rupture for Type 304 stainless steel at a series of temperatures can be summarized as follows:

Temperature (°F)	Pressure difference (psi) for rupture			
	In 10 hr	In 100 hr	In 1,000 hr	In 10,000 hr
1,300-----	1,000	800	600	450
1,500-----	575	425	250	50
1,800-----	225	100	55	10

The data for 10,000 hours were obtained by extrapolation from the 1,000-hours data.

Fuel Cladding—Beryllium

The attractiveness of beryllium as a nuclear fuel cladding material results from its low neutron-capture cross section. The limitations on the use of beryllium as a fuel-cladding material have not been fully assessed. In the

following sections the current state of knowledge of the capabilities of beryllium as a fuel-cladding material in gas-cooled reactors is discussed.

Fabrication. Development of techniques for fabrication of beryllium into sheets and rods by hot extrusion and rolling has progressed to the point where these products might be called commercially available. Extruded tubing is available from sources in this country and abroad; however, its quality is presently quite poor and its mechanical and physical properties are not completely adequate. The tube manufacturing problems are presently being studied. Development of welding and brazing techniques for joining beryllium has met with some success, and it is believed that these problems can be considered solved.

Mechanical Properties. The strength properties of beryllium appear adequate for its use at surface temperatures up to, perhaps, 1,300° F. The low ductility of beryllium poses a serious design problem,¹⁻³ however, since only limited deformation of beryllium takes place before fracture. This poor ductility has been improved in sheet material by controlled fabrication techniques.

Irradiation Damage. The most serious irradiation-caused damage to beryllium is the helium produced internally as a result of the $(n, 2n)$ and (n, α) reactions. Ells⁴ has shown that the helium can cause swelling of the beryllium upon heating to temperatures around 600° C and above after cold irradiation. Whether this swelling problem persists when the metal is irradiated at high temperature is not known. Beryllium is also subject to the cavitation creep phenomenon that exists for magnesium or Magnox.

Compatibility of Beryllium with Coolant. Beryllium is thought to be compatible with

CO₂ to temperatures slightly in excess of 1,100° F. Huddle⁵ states that beryllium is very oxidation-resistant below 750° C. However, Livey and Williams⁶ report that a "break-away" corrosion rate occurs at 650° C (1,200° F) when beryllium is exposed to dry CO₂. Preliminary tests of beryllium in a static helium atmosphere with graphite at 1,400° F showed severe corrosion in 1,000 hours.⁷ Sheet specimens of beryllium were largely converted to an oxide scale which spalled badly.

Compatibility with Fuels. Beryllium appears to be compatible with uranium metal up to the useful temperature limit of the uranium. The limiting temperature of usefulness for beryllium will be reached before a beryllium-UO₂ compatibility problem occurs.

Fuel Cladding—Graphite

As a result of its low thermal coefficient of expansion, high thermal conductivity, and a low modulus of elasticity, graphite has excellent resistance to rupture from thermal stresses. Its strength increases with increasing temperature, reaching a maximum at about 4,500° F. It is cheap and readily fabricable into a variety of shapes, thus allowing the designer considerable freedom to adjust the surface area for heat transfer.

The reaction of graphite with coolant gases is the most serious limit on its usefulness as a fuel element structural material. The rate of oxidation of graphite by CO₂ is affected by the available surface area, radiation, temperature, pressure, and flow rate. As presently understood, operation with unprotected graphite in CO₂ is limited to a maximum surface temperature in the range 600 to 650° C. In a helium-cooled system, the graphite sur-

¹ The Metal Beryllium, White and Burke (Eds.), *Am. Soc. for Metals*, Cleveland, 1955.

² J. F. W. Bishop, *On the Ductility of Polycrystalline Beryllium*, IGR-TN/S-791.

³ F. M. Yans, *Third Dimensional Ductility and Crack Propagation in Beryllium Sheet*, NMI-1212.

⁴ C. E. Ells, *The Swelling of Beryllium from Neutron Induced Gases*, CR Met-809.

⁵ R. A. U. Huddle, "Oxidation Behavior of Reactor Metals," *Problems in Nuclear Engineering*, Pergamon Press, New York, 1957.

⁶ D. T. Livey and J. Williams, *Some Aspects of the Fabrication Technology of Beryllium and Beryllia*, A/Conf. 15/P/319, 1958.

⁷ *Gas-Cooled Reactor Project Semiannual Progress Report for Period Ending Dec. 13, 1958*, ORNL-2676, p. 101.

face temperature can be much higher without imposing difficult specifications on the purity of the helium. Surface temperatures as high as 2,500° F are thought to be permissible in helium provided the impurity level of CO₂ and water vapor is maintained below approximately 5 ppm.

Another serious problem in the use of graphite as a fuel container material is the release of volatile fission products. Several means are available, in principle, for preventing the uncontrolled release of fission gases and protecting the graphite from corrosion as well.

Graphite with very low permeability is being developed using special impregnation techniques to reduce the open porosity of standard grades of graphite. Grades of graphite have been developed in which the permeability has been reduced from that of AGOT graphite by a factor of 10⁶, and a total reduction of 10⁷ to 10⁸ now appears possible. Such a material, with or without a suitable coating, should be satisfactory as a structural material for fuel elements. Tubular samples of this material in sizes up to 3.5-inch diameter are presently available.

The most promising technique for protecting and sealing graphite structures is the use of impervious oxidation-resistant coatings. Both carbide and disilicide coatings have been tried with limited success. These coatings are generally brittle and subject to mechanical damage. Completely impervious coatings of sili-conized SiC have been applied, however, to graphite bodies of moderate size. This coating is oxidation-resistant, tightly adherent, and refractory and, if properly applied, should protect the graphite up to approximately 2,500° F in most environments. Coated specimens have successfully withstood 100 psi of internal helium pressure at 2,000° F. Preliminary irradiation tests on such specimens have shown them to be resistant to fracture from density changes when irradiated at approximately 150° F. Further, the irradiated

specimens retained their oxidation resistance at 2,000° F.

Dimensional and density changes in graphite due to irradiation are known to occur but are not severe at moderate temperatures. The effect of irradiation at very high temperatures (>1,100° C) on dimensional changes and physical properties must be studied extensively. Probably the most significant effect is that produced by fission fragment damage which may produce density changes of several percent. This type of damage may be minimized by limiting the amount of graphite exposed to fission fragments.

Fuel—Uranium Metal and Alloys

The behavior of uranium metal under irradiation is characterized by reasonably good fission gas retention, but gross swelling occurs if certain temperature or burnup limits are exceeded. The swelling is caused primarily by fission gas expansion within the metal, although the exact mechanism for this gas agglomeration and expansion is not completely understood. The amount of swelling which can be tolerated in reactor fuel is a matter of design, although volume increases greater than 10 percent are likely to be detrimental in any design.

British experience with Calder Hall indicates that pure uranium metal will withstand quite satisfactorily burnups of up to ~1,000 Mwd/MT at ~500° C central temperature. Burnup of ~3,000 Mwd/hr is expected in fuel elements for commercial gas-cooled reactors. In general, failures of the Calder Hall fuel elements have been attributed to manufacturing defects and mechanical properties of the Magnox cladding rather than to swelling of the uranium.

Irradiation experience on the temperature dependence of the growth of uranium and uranium-rich alloys under irradiation has been limited in the United States to capsule experiments. The data available, together with comparable British data, are presented in Figure 2.9. Additions of small amounts of

FIGURE 2.9.—Effect of temperature on the growth of uranium and uranium-rich alloys under irradiation.

molybdenum do not eliminate the swelling problem. Niobium and fission additions likewise have not alleviated the swelling problem.

Gamma-stabilized uranium alloys, such as the U-10 percent molybdenum alloy which has been studied in this country, offer a partial solution to the swelling problem. The uranium-10 percent molybdenum alloy appears to be useful up to temperatures of, perhaps, 1,200° F.

Fuel—Bulk Uranium Dioxide

Extensive studies in the United States, Canada, and the United Kingdom have shown that UO_2 is one of the best fuel materials available. It is refractory and has good compatibility with most structural metals of interest for service at high temperatures. It has the further advantage that it does not swell under irradiation. The nonvolatile fission products are retained within the UO_2 lattice even at burnups in excess of 50,000 Mwd/T. At such

burnups a considerable fraction of the material is fission product oxide, rather than uranium dioxide.

Like every other reactor fuel, UO_2 also has some undesirable features. Its thermal conductivity is very low and decreases with increasing temperature. Thus, at practical power generation rates and heat fluxes, all bulk UO_2 shapes other than very thin sections attain high central temperatures. In addition, the combination of low rupture strength and ductility and high thermal expansion produces extensive thermal fracture of UO_2 in shapes other than thin plates. Thus some type of mechanical support is required for UO_2 elements. Finally, at high temperatures the gaseous fission products, primarily xenon and krypton, are released from the lattice by a diffusion mechanism, and a cladding must be provided to contain the fission products if it is desirable to avoid gross radioactive contamination of the main coolant stream. The ultimate life of a fuel element

containing UO_2 is limited in a gas-cooled reactor by the pressure buildup within the cladding material.

Many variables are known to affect the rate of fission gas release from UO_2 . The rate-controlling step is the diffusion of the gases from the lattice to an externally accessible surface. Hence, factors such as density, particle size, etc., which alter the diffusion path length, affect the release rate. An increased temperature greatly increases the release rate, since the activation energy for the diffusion process is of the order of 60 kcal for xenon and krypton. The diffusion mechanism is apparently altered as the oxygen content of the UO_2 changes; several investigators have reported increased rates of release of xenon or krypton as the oxygen-to-uranium ratio increased. It is still uncertain, however, whether this is due to a diffusion effect or merely to the lower conductivity of the material which has a higher oxygen-to-uranium ratio.

Recent evidence⁸ indicates that the rate of release of fission gases increases when grain growth occurs. While some investigators associate the high release solely with the high temperature required for grain growth, others believe that grain growth *per se* results in enhanced xenon and krypton release, perhaps by a short-circuit diffusion mechanism at grain boundaries. The current status of fission gas release information may be summarized by the statement that fission gas release becomes significant whenever the temperature of the UO_2 becomes sufficiently high to produce significant structural changes in the material. This temperature is said by investigators at Chalk River to be in the neighborhood of 1,500–1,600° C.⁸

Fuel—Uranium Carbide Dispersed in Graphite

Fabrication. There has been a considerable effort to fabricate graphite matrix fuel elements for various reactor systems where the fuel material is very dilute in the graphite

matrix. As much as 250 mg of uranium per cubic centimeter has been incorporated into graphite with little effect on the physical properties of the graphite. The reaction

was used to obtain UC. Fuel elements with high uranium densities have received little attention and, while several possible fabrication processes have been proposed, no process has been employed to the extent that it can be said to be practical. However, a fuel consisting of uranium and thorium carbide dispersed in graphite is being developed for the HTGR reactor. Fuel compacts containing up to 40 weight percent of thorium plus uranium have been experimentally fabricated using a hot pressing technique.

Radiation Behavior. There have been no long-time, high-temperature studies of the radiation behavior of UC-graphite fuel elements. Currently, graphite fuel elements are being tested by Battelle Memorial Institute. Spherical UC-graphite fuel elements, 1½ inches in diameter, containing 4½ g of U 235, have been irradiated to burnups from 1.5–1.9 atom percent, at surface temperatures of approximately 1,300° F. The spheres had dimensional changes of 0.07 to 1.10 percent and weight losses up to 4.48 percent that varied with the process of manufacture. General Atomic is testing the HTGR-type of graphite fuel compacts. Irradiation of these compacts at high temperature and to burnups equivalent to approximately 60 percent of the anticipated HTGR reactor exposure have been carried out and no adverse effects observed.

2.4. COMPONENTS

The current status of the technology of various important components of gas-cooled reactors is summarized here. It is often expedient to discuss these components in the light of British experience with the Magnox reactors. In such cases the basic differences between the natural and partially enriched systems are

⁸ W. B. Cottrell, H. N. Culver and G. Samuels, *Release of Fission Products from UO_2* , ORNL CF-59-6-3.

pointed out specifically if they affect the component design and performance which is extrapolated from that experience.

Pressure Vessels

Once the materials of the system and the fuel element have been specified for given heat-removal conditions, the power level of a gas-cooled reactor, and thereby the cost of the power, is principally dependent on the size of the reactor and the pressure level, both of which are determined by the ability of the industry to fabricate pressure vessels.

Current restrictions on the sizes of reactor pressure vessels and their maximum operating pressures and temperatures are based almost entirely on the thickness of pressure vessel wall which can be field welded, inspected, and stress relieved. Another consideration is the thickness of material which can be formed in economically sized sections in existing industrial equipment. British experience has ranged from the 2-inch wall thickness, which was felt to be the maximum at the time of construction of the Calder Hall reactors, up to the 4-inch plate, which is now being used in the Bradwell station spherical pressure vessel in the region of vessel penetrations. A 3-inch thickness is the current standard in the four gas-cooled civilian power stations under construction in Great Britain. United Kingdom Atomic Energy Authority personnel indicate that future stations will utilize 3½-inch-thick, 65-foot-diameter spherical vessels. There are references in the British literature to the fact that vessels 70 feet in diameter having 4-inch wall thicknesses are currently feasible based on existing technology and that no problems are foreseen in the field erection of 5-inch plate. In this latter case, the problems of fabrication were felt to be controlling. Field-welding and radiographing have presented no insurmountable problems. Stress relieving is done in place by erecting a grid of electrical heaters inside the tank, with external insulation applied as required to obtain the desired temperature.

The steels available for the construction of gas-cooled reactor pressure vessels are of two broad classes: the carbon steels, such as those covered by ASTM specifications A201 and A212, and the recently introduced high-strength heat-treatable steels, such as United States Steel T-1 and United States Navy HY-80. Manufacturers have a strong preference for the use of carbon steels in the immediate future. No undue difficulties are anticipated, and the degree of confidence in the integrity of such pressure vessels is high.

An extensive survey of the technology and the experience of American industry has established 4 inches as a minimum upper limit for field erection of the carbon steels. In this thickness, weldability and formability have been readily achieved in an experiment conducted by ORNL, and the existing technology is directly applicable for the successful inspection and stress relief heat treatment required of the ASME Boiler and Pressure Vessel Code.

As with the use of any steel in a stressed structure, the question of notch ductility is important. As incorporated into a vessel, these steels should be in their most notch-ductile conditions (supplied to ASTM specification A300). The impact resistance of a steel is reduced by fast neutron irradiation. The increase in transition temperature of a pressure vessel steel under irradiation is illustrated in Figure 2.10. In the operation of British reactors, the effect of irradiation on the safe working stress has been carefully considered and in practice, pressure vessels are pressurized only when the temperature is in excess of the transition temperature of the steel.

The first large French plants (G-2 and G-3) utilize prestressed concrete vessels with steel liners. The concrete is stressed by circumferential cables acting in channels in the structure. The container is approximately 46 feet in diameter and 60 feet long, installed horizontally.

FIGURE 2.10.—The effect of fast neutron irradiation on the notch-impact properties of carbon steel.

Reactor Core Support Structure

The current technology of core support structures for gas-cooled power reactors applies almost wholly to stacked-graphite cores which range in size up to 50 feet in diameter and 30 feet in height and contain as much as 2,180 MT of graphite. The base of the structure is of "egg crate" or beam construction to provide a rectangular lattice. Base plates attached to the lattice carry the graphite. Ball races are provided to allow for differential thermal expansion between the graphite and base plates. The entire grid structure may be mounted on an internal skirt, which transmits the load directly to an external skirt, or on brackets, which similarly transmit the load through the pressure vessel wall without introducing shear or bending stresses. Radial restraint is normally accomplished by the use of appropriate spring-loaded circumferential straps. Usually a grid structure rests on the top of the core for top restraint. This top grid serves as a mounting for the burst-slug detection system and as an indexing system for fuel handling and servicing machinery operating through the top of the pressure vessel.

All British gas-cooled power reactors use core-support structures which differ only in detail from the general structure described

above. Operating conditions and allowable stress levels determine the material of construction for any particular application.

Coolant Circulators

The basic technology required for the centrifugal compressors which are used to circulate the coolant in gas-cooled reactors has evolved from the construction of large compressors for the chemical, petroleum, and refrigeration industries. Commercial compressors have been built in capacities up to 200,000 cfm, with heads substantially in excess of the 10,000 to 15,000 foot-pound/pound maximum currently required for service in gas-cooled reactors. The shaft seals used in the circulators for the British reactors are adaptations of the seals designed for hydrogen-cooled generators.

The principal characteristics of the coolant circulators used, or proposed for use, in gas-cooled power reactors are listed in Table 2.3. The power consumed by the compressors typically varies from 2 to 3 percent of the total thermal reactor power. The volume flow ratings of all the compressors shown have roughly the same relationship to the reactor thermal power, regardless of whether helium or CO_2 is used as a coolant. It may be noted

that the head values for helium compressors are substantially higher than those for CO₂ equipment. Correspondingly, the values of mass flow are lower for helium compressors than for CO₂ compressors.

The available data on drive arrangements, operating speeds, and seal configurations are presented in Table 2.4. The coolant system serves as a thermal coupling between the reactor fuel elements and the steam generators. The degree of coupling must be varied to provide a proper match between these two systems. Long-time variations can be accommodated by changing the mean pressure level in the coolant system. More rapid changes in demand must be controlled by changing the rate of coolant flow through the steam generator. This can be accomplished either by changing the operating speed of the compressor, thus changing the rate of coolant flow in the entire system, or by bypassing a portion of the gas flow around the steam generator while operating the compressors at a constant speed. All the compressors described in Table 2.4 make use of speed adjustment to control the rate of coolant flow.

A variety of means may be employed to achieve required changes in compressor speed. As indicated in Table 2.4, steam turbines can be used (either as direct drives or through a turbine-generator set), as well as fluid couplings driven by induction motors, or the relatively complex Ward-Leonard d-c drive-motor system. The use of a steam turbine requires an auxiliary steam system to permit reactor startup.

It is possible to eliminate the need for a shaft seal in its usual form by enclosing the compressor and its drive motor in a common enclosure. This construction has been proposed for the compressors for the General Atomic prototype reactor and for the British advanced reactor (AGR). Enclosed construction presents certain problems, e.g., it is necessary to prevent lubricating oil (for the bearings) from being introduced into the main coolant system. It is thought that the

devices used for this purpose can be much simpler in construction and operation than the shaft seals which must withstand the entire pressure differential between the coolant system and the atmosphere. Enclosed construction tends to minimize the adverse effects of shaft seal failure but does complicate compressor maintenance under some conditions.

Oil or water is used as a buffer fluid in all the proposed shaft seals. Thus purification systems are required to insure that the coolant system does not become contaminated by the buffer fluid. Many of the shaft seals are arranged so that they can be replaced without depressurizing the coolant system. There is no doubt that acceptable compressors can be built for all the proposed gas-cooled power reactors with steam cycles.

Valves

Isolation valves in the primary coolant circuit of a gas-cooled reactor present a major problem because of their size and the low leakage requirements at the seat and stem. The main isolation valves must be capable of holding against full reactor pressure while maintenance work is carried out on the blower or steam generator of an idle coolant loop. In British gas-cooled reactors, these primary coolant circuit isolation valves range in size from 4½ feet at Calder Hall to approximately 6 feet at Hinkley Point.

In the initial reactor installations, motor-operated gate valves were used. Because of leakage, castings are not ideal for high-pressure gas containment. Further, the extra volume required for storage of the gate when the valve is open has caused a trend toward the use of the butterfly valve in later designs. The Hunterston design has retained the gate valve in the 6-foot size.

Several butterfly-valve designs have been developed for sealing against full operating pressure on either side. Installation of a flexible-metal sealing member is required to reduce leak rates. One valve design allows vane rotation followed by axial motion into

TABLE 2.3.—CHARACTERISTICS OF CIRCULATING COMPRESSORS USED OR PROPOSED FOR GAS-COOLED POWER REACTORS

	British reactors (CO ₂ cooled)					French reactors (CO ₂ cooled)		U.S. reactors (helium cooled)	
	Calder Hall	Berkeley	Bradwell	Hunterston	Hinkley Point	EDF-I	G-2 and G-3	EGCR (KE-AC)	General Atomic (prototype)
Thermal reactor power, Mw	144	550	531	535	950			84	90-115
Number of compressors used	4	8	6	8	6			2	2
Conditions at compressor suction:									
Temperature, °F	284	320	356	400	356	284	214(?)	492	633
Pressure, psia	115	125	132	150	185	370		315	290
Volume flow, cfm	47,300	71,000	79,500	62,900	108,000	62,900		31,500	36,400
Mass flow, lb/sec	500	820	877	750	1,670	2,275		59.3	66.8
Pressure rise produced:									
ΔP, psi	5.53	7.34	7.6			22		11.5	10.0
Head, ft-lb/lb	1,260	1,520	1,650			1,550		14,200	14,600
Motor power, hp	1,495	2,675		2,255	4,000 (?)	9,000(?)	3,800(?)	~1,000	~2,500
Type of unit	Overhung, centrifugal	Overhung, single stage, axial flow	Single stage, axial flow	Vertical, overhung, centrifugal	Single stage, axial flow			Multistage axial flow	Overhung, centrifugal

TABLE 2.4.—DRIVE ARRANGEMENTS AND SEAL CONFIGURATIONS FOR COOLANT COMPRESSORS

	British Reactors					French Reactors		United States Reactors	
	Calder Hall	Berkeley	Bradwell	Hunterston	Hinkley Point	EDF-I	G-2 and G-3	EGCR (KE-AC)	General Atomic (prototype)
Drive Equipment:									
A-c induction motor		X	X		X			X	X
D-c motor	X			X					
Steam turbine							X		
Speed control:									
Speed range (%)	10-110	20-100	20-110					20-110	
Fluid coupling		X	?					X	
Turbo-alternator					X				X
Steam turbine only							X		
Ward-Leonard (d-c)	X								
Mercury arc				X					
None (bypass valve)									
Drive Auxiliaries:									
Geared speed increaser								X	
Geared speed reducer							X		
Normal Design Speed (rpm):									
Motor	854		3,000(?)	1,000(?)				1,800	3,475
Compressor	854		3,000	1,000		2,950		6,900	3,475
Primary Shaft Seal:									
None									X
Oil-buffered	X	X	?	?	X				
Water-buffered								X	
Axial-sleeve type								X	
Radial type	X	?	?	?	?				
Number of seals used	1	1	1	1	1			2	
Replaceable at pressure	Yes	Yes	?	?	Yes			Yes	

the seat, thus providing satisfactory throttling action. Another valve under development has a heated seat ring into which the vane rotates. On cooling, a shrink-fitted seal is obtained. The opening operation can be performed in 1 minute.

Other valves judged to be satisfactory are gate valves for fuel-charging nozzles, relief valves for high-pressure discharge, and check valves at blower discharge (used in the French G-2 and G-3 systems). Lower temperature valves installations and valves for applications where leakage is not extremely critical can be adequately specified using existing valve technology.

Piping

The high temperature of the outlet gas from a gas-cooled reactor poses a problem in the selection of a suitable material for the piping between the reactor and the steam generator. Conventional steels must be cooled to meet ASME Code requirements. A more desirable approach is to use an austenitic stainless steel, such as Type 347, for this piping. The actual fabrication of such piping is straightforward and presents no developmental problems.

On the other hand, it is impossible to avoid locations in which the stainless steel must be welded to the carbon steel of the reactor vessel and the steam generator. Metallurgical problems arise in welding carbon steel to stainless steel; in particular, severe thermal stresses result from different coefficients of expansion.

Some technology already exists for the making of such joints in smaller sizes than required for this application. In conventional coal- or oil-fired steam generators, the superheater sections are sometimes made of stainless steel tubing which is joined to the rest of the system, ordinarily low-alloy steel, using very special welding procedures.

Expansion Joints

Bellows expansion joints in sizes up to 6 feet 1 inch in diameter have proved satisfac-

tory in the primary coolant circuits of the British gas-cooled reactors. The bellows are designed to absorb angular movement in the plane of the piping, with axial loading carried by an external trunnion and pin arrangement or by an internal pivoted strut. The bellows are of one-piece construction, fabricated by rolling and seam-welding a plate and then rolling-in the convolutions. Chromemoly steel is the normal material for fabrication for temperatures up to 750° F and pressures up to 165 psi. For less critical applications, circular and rectangular bellows can be fabricated up to sizes of at least 25 feet.

Steam Generator

At least two types of steam generator may be employed in conjunction with gas-cooled reactors. One of these is the conventional steam generator in which water is circulated in the boiler, water and steam are separated in the steam drum, and the steam is passed through a superheater. In the other type, the water is fed directly to the steam generator inlet and flows continuously to the outlet with no vapor separator interposed in its path. The latter arrangement has the advantage of eliminating many tube-to-header joints and leads to a structure which is less complex, lighter in weight, and smaller in size. It has the disadvantage that mineral matter in the feedwater is likely to deposit in the region where the final portion of the evaporation process takes place. This in turn means that exceptionally good feedwater treatment is required.

A traditional disadvantage of the once-through boiler in oil-fired plants has been a tendency for local hot spots to form because of spatial irregularities in the combustion process. These variations make it difficult to obtain the required flow distribution to maintain a satisfactory temperature distribution. However, in a nuclear plant, the temperature distribution throughout the steam generator is much better determined, so the hot-spot problem should not present difficulties.

In recent years there has been a strong trend to increase pressures and temperatures in steam cycles. Experience in the past ten years seems to favor peak steam temperatures of 1,050 to 1,100° F, because metallurgical difficulties, particularly with hot spots in the boiler and superheater, have been substantial at higher temperatures. A majority of the new plants now in design or under construction call for peak steam temperatures around 1,050° F and pressures of about 2,000 psi.

Various steam cycle arrangements have been proposed for the nuclear plants. The dual-pressure cycle as used by the British leads to somewhat better efficiencies and possibly lower capital costs for the low-pressure, low-temperature Magnox reactors. There seems to be little advantage in this cycle for the higher temperature and pressure conditions possible with stainless-steel-clad UO_2 fuel elements. The advantage of the reheat cycle used in most modern steam plants is sufficient to justify the extra complication in cost in plants larger than about 150 MWe.

Control Rod Drive Mechanisms

Current practice for control rod drive mechanisms in gas-cooled reactors involves some form of cable suspension of the rod with a motor-driven winch and clutch assembly. Springs are normally used for cushioning of the rod at the end of its travel, and a shock-absorbing device is installed at the lower end to provide an emergency stop in case of cable breakage. This shock-absorber consists of a shear-pin arrangement and a broaching device for absorbing the rod energy by metal displacement. Instrumentation for detecting cable tension and position is normally provided.

Since many control rods are required in large reactors, the failure of a few units to operate is not normally a serious matter. The current status of the art reflects this fact. No serious efforts are under way to develop a more sophisticated control rod drive for the large gas-cooled reactors. The small, high-

enrichment reactors, such as the prototype of the Army Gas-Cooled Reactor, require fairly elaborate control rod mechanisms, and such units have been designed and fabricated.

Fuel-Handling and Servicing Mechanisms

Fuel-handling and servicing machinery with the following capabilities is currently in use or considered to be technically feasible:

1. move (or be moved by crane) into position indexed with fuel charge nozzle and attach by a pressure-tight connection to the nozzle,
2. contain pressure equivalent to reactor pressure for operation with reactor on stream,
3. remove and store fuel-charge-nozzle shield plug,
4. insert and position fuel-charge chute,
5. charge or remove fuel from as many as 64 fuel channels through one fuel-charge nozzle by using appropriate indexing and position-sensing mechanisms,
6. supply coolant to the channel being operated on in order to prevent overheating due to blockage of normal coolant flow by the mechanism,
7. store irradiated fuel elements in a shielded location for transfer to a pool storage facility,
8. handle and can ruptured fuel elements prior to pool storage,
9. perform certain maintenance functions on fuel channels and control rod mechanisms.

Any mechanism designed for such a variety of operations is certain to be complex mechanically.

The successful operation of the reactor is dependent on the reliability and effectiveness of the fuel charge-discharge machine. The moving parts must operate accurately so that the various positioning operations will be correct even after extensive use and wear of the machine. The positioning must remain cor-

rect to prevent interference during installation or removal of fuel elements. This in turn means that lubrication of the moving parts must be adequate to ensure relative freedom from wear and a low coefficient of friction.

Lubrication of the moving parts of the charge machine poses some exceptionally difficult problems. Not only must the machine operate at a temperature of from 500 to 600° F, but it must also operate in a high radiation field. Aside from temperature effects, the radiation dose level is such that under operating conditions the fast neutron dose would result in the destruction of petroleum-base lubricants in about 10 hours at some critical locations. At high temperatures the deterioration would occur even more rapidly. Some of the synthetic lubricants, such as the silicones, which are better suited to operation at high temperature than petroleum-base lubricants, are as sensitive to radiation as the petroleum-base lubricant and hence are not appropriate for this application. Petroleum-base lubricants ordinarily may not be used at temperatures above ~450° F, and, even at that temperature, their life is limited. If a circulating-oil system is used, substantially greater life can be obtained, but this poses serious additional limitations on the design of the mechanism. The only lubricants which seem to show promise of being suitable to this application are molybdenum disulfide, graphite, and meta-linked polyphenyl ethers (which were specially developed for good radiation resistance and which can be used up to 700° F). The available data indicate that molybdenum disulfide is probably the best material, but there are many practical problems involved in attempting to use any of these lubricants. It seems clear that the design of the fuel charge machinery will have to be predicated on the use of components that are well adapted to the unusual lubrication requirements that will prevail.

To date no charge machine capable of remote handling of the fuel of a gas-cooled reactor has been built and proof-tested satis-

factorily. The British hope to test their first prototype early in 1960.

Burst-Slug Detection Systems

It is important to be able to locate and remove ruptured fuel elements from the system, since gross contamination of the primary coolant system can lead to serious maintenance problems and loss of reactor operating time. A complex system was developed to handle this problem in British gas-cooled reactors. A similar system is used by the French and is under consideration for use in United States gas-cooled reactors.

The method in general use employs a network of tubes to sample the gas stream from each individual fuel channel. Groups of these tubes connect to multiport valves through which a sample can be removed from each channel of the group at intervals and tested for activity. The sampling method consists of filtration of all active particulate matter, followed by electrostatic precipitation of the daughter products of xenon and krypton onto a charged wire. After a suitable sampling interval, the wire is moved into a scintillation counter where the beta activity is measured. This system works adequately for presently planned reactors.

Internal Insulation

The use of internal insulation in gas-cooled systems results in lower operating temperatures of the containment vessel and ducting. The British have employed internal insulation in reactor pressure vessels, but little information on the details of the application are given in the available literature. In the United States there is considerable interest in the subject, particularly in the case of water-moderated, gas-cooled reactors. The Army Gas-Cooled Reactor program has developed a reliable application of internal insulation for the core fuel tubes which operates in contact with water on the outside and 1,200° F gas on the inside. Thermoflex and Refrasil

insulations have been found to suffer no loss of insulating properties under irradiation up to 10^{20} nvt, and adequate means of canning the material have been developed so that loss of binder at high temperatures does not result in dusting into the gas stream. The canning method has been tested and found to be satisfactory under rapid depressurization. Reflective insulation has also been considered and in many respects is superior to thermal-resistance insulation.

2.5. PLANT DESIGN AND CONSTRUCTION

Slightly Enriched Graphite-Moderated Reactor

The current technical status of the slightly enriched, graphite-moderated, gas-cooled reactor is typified by four reactor designs: the British AGR, Oak Ridge National Laboratory's GCR-2,¹ the KE-AC optimum partially enriched GCPR and the EGCR.² The similarities in these systems reflect their being concurrent and overlapping solutions to the problem of establishing the merit and feasibility of a partially enriched gas-cooled reactor. Their differences reflect, in part, differences in design philosophy; more significantly, they reflect the lack of clear-cut limitations on performance in some areas of the relevant technology, and it is, of course, partially to clarify these limitations that the EGCR is being built.

It must be recognized at the outset that the gas-cooled reactor that utilizes slightly enriched, stainless-steel-clad UO_2 fuel does not now have a current technology in the sense that demonstrated solutions to all design and operating problems are in hand. The system is current in the sense that the present technology appears to warrant construction of experimental reactors.

The principal characteristics of the reactors referred to above are given in Tables B.1 through B.4 of Appendix B. The differences

between these designs are noted in the following discussion of the technical status of reactors of this type.

The important considerations that originally led to the use of enriched fuel are the following:

1. Enrichment permits the use of materials (for example, UO_2 and stainless steel), capable of operation at higher temperatures, and higher temperatures lead to higher thermodynamic efficiency in the power-recovery cycle.

2. Enrichment permits undermoderation in the fuel lattice, that is, more fuel per unit volume of moderator and hence higher power density.

3. Enrichment permits greater fuel exposure within the limitations set by the physical properties of the fuel.

The decisive aspect of the slightly enriched gas-cooled reactor is the use of UO_2 as the fuel material rather than uranium metal. Reactivity considerations prevent the use of natural UO_2 in a graphite-moderated reactor. The oxide is preferred to metal because there are severe limitations on the operating temperature of uranium metal. There appears to be no clear-cut limit on the operating temperature of UO_2 , at least up to its melting point of $\sim 2,760^\circ\text{C}$, except for that imposed by the evolution of fission product gases from the fuel body and the resulting buildup of gas pressure inside the fuel capsule.

Results of very extensive experiments at Westinghouse and at Chalk River indicate that the amount of fission product gas evolution is a marked function of the UO_2 temperature, the fuel exposure, the rate of exposure, and the density and composition of the oxide. In pressurized water reactors, it has proved feasible to design UO_2 fuel capsules to withstand the pressure exerted by 100 percent of the stable isotopes of xenon and krypton produced in the capsule, up to exposures in excess of 10,000 Mwd/MT. This is possible both because of the high external coolant pressure exerted on the capsule and because the capsule is operated at a tem-

¹ *The ORNL Gas-Cooler Reactor*, ORNL-2500 (Apr. 1, 1958).

² *Experimental Gas-Cooled Reactor Preliminary Hazards Summary Report*, ORO-196 (May 1959).

perature at which plastic deformation of the wall takes place exceedingly slowly (provided, of course, that the stresses remain within the elastic limit). Thus the design of UO_2 fuel elements for a pressurized water reactor is insensitive to uncertainties in the fission gas evolution rates. The main issue, so far as fission product evolution from the oxide is concerned, is to determine the degree of coolant contamination that would result from failure of a capsule.

In the slightly enriched GCR, however, in order to take full advantage of enrichment and of the materials made usable by the enrichment, it is desirable to operate the fuel capsule surface at as high a temperature as possible: $1,300^\circ\text{F}$, or above for stainless steel, with hot spots above $1,600^\circ\text{F}$. At these temperatures, it is not the tensile strength of the stainless steel that limits performance of the element, but rather the creep properties. The evolution of fission product gases from the oxide thus becomes the decisive design consideration. Depending upon the internal temperature distribution in the fuel, on the volume provided inside the capsule for fission gas accumulation, and possibly also on the rate of fuel exposure, an exposure may be reached at which the gas pressure inside the capsule will exceed the coolant pressure outside, and the capsule will begin to balloon outward, growing away from the oxide pellets. The consequences of this distortion would include the following: (1) a thermal barrier (gas gap) between the oxide and the capsule wall, resulting in ever-increasing oxide temperature and increasing fission product evolution rate; (2) alteration of the cooling-passage geometry (presumably not the limiting effect); and (3) distortion of the capsule to the point of rupture (rupture occurs at rather small deformations, perhaps as little as 2 percent, in the complex biaxial stress situation that prevails here).

The present design of the EGCR fuel element is based upon an arbitrary allowance of 1 percent reduction of the coolant flow area (0.005 inch increase in capsule diameter, 0.7

percent strain) and the assumption of 50 percent release of the stable xenon and krypton from the oxide. With this arbitrary definition of a fuel element failure (1 percent reduction of flow area) it is believed that an average fuel exposure of 10,000 Mwd/MT can be achieved in the EGCR.

Since the creep rate of stainless steel is temperature-dependent, such estimates of fuel life are critically dependent upon the estimated distribution of fuel element surface temperature. With the distribution of surface temperatures estimated by Allis-Chalmers for the EGCR, a "failure" rate of one capsule per 12 days, on the average, is predicted. Since an average exposure of 10,000 Mwd/MT will be accumulated in roughly 1,000 days, and since there will be some 10^4 capsules in the reactor, this implies that approximately 1 percent of the capsules will "fail" before reaching their design exposure. If all 42 capsules in a given fuel channel had to be removed following "failure" of one of them, a much higher effective failure rate would be implied and the average exposure would be greatly reduced. It is calculated that as many as 12 ruptured capsules can be tolerated in the reactor at any one time.

In the light of present information, it does not appear that the exposure of EGCR fuel elements will be severely limited by fission product evolution, and in this sense the fuel element may be said to be feasible. However, there is no basis for guaranteeing an exposure of 10,000 Mwd/MT. It is certainly not possible at the present time to define the limits of performance of fuel elements of this general type. Whether or not central oxide temperatures of $4,000$ or $5,000^\circ\text{F}$ will be feasible, without seriously limiting fuel exposure, cannot now be stated with assurance. The proper optimization of fuel element design, for example, the compromise between heat transfer and neutron economy in selecting the proper degree of subdivision of the fuel, is seriously hampered by lack of a well-established basis for allowing for fission products. It is thus

rather difficult to appraise the current status of reactors of this type.

Another important aspect of the design of rod-cluster fuel elements is the question of mechanical distortion due to temperature asymmetry around the rods of the cluster. Although not peculiar to gas-cooled reactors, this problem is particularly important in this case because of the characteristically large temperature differences between fuel surface and coolant and the large temperature rise of the coolant as it passes through the cooling channel.

The degree of distortion that may occur depends upon the unrestrained length of fuel between supports, the coolant flow conditions, the heat fluxes, and the amount of mixing of flow that takes place between adjacent fuel assemblies in the fuel channel. The bowing of fuel rods may be unstable, that is, bowing of the rods may increase the asymmetry of the temperature distribution, causing still further bowing, until a rod touches its neighbor or the cooling channel wall and then burns out. The problem of thermal stability of rod-cluster fuel elements has been subjected to extensive experimental and analytical studies, both here and abroad, and some initial success has been achieved in obtaining agreement between analytical predictions and experimental results.³

Design of the moderator-reflector assembly involves primarily the following considerations: (1) an adequate support structure; (2) satisfactory nuclear quality of the graphite with respect to its average absorption cross section and density; (3) provision for differential thermal expansion between the support structure and the graphite; (4) provision for dimensional changes of the graphite as a result of fast-neutron irradiation, in order to maintain straight fuel channels; and (5) provision for overall stability of the reactor in the event of earthquakes. By far the most

troublesome of the problems is the graphite shrinkage.

Item 1 is provided for through the use of a massive grid structure inside the reactor pressure vessel, with the loads being carried through the pressure vessel near the perimeter of the grid to bear on external supports resting on a solid foundation. Thus the pressure vessel carries no dead load other than its own weight. For external supports, some designers favor a cylindrical skirt arrangement (GCR-2), others favor vertical column supports spaced on a circle below the vessel (KE-AC GCPR). In any event, there appears to be no feasibility problem involved.

Item 2 is simply a matter of the economics of high-grade graphite vs. the cost of fuel enrichment. Studies at the Oak Ridge National Laboratory and at Allis-Chalmers indicate that, for graphite densities in the range 1.65 to 1.75 g/cm³ and cross sections between 3.7 and 4.5 mb, there is very little effect on the power cost. Higher densities (for example, 1.90 g/cm³) do not appear to be worth their extra cost. (This comment does not apply to the possible use of high-density graphite as a cladding material.)

In many of the British designs, ball bearings or roller bearings are provided between the graphite stack and its support plate to accommodate differential thermal expansion (item 3). In the KE-AC designs, the graphite blocks are supported on several individual steel plates, slightly less than 32 inches square (4 times 4 full lattice cells). These plates, in turn, are supported on the lower base plate by lugs which permit leveling the core support during construction to compensate for mechanical distortion and tolerances. Graphite blocks in the bottom course of the reflector are individually keyed to the support plates; thus differential thermal expansion needs to be accommodated only over a 16-inch square area of a single block.

Items 4 and 5 must be considered together. The essential problem is to maintain the mechanical integrity of the core throughout its

³ See, for example, *Gas-Cooled Reactor Project Semiannual Progress Reports*, ORNL-2676 (Mar. 2, 1959) and ORNL-2767 (Aug. 14, 1959).

life under any foreseeable circumstances and to preserve alignment of all fuel and control rod channels within sufficiently narrow limits to preclude any jamming of fuel elements or control rods. In the United Kingdom designs, this problem is dealt with in two steps. To allow for irradiation-induced dimensional changes in the graphite, which occur preferentially in a direction parallel to the extrusion axis of the graphite, special tiles are provided at regular intervals throughout the stack which bear against each other in the direction perpendicular to their extrusion axes and hold the main moderator blocks slightly separated from each other. In this way, no harmful result is expected to follow from growth of the graphite blocks. The entire core is restrained radially by a series of circumferential bands which force the tiles into essentially solid sheets passing horizontally through the core and binding it into a rigid mass. It must be conceded, however, that the adequacy of this solution to the problem of dimensional changes has not been fully established. The tiles do not eliminate distortion of the core; they merely reduce it by roughly a factor of 2. Radiation-induced dimensional changes in graphite depend upon the temperature at which irradiation takes place. At temperatures above about 300° C, the graphite shrinks and the amount of shrinkage is greater the higher the temperature.

In the Title I design of the EGCR, each vertical stack of graphite blocks forms a single rigid column, which is secured at the bottom by the steel base plate and at the top by a steel egg-crate grid and circumferential girder. The rigid column is achieved by a cruciform key between adjacent blocks, together with a 4-inch graphite stud threaded into each of the mating blocks. The principal difficulties foreseen for this approach are: (1) the threads may be subjected to excessive tensile stresses, both by differential shrinkage of the stud and the surrounding block and as a result of manufacturing tolerances on the threads; and (2) a gradient of the fast-neutron flux across a

single 16-inch block may cause one side to shrink more than the other and cause distortion of the fuel channel.

The selection of the proper coolant for the graphite-moderated, gas-cooled reactor involves consideration of: (1) heat-transfer and heat-transport characteristics, including the effect on thermodynamic performance; (2) chemical compatibility of the coolant with the moderator and with the fuel-cladding material; (3) availability and cost of the coolant; and (4) influence of the coolant on the design of the system. These matters have been thoroughly discussed in several reports,^{1,4,5} and the principal conclusions are summarized here.

In terms of pumping power required to transport a given quantity of heat, subject to specific limitations of fuel surface temperature and coolant inlet and outlet temperatures, helium and CO₂ are roughly comparable; the advantage one way or the other depends upon such accessory conditions as maintaining a constant void fraction and on the specific power of the fuel. In general, helium requires somewhat less pumping power than CO₂; hydrogen requires about one-fifth as much pumping power as helium; nitrogen, air and CO require roughly four times as much, and argon some 20 times as much. Barring hydrogen on the grounds of chemical activity and explosive hazard, helium and CO₂ are the obvious choices.

Helium is chemically inert and compatible with all fuel and moderator materials throughout the temperature range of interest. Gaseous impurities may be present, however, the concentration of which must be kept below specified limits to prevent corrosive attack on the graphite or the fuel cladding. These include CO, CO₂, H₂O, and methane. Some are evolved from the graphite during early stages of reactor operation; others, such as water, might enter through small leaks in the steam

⁴G. Samuels, *Comparison of Gases for Use as the Coolant in a Gas-Cooled Reactor*, ORNL CF-58-4-108 (Apr. 28, 1958).

⁵*Gas-Cooled Reactor Coolant Choice*, ORNL-2699 (May 21, 1959).

generator, or might be present following an operation in which the system had been open to the atmosphere. Present limitations on concentrations of impurities are somewhat arbitrary, their being no clear-cut criteria for determining allowable concentrations. On the basis of not more than 1 percent burnout of the graphite in 20 years and not more than 500-pound mass transfer of graphite, the following limits were set for the coolant for the EGCR: 1.8 percent CO_2 , 0.1 percent H_2 , 0.1 percent N_2 , etc.

Carbon dioxide, through a series of complex reactions with graphite that depend both upon the temperature and on the presence of radiation, removes carbon from the graphite stack and may deposit it in cooled portions of the system, such as the steam generators. Up to about 600° C, the reaction rate varies slowly with temperature, while above 600 to 650° C, it is a marked function of the temperature. It is not possible at present to guarantee satisfactory long-term performance of a CO_2 -cooled graphite reactor with a maximum graphite temperature in excess of 600 to 650° C, and it was entirely for this reason that the decision was made originally to adopt helium as the coolant for the GCR-2 and subsequently for the EGCR. Further information with respect to such possibilities as inhibition of the reaction rate by addition of CO or protection of the graphite by coatings, such as siliconized silicon-carbide, may, however, result in an increase in the allowable graphite temperature in a CO_2 -cooled reactor.

If the helium-recovery program of the Bureau of Mines is put into effect, some 75 billion cubic feet of helium will have been made available by about 1975, and much of it will remain in storage at that time. (Current annual consumption is about 0.4 billion cubic feet per year.) Given the helium inventory and leakage losses assumed for GCR-2 (1 percent per day), the helium estimated to be available for reactor use would support an electrical capacity of 50 million kw for 20 years. It is more likely, however, that the leakage

losses would be brought well below the assumed 1 percent per day, and the supportable capacity would be correspondingly greater.

The loss rate of 1 percent per day for GCR-2 adds 0.05 mill/kwhr to the power cost, at a price of \$22.70/1000 scf. At the higher cost of \$44/1000 scf that is estimated to be necessary to pay for the recovery program, the added power cost would be 0.1 mill/kwhr.

It does not appear that the cost or availability of helium should seriously limit its use as a reactor coolant in the United States. However, there is no evidence of even a remotely comparable supply of helium abroad. While it can be shown that shipment of helium from the United States to Europe would add at most 0.2–0.3 mill/kwhr to the power cost, the incentive to develop satisfactory means for using CO_2 at higher temperatures in European graphite-moderated reactors is very strong.

D₂O-Moderated Gas-Cooled Reactor

In the gas-cooled reactor the basic neutron economy is determined by neutron physics characteristics of the moderator and whatever structural materials must be put into the moderator to contain the coolant and the fuel. The excellent neutron physics characteristics of D_2O make it attractive as a moderator for gas-cooled reactors. Design studies have indicated that good neutron economy can be obtained with D_2O -moderated reactors of practical design and that natural uranium can probably be used as fuel without causing an unreasonable reduction of thermal performance.

A development program on the heavy-water-moderated, gas-cooled reactor has been sponsored in the United States by the East Central and Florida West Coast Nuclear Groups of utility companies. The design studies made for these groups by the General Nuclear Engineering Corporation have lead to a proposal to build a 50 MWE prototype of this reactor.

The physical characteristics of the moderator determine the requirements of this reactor and thus define engineering problems in

the design of this reactor which, in turn, put further restrictions on the neutron physics characteristics of the system. It is instructive to consider briefly the differences between heavy water and graphite-moderated reactors as a result of differences in physical properties of the two materials, compared with the differences resulting from nuclear properties which give heavy water a basic advantage with respect to reactivity, neutron economy, and power density. The fundamental difference is that graphite is a high-temperature moderator, and heavy water must be kept at a low temperature. Since D_2O is a liquid, the provision of adequate cooling at any desired practical temperature is a relatively simple and straightforward engineering task. A more difficult engineering problem is that of insulating the gas channels from the D_2O in order to minimize the flow of heat from the gas channels into the moderator. The avoidance of such heat flow is necessary because the heat which leaks into the moderator cannot be utilized efficiently for the generation of mechanical energy at the low moderator temperature. To the extent that leakage heat is not utilized or is utilized at low efficiency, the overall efficiency of the nuclear power plant will suffer.

The reference design for the D_2O -moderated, gas-cooled reactor consists of a tank of heavy water penetrated by pressure tubes which contain the fuel elements through which the gas coolant flows. The tubes are insulated from the surrounding moderator. The gas coolant is at high pressure in order to attain reasonable power densities. The pressurized tube concept, which is so natural with heavy water, does not seem readily applicable to high temperature, graphite-moderated reactors. One of the outstanding virtues of the pressure-tube system is that it has no intrinsic size limit. In the designs which have been studied in the United States to date, CO_2 is considered as the coolant. It should be pointed out in passing that steam is also a possible candidate as the coolant for a reactor of this type.

A reasonable value for the specific power in

heavy-water systems of this type is about 15 Mw(th)/MT, which is somewhat higher than current graphite-moderated designs offer [7-10 Mw(th)/MT]. There are several possible fuel elements for use in reactors of this type, depending on the specific application of interest. Three categories that are under study at the present time are discussed below. (1) Uranium-metal fuel clad with magnesium or aluminum: The heavy water-moderated reactor has a comfortable margin of reactivity when fueled with uranium-metal elements having low cross section metallic claddings. This extra reactivity can be used to attain high power density through flux flattening, undermoderating the reactor, or shortening the reactor to improve conditions for gas heat transport; or it can be used to give long fuel life if natural uranium elements can be developed which will withstand the radiation damage. The limitation of these fuel elements is the maximum gas temperature, which is about 750° F, and which is considerably lower than is desirable for a really efficient power-recovery cycle. (2) Beryllium-clad uranium oxide elements: This element has not yet been developed on a production scale, although it is the reference design for the United Kingdom's AGR. It is probable that the incentive is considerably greater for development of beryllium cladding for heavy water reactor fuel elements than for graphite-moderated reactor fuel elements. The indications are that the heavy water reactor would achieve quite attractive performance with beryllium-clad fuel elements and, if plutonium recycle were used with such elements, the equivalent enrichments attainable by the addition of the recycled plutonium to natural uranium would be sufficiently high to approach optimum thermal performance. (3) Stainless steel-clad uranium oxide elements: Calculations indicate that there is little probability of operation with natural uranium in a pressure-tube heavy water reactor fueled with stainless steel-clad elements. However, natural uranium feed with plutonium recycle would probably

be feasible in such a system. The stainless steel-clad elements being developed for the EGCR are directly applicable, with some modification in dimensions, to the heavy water-moderated reactor.

It is planned that the initial Florida reactor core will utilize stainless steel-clad UO_2 elements, and future cores will use beryllium-clad UO_2 elements. The reference fuel element design is a bundle of 19 cylindrical capsules in a fuel cluster. The cluster is surrounded by an insulating liner which reduces the inner temperature of the Zircaloy pressure-tube wall. The detailed performance parameters of the prototype reactor and a typical full-scale reactor are given in Tables B.6 and B.7 of Appendix B.

Highly Enriched Graphite-Moderated Reactors

Development programs exist, both here and abroad, on high temperature, graphite-moderated, gas-cooled reactors using highly enriched fuel. The technology of this reactor concept has been sufficiently advanced under the British program to provide a basis for the construction of a small high-temperature reactor experiment. The HTGR project in the United States program is directed toward the construction of a 40 MWE prototype plant, and requires a parallel extension of reactor technology.

There are many distinctive features in the HTGR design proposed by General Atomic; only the highlights will be discussed here. The core is made up of cylindrical elements which incorporate the fuel, the graphite moderator, and the top and bottom graphite reflector pieces. A portion of the graphite reflector provides an elastic type of restraint for the core. It is pivoted at the bottom and moves inward under the load imposed by differential helium coolant pressure. The primary coolant gas is used to cool the pressure vessel, the structures inside the vessel, and the graphite reflector before passing through the core region. Thus, the need for a secondary coolant system is eliminated. Another im-

portant feature is the use of a concentric pipe system for circulating the helium coolant to the steam generators and back to the reactor. Finally, the gaseous fission products are purged from the graphite-clad elements by helium coolant gas which is bled through each element, two series-connected fission gas traps, and a leak-detection system.

The fuel elements for the first core loading may be stainless steel clad. However, a graphite-clad core may be available for this loading, depending upon the rate of development of a low-permeability graphite. The fuel bearing part of these elements is a mixture of graphite with uranium and thorium in the form of carbides. This is mounted inside a graphite cylinder that has solid end caps, made of graphite, which form the top and bottom reflector pieces. The entire unit is encapsulated in a stainless steel envelope.

Graphite-clad elements will be used in subsequent core loadings. These elements will consist of an inner core, which is a dispersion of fissile and fertile materials in high-density low-permeability graphite, a low-permeability graphite can, and a graphite outer sleeve. The sleeve is joined to the solid-graphite reflector sections at the ends.

Since the length of an element is equal to the height of the core, the volume fraction of metal in the core region when a metal-clad core is used is minimized. The high conductivity of the graphite tends to reduce any temperature asymmetries around the periphery of an element. However, the effect of the temperature asymmetries upon the structural integrity of the metal cladding is important, and should the external pressure not be sufficient to force the cladding onto the graphite sleeve at operating temperatures, the advantages of the high thermal conductivity of the graphite are partially lost if the graphite does not contact the cladding at all points.

The stainless steel capsule proposed for the fuel element consists of a stainless steel sleeve 0.010 inch thick, 4.15 inches in diameter, and 140 inches long. There is a short region 0.223

inch thick two feet from the upper end which serves as a spacer for the upper end of the fuel element. Relatively massive stainless steel end caps are threaded to the $\frac{1}{2}$ -inch-thick, 4.3-inch-diameter graphite tube which constitutes the basic structure of the fuel element. The nature of the threaded joint is such that the end caps are rigidly attached to the graphite tube. The thin stainless steel tube is designed to fit closely over the outer surface of the graphite tube, with a diametral clearance of only a few thousandths of an inch, and is welded at either end to end caps. In view of the fact that the graphite is much stronger than the stainless steel, the difference in the thermal expansion between the two materials has to be accommodated through elastic and plastic deformation of the stainless steel tube. The pressure outside the stainless steel cladding under operating conditions will be approximately 300 psi, while the internal pressure will be approximately 60 psi. The resulting radial pressure differential will cause the thin stainless steel tube to be maintained in contact against the graphite. Thermal cycling tests of these metal-clad graphite elements have shown that the cladding integrity can be maintained for a large number of cycles and without wrinkles or tears occurring in the metal cladding.

The problem of differential thermal expansion between the capsule and the graphite tube has been recognized, and a half-round spiral groove $\frac{1}{16}$ -inch in radius has been provided in the exterior surface of the graphite tube. A small wrinkle will be formed in the stainless steel tube at room temperature by imposing an appropriate external pressure after fabrication. This helical groove, which will have a pitch of about $3\frac{1}{4}$ inches, should accommodate both the axial and the circumferential differential thermal expansion between the capsule and the graphite. The combined buckling and thermal cycling characteristics of the cladding are not known under service conditions.

Graphite-clad elements are more attractive than stainless steel-clad elements from a per-

formance standpoint. The gaseous fission products which escape through the cladding around the fuel-bearing material are purged from the elements by bleeding the helium coolant through the annular gap between the cladding and the outer graphite cylinder. This gas then flows through a fission gas trapping system where the fission products are removed from the helium. The control of the fission gases is the crucial issue in determining the feasibility of this concept.⁶

The annular reflector region that surrounds the core is 2 feet thick and consists of two segmented concentric rings. The segments in the outer ring are rigidly attached to the core and reflector support plate, while the inner portion is hinged to this plate. Gas seals are inserted between the segments of the inner ring, and another seal spans the gap between the upper end of the ring and the shroud mounted inside the pressure vessel. The shroud, in combination with the inner reflector region, forms an envelope which prevents the incoming coolant gas from bypassing all or a part of the core region. The difference in gas pressure across the graphite ring causes its upper portion to move inward and provide an elastic type of constraining force for the elements within the core. This arrangement offers a unique and simple solution to the core-restraint problem provided satisfactory gas seals are developed.

The pressure vessel is a circular cylinder with elliptical heads. It is 14 feet in diameter and $34\frac{1}{2}$ feet high, with a 2-inch wall thickness. The material is ASTM type SA-212, grade B, carbon steel plate. The vessel is supported by eight lugs evenly spaced around the circumference at a position immediately below the coolant pipe nozzles.

The core and reflector are supported by a 3-inch grid plate bolted to an underlying beam structure similar to the Calder Hall diagrid. The beam structure is supported by a ring which is an integral part of the bottom pressure-vessel head. All the structure and

⁶ For a detailed review of this problem see OBNL-2900, Part 3.

the elements inside the vessel are thus supported by the bottom head.

The thermal shield, composed of three concentric shells, serves as a liner for the pressure vessel. Part of the coolant gas flows through the three passages formed by the thermal shield and the pressure vessel and part flows in the annulus between the upper plenum shroud and the thermal shield. Various seals are used to keep the coolant gas separated from the hot gas in and above the reactor core. The gas in the annulus between the pressure vessel and the outer thermal shield shell cools the bottom head of the pressure vessel in the plenum at the bottom of the core. The pressure vessel is of conventional design and of a material commonly used in industry and, therefore, should present no unusual fabrication problems.

The coolant piping consists of a concentric piping arrangement with a 30-inch outside diameter, schedule-30, carbon steel outer pipe and a 20-inch-inside diameter, Type 304 stainless steel inner pipe with a $\frac{1}{4}$ -inch wall. The piping configuration is such that it is inherently flexible, and the need for expansion joints is thereby eliminated. A liner is used inside the inner pipe so that it remains at a low temperature. The concentric arrangement is attractive in that it results in a low pressure differential across the inner pipe and keeps the thermal expansion low, since the inner pipe is cooled by the gas in the annulus. This results in low stresses caused by thermal expansion and small reactions at the pressure vessel and steam-generator attachment points. Gate valves are specified for closure of the pipes for equipment isolation.

The principal characteristics of United States HTGR designs are given in Tables B.8 and B.9 of Appendix B.

Any radioactive material circulating in the primary piping system represents an obvious hazard. While there are at least ten potential sources of such contamination, only three are significant: (1) activation of the coolant, (2) recoils from the fuel capsule wall, and

(3) fuel capsule leaks. The maximum total activity expected in the gas system of a 225 MWE system cooled with helium from each of these sources is 67, 18, and 7 curies, respectively, and the activity is diffused over a large volume; moreover, the helium activity is a low-energy beta. In the GCR design, the dose outside the heat exchanger during operation should be less than 3 mr/hr.

The maximum credible accident (mca) that can occur with this type of reactor is a rupture in the gas system pressure envelope. Such a rupture would lead to a crack and a "blow-down." The principal hazard associated with such a rupture would be contamination of the atmosphere downwind of the plant by the activity carried in the gas system. Calculations show that even this accident does not lead to an unduly hazardous situation unless many of the fuel capsules are ruptured or melt. Detailed considerations of the mca have led, in the case of EGCR, to the conclusion that less than 1 percent of the fuel capsules would fail in such an accident. In view of the improbability of a nuclear excursion and the previously mentioned provisions for the removal of afterheat, there appears to be no mechanism available to raise the temperature of a significant fraction of the fuel capsules into the highly critical range.

2.6. HAZARDS

Detailed descriptions and hazards analyses have been presented for gas-cooled reactors in general and for at least one specific gas-cooled reactor, i.e., the EGCR.^{1,2} One of the outstanding features of gas-cooled reactors is that there seems to be a reasonable possibility that even the high-performance reactors will not require containment because of their intrinsic safety. It goes without saying that in the very long run the ultimate safety of a given

¹ *The ORNL Gas-Cooled Reactor—Materials and Hazards*, ORNL-2505 (June 19, 1958).

² *Experimental Gas-Cooled Reactor Preliminary Hazards Summary Report*, ORO-196 (May 1959).

reactor type may be one of the most important issues in the economic and administrative judgments and choices which must eventually be made.

The most important accident to consider in a gas-cooled reactor results from loss of coolant and/or coolant flow. Therefore, it is necessary that the system be designed so that the afterheat can be removed in spite of a complete electrical power failure. This heat removal must be accomplished by radiation and thermal convection, assisted by the heat capacity of the moderator. If the afterheat cannot be removed satisfactorily, containment will be necessary, at least in high-performance systems characterized by an average specific power $>10 \text{ Mw(th)}/\text{MT}$ of uranium. In solid-moderated systems, thermal radiation and emission of gamma energy to the moderator will result in the transmission of the bulk of the afterheat to the moderator, which, because of its large heat storage capacity, rises only slowly in temperature. Calculations have been performed both for the idealized case, in which all the heat is dissipated by radiation with no gas circulation, and for the case in which there is no thermal radiation and the heat from the reactor core is removed by thermal convection in the gas system. The analyses indicate that it is possible to so design a system that the loss of coolant or coolant flow does not constitute a serious hazard insofar as heat removal is concerned.

In order to make these considerations quantitative, the amounts of energy that might be

released from various components of a typical system (GCR-2) are listed below:

	<i>Energy available for release (Btu)</i>
Fission product heat during first 2 hours after shutdown -----	5.8×10^7
Equilibrium irradiation energy in graphite which would be released by annealing at $1,000^\circ \text{C}$ -----	1.8×10^6
Total irradiation energy in graphite after 20 years of exposure -----	See footnote 3
Heat released in 2 hours of oxidation of the graphite -----	$<2.2 \times 10^4$
Total energy in helium expanded and cooled to atmospheric conditions -----	8.0×10^6
Internal energy available in helium ⁴ -----	4.86×10^6
Total energy in water and steam expanded and cooled to atmospheric condition -----	1.1×10^7
Energy available from "flashing" water in steam generators ⁵ -----	7.25×10^6
Thermal energy in graphite above 75°F -----	5.1×10^6
Thermal energy in steam generator metal above 75°F -----	1.6×10^6
Thermal energy in pressure vessel metal above 75°F -----	7.6×10^6

³ This energy is of the same order of magnitude as most of the energies listed, but there is no apparent mechanism to release it.

⁴ Isentropic expansion to standard temperature and pressure.

⁵ Isentropic expansion to atmospheric pressure of steam evolved (steam assumed to remain as a vapor).

Rather large quantities of energy are involved, but, because of the large thermal capacity of the system, none of these sources appears capable of causing a serious accident for which additional containment need be provided.

BIBLIOGRAPHY

This bibliography contains 464 selected, unclassified gas-cooled reactor references found in Nuclear Science Abstracts. An effort has been made to include only that material which has been published since the first International Conference on the Peaceful Uses of Atomic Energy.

The references are grouped by country and are separated into reports, articles published in technical journals, patents, and books. The United States reports are further divided by issuing installations.

The reports are arranged alpha-numerically according to report number. Published articles appear alphabetically by the journal title. Patents are listed numerically and books are arranged alphabetically by the first author. Reports with the number prefix A/CONF. 15/P/ refer to papers from the 1958 International Conference on the Peaceful Uses of Atomic Energy.

Author and report number indexes follow the list of references.

UNITED STATES

Reports

*ACF Industries, Inc., Nuclear Products—Erco Div.**

1. "Prescoping Optimization for Selection of Enriched Uranium Gas-Cooled Power Reactor Design," W. FARMER, E. HILL, P. LANG, K. TRICKETT, and S. MATOVICH, Feb. 14, 1958, AECU-3993.
2. "Physics of the Prototype Enriched GCPR," P. M. WOOD, p. 49-53 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

*See also Kaiser Engineers Div., Henry J. Kaiser Co., and Nuclear Products—Erco Div., ACF Industries.

3. "Design Basis for the PEGCPR Fuel Assembly," R. S. MILLER, S. B. BURWELL, J. K. DAVIDSON, and W. J. O'LEARY, p. 54-62 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
4. "Reactor Core Structure of the PEGCPR," F. R. RULIEN, p. 63-9 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
5. "Reactor Hazards," P. I. PERRY, p. 70-7 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
6. "Service Machine and Control Rod Drives," G. W. FIESER, p. 98-109 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Aerojet-General Nucleonics

7. "Safety Aspects of Heterogeneous Compact Gas-Cooled Reactors," D. R. SAWLE and F. J. THOMAS, A/CONF.15/P/1884.
8. "Army Gas-Cooled Reactor Systems Program, Heat Transfer in an Annulus with Asymmetric Heating, Final Report," A. V. LISIN, W. V. MACKIEWICZ, and W. C. REYNOLDS, Feb. 28, 1959, IDO-28534.
9. "Army Gas-Cooled Reactor Systems Program, Monthly Progress Report [for] May 1959," June 29, 1959, IDO-28541.

*Alco Products, Inc.**

American Radiator and Standard Sanitary Corp., Atomic Energy Div.

10. "Comparison of Calder Hall and PWR Reactor Types," A. PUISHES, D. P. HERRON, D. R. MASH, and J. W. WEBSTER, March 1, 1957, AECU-3398.

*See Sanderson and Porter, New York and Alco Products, Inc.

11. "Plutonium Recycle in the Calder Hall Type Reactor," L. J. BARBIERI, J. W. WEBSTER, and K. T. CHOW, Jan. 1, 1958, ASAE-S-8.

Argonne National Laboratory

12. "A Design Study of Two Helium-Cooled Reactors," ROBERT H. ARMSTRONG, ADOLF LOVOFF, and LYMAN J. TEMPLIN, p. 445-61 [of] Nuclear Science and Technology, Vol. 1A, Issue 3, December 1955, TID-2508.

Atomic Power Development Associates, Inc.

13. "Leakage from Inert Gas System, Technical Memorandum No. 14," W. W. KENDALL and W. L. CHASE, April 4, 1958, AECU-4097.

Atomics International Div., North American Aviation, Inc.

14. "Some Corrosion Considerations of High Temperature Gas-Cooled Reactors," S. J. YOSIM, R. L. MCKISSON, A. M. SAUL, and D. E. MCKENZIE, A/CONF.15/P/1077.
15. "An Evaluation of the Calder Hall Type of Nuclear Power Plant," W. F. BANKS, G. A. SCHNEIDER, W. T. MORGAN, and E. B. ASH, Jan. 18, 1957, NAA-SR-1833.
16. "A Further Evaluation of the Calder Hall Type of Nuclear Power Plant," W. F. BANKS, June 28, 1957, NAA-SR-1955.

Battelle Memorial Institute

17. "Study of a Liquid-Metal Neutron Absorber for the Control of a Gas-Cooled Power Reactor," BENJAMIN B. GORDON, J. JAMES STONE, and HAROLD M. EPSTEIN, July 30, 1956, BMI-1119.
18. "Hazards Summary Report for the GCRE Critical-Assembly Experiments," WILLIAM S. HOGAN, DAVID A. DINGEE, FRANCIS J. JANKOWSKI, and JOEL W. CHASTAIN, July 22, 1958, BMI-1240 (Rev.).
19. "An In-Pile Gas-Cooled Loop Installed at the Battelle Research Reactor," GEORGE T. BODNAR, WALLACE J. STEVENS, GERALD A. FRANCIS, SHERWOOD L. FAWCETT, and ALBERT M. ROCKWOOD, Sept. 16, 1958, BMI-1290.
20. "Properties of Uranium Dioxide-Stainless Steel Dispersion Fuel Plates," STAN J. PAPROCKI, DONALD L. KELLER, and JOSEPH M. FACKELMANN, April 28, 1959, BMI-1339.
21. "Experiences with the Design, Construction, and Operation of In-Pile Recirculating Gas-Cooled Loops for Fuel-Subassembly Tests," S. L. FAWCETT and G. A. FRANCIS, p. 217-40

[of] Test Reactors Meeting for Industry, Idaho Falls, Idaho, May 13-15, 1959, Part II. Utilization of Test Reactors, IDO-16520.

Brookhaven National Laboratory

22. "Summary Hazards Report on Enriched Fuel Element Loading for the BNL Research Reactor," N. CORNGOLD, May 24, 1956, BNL-3146.
23. "A Gas-Cooled Liquid-Metal Reactor," HAROLD L. FALKENBERRY, C. J. RASEMAN, WILLIAM A. ROBBA, THOMAS V. SHEEHAN, and LINCOLN D. STROUGHTON, p. 125-41 [of] Nuclear Science and Technology (Extracts from Nuclear Science and Technology, Vol. 1, Issues 1 to 3, February-June 1955), TID-2506 (Del.).

Bureau of Mines, Morgantown, W. Va.

24. "High-Temperature Systems for Nuclear Process Heat," JAMES P. MCGEE, p. 305-24 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Civilian Reactor Fuel Element Review Group, AEC

25. "Report on Civilian Reactor Fuel Elements," June 1959, TID-8505.

Commonwealth Edison Co.

26. Commonwealth Edison Company—Public Service Company Report on Power Generation Using Nuclear Energy," LESTER A. EVERS and WILLIAM M. KIEFER, p. 139-80 [of] Nuclear Science and Technology (Extracts from Reactor Science and Technology, Vol. 2, Issues 1 to 4, April-December 1952), TID-2503 (Del.).

du Pont de Nemours (E. I.) & Co., Atomic Energy Div.

27. "Power Reactor Studies, Quarterly Progress Report [for] August, September, and October 1957," L. ISAKOFF (comp.), Feb. 1958, DP-265.
28. "A Preliminary Evaluation of Gas Cooling of Power Reactor Moderated by Heavy Water," R. C. HOLMES, L. BARNATH, and E. W. HONES, Aug. 1958, DP-307.

Ford Instrument Co.

29. "15 MW Gas-cooled Closed-Cycle Reactor Power System Study, Volumes 1 and 2, Final Report," April 1, 1957, FICO-101.

30. "Control Rod Drive Development for Closed Cycle Gas-Cooled Reactor Program," March 1958, FICO-106.
31. "Fission Product Diffusion Investigation for Closed Cycle Gas-Cooled Reactor Program," March 1958, FICO-107.
32. "Kinetics Study for Closed Cycle Gas-Cooled Reactor Program," March 1958, FICO-108.
33. "Control Material Burn-Up Study for Closed Cycle Gas-Cooled Reactor Program," March 1958, FICO-109.

General Atomic Div., General Dynamics Corp.

34. "Gas-Cooled Marine Nuclear Power Plant, Interim Report," W. I. THOMPSON, F. ADLER, K. CASE, W. H. BOND, E. C. CREUTZ, E. A. HALE, A. C. MCCLURE, D. NICOLL, L. W. NORDHEIM, D. F. PUTNAM, N. ROSTOKER, G. T. SCHNURER, M. T. SIMNAD, H. P. SLEEPER, and G. B. WEBB, Jan. 25, 1957, GA-56(Pt. I).
35. "Supplementary Studies on Gas-Cooled Reactor for Marine Propulsion," W. I. THOMPSON, E. BJORNERUD, L. H. CHEN, G. HINMAN, A. C. MCCLURE, D. NICOLL, H. POPPENDIEK, N. ROSTOKER, G. SCHNURER, and G. WEBB, Aug. 14, 1957, GA-179(Pt. I).
36. "Evaluation of Coolants and Moderators for the Maritime Gas-Cooled Reactor," Dec. 10, 1958, GA-570.
37. "Gas-Cooled Reactor Plants for Marine Propulsion," T. H. PIGFORD, p. 156-69 [of] Proceedings of the 1958 Nuclear Merchant Ship Symposium, Held in Washington, D.C., August 21, 1958, TID-7563.
38. "The Marine Gas-Cooled Reactor Program," S. A. BERNSEN, J. C. BOKROS, R. C. HOWARD, T. H. PIGFORD, and W. P. WALLACE, p. 243-304 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

General Electric Co., Hanford Atomic Products Operation

39. "A Parametric Study of the Gas-Cooled Reactor Concept," March 1, 1958, HW-54727(Rev.).
40. "An Addendum to a Parametric Study of the Gas-Cooled Reactor Concept," March 1, 1958, HW-54727(Add.)(Rev.).
41. "A Re-Optimization of Several Process Parameters for the Gas-Cooled Reactor Concept," R. C. WALKER, G. E. WADE, E. R. RUDOCK, W. A. MASSENA, and E. R. ASTLEY, June 30, 1958, HW-56009.
42. "Gas-Cooled Power Reactor Coolant Choice," H. W. HEACOCK and R. E. NIGHTINGALE, June 12, 1958, HW-56362(Rev.).

43. "Graphite Studies for the Gas-Cooled Reactor Program," D. E. BAKER, p. 185-92 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
44. "Parametric Studies of the Gas-Cooled Reactor Concept," HAROLD W. HEACOCK, p. 234-42 [of] Information Meeting on Gas-Cooled Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

General Nuclear Engineering Corp.

45. "Design Concept for a Gas-Cooled, Pressure Tube D₂O Moderated Reactor," G. M. ROY and M. F. VALERINO, p. 216-33 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Kaiser Engineers Div., Henry J. Kaiser Co.

46. "Introduction to the KE-ACF Gas-Cooled Nuclear Power Plant," J. S. RITCHIE, p. 1-12 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
47. "Reactor Pressure Vessel and Charge Machine," H. D. YOUNG, p. 78-97 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
48. "Non-Nuclear Mechanical Components," J. P. ERNST, p. 110-18 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Kaiser Engineers Div., Henry J. Kaiser Co., and Nuclear Products Erco Div., ACF Industries, Inc.

49. "Gas-Cooled Power Reactor, Preliminary Design, 55,000 KW Prototype Natural Uranium Nuclear Power Plant," April 1, 1958, IDO-2021(Rev.).
50. "Appendix to Gas-Cooled Power Reactor, Preliminary Design, 55,000 KW Prototype Natural Uranium Nuclear Power Plant," April 1, 1958, IDO-2021(Rev. 1, App.).
51. "Gas-Cooled Power Reactor Feasibility Study, Optimum Natural Uranium Nuclear Power Plant," April 1, 1958, IDO-2022(Rev. 1).
52. "Gas-Cooled Power Reactor, Feasibility Study, 44,000 KW Prototype Partially Enriched Uranium Nuclear Power Plant," April 1, 1958, IDO-2023(Rev. 1).

53. "Gas-Cooled Power Reactor, Feasibility Study, Optimum Partially Enriched Uranium Nuclear Power Plant," April 1, 1958, IDO-2024 (Rev. 1).
54. "Comparison of Optimum Natural Uranium Nuclear Power Plant, Gas-Cooled Power Reactors," Aug. 1958, IDO-24026.
55. "Gas-Cooled Power Reactor Preliminary Design, 30,000 KW Prototype Partially Enriched Uranium Nuclear Power Plant," March 1959, IDO-24027 and App.
56. "Gas-Cooled Power Reactor Site Feasibility, Oak Ridge, Tenn., 30,000 KW Prototype Partially Enriched Uranium Nuclear Power Plant," March 1959, IDO-24028.
57. "Gas-Cooled Power Reactor Site Feasibility, Savannah River Project, Aiken, S.C., 30,000 KW Prototype Partially Enriched Uranium Nuclear Power Plant," March 1959, IDO-24029.
58. "The Method for the Optimization of a Helium-Cooled Graphite Moderated Reactor Power Plant," L. L. KINTNER, P. M. WOOD, J. T. JACKMAN, and J. P. ERNST, p. 13-48 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Kidde (Walter) Nuclear Labs., Inc.

59. "A Conceptual Study of a 100 KW Gas-Cooled Reactor Power Plant for a Remote Location," J. H. FRANKFORT, May 31, 1955, WKNL-46 (Del.).

Los Alamos Scientific Lab.

60. "Turret: A High Temperature Gas-Cycle Reactor Proposal," R. PHILIP HAMMOND, HAROLD M. BUSEY, KENNETH R. CHAPMAN, FRANKLIN P. DURHAM, JOHN D. ROGERS, and W. R. WYKOFF, Jan. 23, 1958, LA-2198.

Martin Co., Nuclear Div.

61. "The Origin and Disposal of Power Reactor Wastes," GEORGE P. DIX, RICHARD C. GROCUP, and JOHN M. LEFFLER, Feb. 5, 1958, MND-1235.
62. "Development of Iron-Aluminum Base Alloys for Gas-Cooled Reactor Components, Quarterly Progress Report No. 2 [for] March 1 through May 31, 1958," J. MUELLER, June 1958, MND-DB-2522.

63. "Development of Iron-Aluminum Base Alloy for Gas-Cooled Reactor Components, Final Report," FRANK TATE, April 1959, MND-DB-2525.

Nuclear Development Corp. of America

64. "Design of an Experimental Reactor for High Operating Temperatures," L. DAVIDSON, J. DEFELICE, and J. A. KLAPPER, Sept. 28, 1956, NDA-64-101.

Oak Ridge National Laboratory

65. "Design Study for a Graphite-Moderated Gas-Cooled Reactor Using Partially Enriched Uranium," R. A. CHARPIE, M. BENDER, A. P. FRAAS, W. T. FURGERSON, W. H. JORDAN, W. D. MANLY, R. V. MEGHREBLIAN, A. M. PERRY, and R. D. SCHULTHEISS, A/CONF.15/P/450.
66. "High Temperature Ceramic Power Reactors Using SiC-Si Fuel Elements," J. R. JOHNSON, R. L. HAMNER, A. J. TAYLOR, and J. M. WARDE, Jan. 10, 1956, CF-56-1-30 (Del.).
67. "Comparison of Gases for Use as the Coolant in a Gas-Cooled Reactor," G. SAMUELS, April 28, 1958, CF-58-4-108.
68. "Temperature Structure in Gas-Cooled Reactor Fuel Elements and Coolant Channel," L. G. EPEL and W. T. FURGERSON, May 27, 1958, CF-58-5-97.
69. "An Analysis of the Magnesium-CO₂ Reactions in Terms of a Reactor Environment," J. L. SCOTT, May 15, 1958, CF-58-5-100.
70. "Analog Computer Study of the Operation of the Gas-Cooled Loop on the Oak Ridge Research Reactor," F. P. GREEN, F. H. NEILL, B. E. SHORT, and M. L. WINTON, July 7, 1958, CF-58-7-48.
71. "Review of S and P 1963, The Pebble Bed Reactor," July 17, 1958, CF-58-7-65.
72. "Technical Scope of Gas-Cooled Reactor Fuel Element Irradiation Program," Aug. 1, 1958, CF-58-8-4.
73. "Supplementary Comments to ORNL CF-58-5-97," L. G. EPEL, Aug. 12, 1958, CF-58-8-37.
74. "Nuclear Gas Engine," A. P. FRAAS, Sept. 25, 1958, CF-58-9-12.
75. "Evaluation of the Iodine Vapor-Fission Gas Adsorption Traps for ORR-705 Capsule Experiment, GCPR Capsule Irradiation Program," R. E. ADAMS and W. E. BROWNING, Dec. 23, 1958, CF-58-12-10.
76. "Literature Pertinent to a Steam Generator Design for a Gas-Cooled Reactor System," R. I. GRAY, Oct. 24, 1958, CF-58-10-87.

77. "Temperature Structure in the Gas-Cooled Reactor Fuel Elements Using a Scalloped Coolant Channel," L. G. EFEL, March 16, 1959, CF-59-3-39.
78. "The ORNL Gas-Cooled Reactor, Part I, Summary Report, Parts 2, 3, and 4, Plant Design," April 1, 1958, ORNL-2500 (Pts. 1-4).
79. "The ORNL Gas-Cooled Reactor, Materials and Hazards," May 22, 1958, ORNL-2505 (Rev.).
80. "The ORNL Gas-Cooled Reactor Advanced Concepts," Oct. 2, 1958, ORNL-2510.
81. "The HGCR-1, A Design Study of a Nuclear Power Station Employing a High-Temperature Gas-Cooled Reactor with Graphite-UO₂ Fuel Elements," W. B. COTTRELL, C. M. COPENHAVER, H. N. CULVER, M. H. FONTANA, V. J. KELLEGHAN, and G. SAMUELS, July 28, 1959, ORNL-2653.
82. "Gas-Cooled Reactor Project Semiannual Progress Report for Period Ending December 31, 1958," March 16, 1959, ORNL-2676.
83. "Gas-Cooled Reactor Coolant Choice," June 15, 1959, ORNL-2699.
84. "High-Temperature Ceramic Power Reactors Using a SiC-Si Fuel Elements," J. R. JOHNSON, R. L. HAMNER, A. J. TAYLOR, and J. M. WARDE, p. 111-21; 125-31 [of] Nuclear Science and Technology, Vol. 2, No. 2 (deleted), June 1956, TID-2022 (Del.).
85. "Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958," TID-7564.
86. "The ORNL Gas-Cooled Reactor Design," A. P. FRAAS, p. 119-37 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
87. "Materials Selections and Development Work," W. D. MANLY and J. H. COOBS, p. 138-64 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
88. "Materials Compatibility and Fuel Element Test Program," D. B. TRAUGER, p. 165-84 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
89. "Cost as a Design Criterion," M. BENDER, p. 325-33 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
90. "Advanced Design Studies," A. M. PERRY, p. 334-46 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
91. "Release of Activity From Various GCR Systems," W. B. COTTRELL, p. 347-62 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.
92. "Advanced Materials Development," J. H. COOBS, p. 363-72 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Oak Ridge School of Reactor Technology

93. "A Preliminary Study of the Feasibility of a Gas-Cooled, Graphite-Moderated Reactor for Power Production; Reactor Design and Feasibility Problem," D. B. WILLMER, J. H. CROWLEY, Z. LEVINE, R. G. MALLON, M. POTTER, J. B. STICHKA, and J. T. THOMAS, Aug. 20, 1952, CF-52-8-226 (Del.).
94. "Gas-Cooled, Natural Uranium, D₂O Moderated Power Reactor; Reactor Design and Feasibility Problem," R. C. DAHLBERG, E. G. BEASLEY, T. K. DEBOER, T. C. EVANS, D. F. MOLINO, W. S. ROTHWELL, and W. R. SLIVKA, Aug. 1946, CF-56-8-207 (Del.).
95. "Gas-Cooled Pebble Bed Reactor for a Large Central Station; Reactor Design and Feasibility Study," A. SCHOCK, D. F. BRULEY, H. N. CULVER, P. W. IANNI, W. F. KAUFMAN, R. A. SCHMIDT, and R. E. SUPP, Aug. 1957, CF-57-8-12.

Sanderson and Porter

96. "A High Temperature-Gas Cycle Pebble Bed Reactor for Central Station Use," S. T. ROBINSON and R. F. BENENATI, p. 193-215 [of] Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, TID-7564.

Sanderson and Porter, New York and Alco Products, Inc.

97. "Design and Feasibility Study of a Pebble Bed Reactor-Steam Power Plant," May 1, 1958, NYO-8753 (Vol. I).

Stanford University

98. "The Influence of Non-Uniform Heat Flux on the Convection Conductances in a Nuclear Reactor, Technical Report No. 33," W. M. KAYS and W. B. NICOLL, Nov. 1, 1957, AD-151340.

Journal Articles

99. "The Temperature Distribution in a Gas-Cooled Reactor of Special Form," OLAVI HELLMAN, *Ann. Acad. Sci. Fennicae Ser. A, VI*, No. 12 (1958).
100. "The Change of the State of the Coolant Gas in Certain Gas-Cooled Reactors," OLAVI HELLMAN, *Ann. Acad. Sci. Fennicae Ser. A, VI*, No. 13 (1958).
101. "Nuclear Power for Commercial Vessels, II," K. MADDOCKS, *Atomics* 7, 197-201, 224, June 1956.
102. "Nuclear Power for Commercial Vessels, III," K. MADDOCKS, *Atomics* 7, 291-5, August 1956.
103. "Nuclear Power for Commercial Vessels, IV," K. MADDOCKS, *Atomics* 7, 309-14, September 1956.
104. "Nuclear Power and Gas Turbines-2, Joint U.S.-Swiss Design Study," *Atomics* 8, 86-92, March 1957.
105. "C.C.G.C. Reactor for Ships," *Atomics and Nuclear Energy* 9, 162-3, May 1958.
106. "The Liquid-Metal-Fuel Reactor Closed-Cycle Gas-Turbine Power Plant," L. D. STOUGHTON and T. V. SHEEHAN, *Mech. Eng.* 78, 699-702, August 1956.
107. "GCR-2, The ORNL Design Study," *Nuclear Eng.* 3, 277-81, July 1958.
108. "Design Parameters in the ORNL Gas-Cooled Reactor Study," *Nuclear Eng.* 3, 282-5, July 1958.
109. "A New Concept in Power Plants," A. P. FRAAS, *Nuclear Power* 4, No. 38, 118-9, June 1959.
110. "A Preliminary Study of Superheating Boiling Reactors," M. TRESHOW, *Nuclear Sci. and Eng.* 1, 167-73, May 1956.
111. "Magnesium-Clad Wedge S-Shaped Fuel Plates for Gas-Cooled Reactors," GUY H. CANNON, *Nuclear Sci. and Eng.* 5, 219-24, April 1959.
112. "Fission Product Monitoring Techniques in Gas-Cooled Reactors," E. S. COLLINS, P. K. CONN, and J. B. TRICE, *Nuclear Sci. and Eng.* 2, No. 1, Suppl., 209-10, June 1959.
113. "Isotope Deposition Hazards in Gas Cycle Reactors," P. K. CONN, E. S. COLLINS, and J. B. TRICE, *Nuclear Sci. and Eng.* 2, No. 1, Suppl., 210-11, June 1959.
114. "Small Gas-Cycle Reactor Offers Economic Promise," FARRINGTON DANIELS, *Nucleonics* 14, No. 3, 34-41, March 1956.
115. "Progress on Materials—Key to Gas-Cycle Reactor Potential," *Nucleonics* 14, No. 3, 42-4, March 1956.
116. "PWR and Calder Hall. How Do They Compare?," D. P. HERRON and A. PUISHES, *Nucleonics* 15, No. 6, 72-5, June 1957.
117. "Spiral Fuel Element for Gas-Cooled Reactor," R. C. DAHLBERG, and T. C. EVANS, *Nucleonics* 16, No. 4, 106-8, April 1958.
118. "Temperature-Zoned Reactor," F. H. CLARK and C. N. KLAHR, *Nucleonics* 16, No. 5, 114-17, May 1958.
119. "U.S. Designs Gas-Cooled Reactors," *Nucleonics* 16, No. 8, 118-21, August 1958.
120. "New Reactor Concepts," *Nucleonics* 16, No. 9, 78-82, September 1958.
121. "Gas-Cooled Projects," *Nucleonics* 17, No. 2, 52-3, February 1959.

Books

122. "Progress in Nuclear Energy, Series II, Reactors," R. A. CHARPIE, D. J. HUGHES, D. J. LITTLER, and M. TROCHERIS (eds.), New York, McGraw-Hill Book Co., Inc., 1956.
123. "A Comparison of Three Nuclear Power Plant Concepts for Power Generation in the United States," KARL COHEN and E. L. ZEBROSKI, p. 113-31 of "Proceedings of the American Power Conference, March 26, 27, 28, 1958," Volume 20, Chicago, Illinois Institute of Technology, 1958.
124. "Economics of Nuclear and Conventional Merchant Ships," DAVID L. CONKLIN, J. J. GORDON, R. J. JAFFE, S. R. NIXON, R. E. QUILICI, H. N. ROBEY, and M. I. TEMME, Mountain View, Calif., American-Standard, 1958.
125. "Solid Fuel Reactors," JOSEPH R. DIETRICH and WALTER H. ZINN (eds.), Reading, Mass., Addison-Wesley Publishing Company, Inc., 1958.
126. "Advances in Nuclear Engineering, Volume II," JOHN R. DUNNING and BRUCE R. PRENTICE (eds.), New York, Pergamon Press, 1957.
127. "Fluid Fuel Reactors," JAMES A. LANE, H. G. MACPHERSON, and FRANK MASLAN (eds.), Reading, Mass., Addison-Wesley Publishing Company, Inc., 1958.
128. "An Introduction to Reactor Physics," D. J. LITTLER and J. F. RAFFLE, New York, McGraw-Hill Book Co., Inc., 1957.
129. "Management, Economics and Technology for the Atomic Energy Industry, Vol. I: Management and International Affairs, Vol. II: Economics and Technology," WILLIAM J. SMITH (ed.), New York, The Atomic Industrial Forum, Inc., 1956.
130. "Reactors of the World, First Series," New York, Simmons-Boardman Publishing Corporation, 1958.

Patents

131. "Fuel Element Loading Apparatus for Neutronic Reactors," G. A. ANDERSON (to U.S. Atomic Energy Commission), U.S. Patent 2,794,562, June 4, 1957.
132. "Thermal Nuclear Reactor," F. W. FENNING and R. F. JACKSON (to U.S. Atomic Energy Commission), U.S. Patent 2,807,580, Sept. 24, 1957.
133. "Neutronic Reactor System," F. DANIELS (to U.S. Atomic Energy Commission), U.S. Patent 2,809,931, Oct. 15, 1957.
134. "Method of Sustaining a Neutronic Chain Reacting System," E. FERMI and M. C. LEVERETT (to U.S. Atomic Energy Commission), U.S. Patent 2,813,070, Nov. 12, 1957.
135. "Reactor Unloading," M. C. LEVERETT (to U.S. Atomic Energy Commission), U.S. Patent 2,824,056, Feb. 18, 1958.
136. "Air-Cooled Neutronic Reactor," E. FERMI and L. SZILARD (to U.S. Atomic Energy Commission), U.S. Patent 2,836,554, May 27, 1958.
137. "Chain Reacting System," E. FERMI and M. C. LEVERETT (to U.S. Atomic Energy Commission), U.S. Patent 2,837,477, June 3, 1958.
138. "Gas-Cooled Nuclear Reactors," E. LONG and W. RODWELL (to U.S. Atomic Energy Commission), U.S. Patent 2,838,451, June 10, 1958.
139. "Nuclear Reactor," R. V. MOORE, J. H. BOWEN, and K. H. DENT (to U.S. Atomic Energy Commission), U.S. Patent 2,863,815, Dec. 9, 1958.
140. "Nuclear Reactors," E. LONG and J. W. ASHLEY (to U.S. Atomic Energy Commission), U.S. Patent 2,864,759, Dec. 16, 1958.
145. "Interpretation of Critical Experiments on BR-1, Evaluation of the Laplacian," J. P. LAGNAUX, July 23, 1958, NP-7135. (In French.)
146. "Use of Rods with Magnox Fins of Reduced Diameter in the BR-1," J. JULIENS and P. DE MYTTENAERE, May 22, 1958, NP-7136. (In French.)
147. "Experimental Channel for the Study of Heat Removal from the BR-1 Reactor," P. MARIEN, Nov. 15, 1957, NP-7149. (In French.)
148. "BR-1, Calculation of the Energy Accumulated in the Graphite by the Wigner Effect," G. PENELLE, Sept. 4, 1958, NP-7156. (In French.)
149. "Activity Scattered in the Surrounding in Normal Operation of the BR-1," L. VANDEWIELE, Dec. 23, 1957, NP-7246. (In Dutch.)
150. "Heat Removal from the BR-1 Reactor," P. MARIEN, Presented at the European Atomic Energy Society Symposium on Research Reactors and their Applications, Noordwijk, Netherlands, Sept. 10-14, 1957. (In French.)
151. "Experience with the BR-1 Reactor," M. NEVE DE MEVERGNIES, R. SNOY, and G. STIENNON, Presented at the European Atomic Energy Society Symposium on Research Reactors and their Applications, Noordwijk, Netherlands, Sept. 10-14, 1957. (In French.)
152. "Some Technical Aspects of the BR-1 Reactor," G. STIENNON, Presented at the European Atomic Energy Society Symposium on Research Reactors and their Applications, Noordwijk, Netherlands, Sept. 10-14, 1957. (In French.)

BELGIUM**Reports**

141. "Heat Removal from Gas-Cooled Reactors, Comparison of Different Types of Claddings," P. MARIEN, A/CONF.15/P/324. (In French.)
142. "The BR-1 Pile, Part I, Description and Use," G. STIENNON, Part II, "Physical Characteristics," M. NEVE DE MEVERGNIES, C. FAIRCLOUGH, G. STIENNON, and F. LEONARD, A/CONF.15/P/1673. (In French.)
143. "Study of an Enriched Uranium-Graphite System Cooled by Air," H. DOPCHIE, April 13, 1956, NP-6278. (In French.)
144. "BR-1 Operation, Progress Report, Second Quarter 1958," M. DE DORLODOT, NP-7129. (In French.)

CZECHOSLOVAKIA**Reports**

153. "Natural Uranium and Heavy Water Reactors," F. KLIK and J. MARKVART, A/CONF.15/P/2094.
154. "Intensification of Heat Transfer in a Gas-Cooled Nuclear Reactor," F. BERGER and V. STACH, A/CONF.15/P/2486.

Journal Articles

155. "A Thermal Cycle for Nuclear Power Plant with Gas-Cooled Reactor," JAN JUZA, *Jaderná energie* 4, No. 1, 2-9, January 1958. (In Czech.)
156. "Intensification of Heat Transfer in a Gas-Cooled Nuclear Reactor," F. BERGER and V. STACH, *Jaderná energie* 4, 251-6, September 1958. (In Czech.)

FRANCE

Reports

157. "Recent Progress in the Detection of Bursts in the Canning in French Reactors, G1, EL2, G3, EL3," J. GOUPIL, M. GRENON, J. RAFFAILLAC, and A. ROGUIN, A/CONF.15/P/334. (In French.)
158. "Launching a Program of Pressurized Gas-Cooled Reactors," J. HOROWITZ and J. YVON, A/CONF.15/P/343.
159. "Experience Acquired During Two Years of Operation of Reactor G1," DE ROUVILLE, PASCAL, and SCALLIET, A/CONF.15/P/1132.
160. "General Specifications and Original Aspects of Reactors G2 and G3," PASCAL, [J.] HOROWITZ, [J.] BUSSAC, JOATTON, DE LAGGE DE MEUX, and MARTIN, A/CONF.15/P/1133.
161. "Description of Reactors G-2-G3," A/CONF.15/P/1134.
162. "The Chinon Nuclear Power Plant Divisions EDF 1 and ERF 2," M. ROUX and M. BIENVENU, A/CONF.15/P/1135.
163. "Possibilities and Conditions of Use of Water or Gas Type Reactors in a Tanker. I. Pressurized Water Reactor," L. BUREAU DU COLOMBIER, J. HAINZELIN, and P. DUQUESNE, "II. Boiling Water Reactor," J. RICARD and P. DOMENJOUR, "III. Gas Reactor," R. GIBRAT, A/CONF.15/P/1137.
164. "Thermal Exchanges in Gas Reactors," J. LE FOLL, P. GELIN, and E. DE ROBERT, A/CONF.15/P/1145.
165. "Hygrometric Control of Reactor Cooling Gas," A. BRIOT and J. TOURET, A/CONF.15/P/1151. (In French.)
166. "The Fuel Elements in Pressurized Gas Reactors," J. A. STOHR, M. ENGLANDER, and M. GAUTHRON, A/CONF.15/P/1157.
167. "Partial Combustion of a Fuel Cartridge in Reactor G1," DE ROUVILLE, LEDUC, and SEGOT, A/CONF.15/P/1180.
168. "Natural Uranium-Graphite System," A. P. SCHMITT, P. TANGUY, A. TESTE DU BAILLER, and C. P. ZALESKI, A/CONF.15/P/1191.
169. "Vessel and Heat Exchanger of the EDF 1 Chinon Power Plant," LAMIRAL and LANCEL, A/CONF.15/P/1199.
170. "Control of the EDF 1 Power Plant," DELOUX and LAURENT, A/CONF.15/P/1200. (In French.)
171. "Loading and Unloading Problems for the Reactor EDF 1," LEO, KAPLAN, and SEGARD, A/CONF.15/P/1201. (In French.)
172. "Measurement of the Reactivity of the G-1," J. BERNOT, J. C. KOECHLIN, L. PORTES, and A. TESTE DU BAILLER, 1957, AEC-tr-3434.
173. "Measuring the Thermal Utilization Factor," F. ROULLIER and A. P. SCHMITT, 1957, AEC-tr-3435.
174. "French Results on the Natural Uranium-Graphite Lattices," J. C. KOECHLIN, P. TANGUY, and C. P. ZALESKI, p. 97-126 [of] Proceedings of the French-American Conference on Graphite Reactors, [Held at Brookhaven National Laboratory], November 12 to 15, 1957, BNL-489.
175. "Reactivity Measurements in G-1," J. BERNOT, p. 232-7 [of] Proceedings of the French-American Conference on Graphite Reactors, [Held at Brookhaven National Laboratory], November 12 to 15, 1957, BNL-489.
176. "Description of the First French Graphite Reactor and of the Experiments Performed in 1956," J. BUSSAC, C. LUDUC, and C. P. ZALESKI, 1957, CEA-670(A). (In French.)
177. "Diffusion of Graphite—The Effect of Cylindrical Canals," R. CARLE, C. CLOUET D'ORVAL, R. LATTES, J. MARTELLY, T. DE MAZANCOURT, C. ROBERT, M. SAGOT, and A. TESTE DU BAILLER, 1957, CEA-670(B). (In French.)
178. "Critical Size and Flux Distributions in the Shut Down Pile," A. BANCHEREAU, P. BERTHIER, J. -P. GENTHON, C. GOURDON, R. LATTES, J. MARTELLY, R. DE MAZANCOURT, L. PORTES, M. SAGOT, A. -P. SCHMITT, P. TANGUY, A. TESTE DU BAILLER, and A. VEYSSIERE, 1957, CEA-670(C). (In French.)
179. "A Study of Uranium—Thorium Mixed Lattices," P. BACHER, R. ECKERT, and R. DE MAZANCOURT, 1957, CEA-670(D). (In French.)
180. "Variation of the Material Laplacian of G-1 with the Radius of the Uranium Bar," P. TANGUY, 1957, CEA-670(E). (In French.)
181. "Measurements of Reactivity," J. BERNOT, J. C. KOECHLIN, L. PORTES, and A. TESTE DU BAILLER, 1957, CEA-670(F). (In French.)
182. "Relative Measurement of the Fluxes of Thermal, Resonant and Rapid Neutrons in Reactor G1," R. CARLE and T. DE MAZANCOURT, 1957, CEA-670(G). (In French.)
183. "Measurement of the Antireactivity of a Control Rod of G1, by a Slow Oscillation Method," D. BRETON, J. LEROY, and R. VIDAL, 1957, CEA-670(H). (In French.)
184. "Measurement of the Resonance Absorption," J. P. ANTHONY, P. BACHER, L. LHEUREUX, J. MOREAU and A. P. SCHMITT, 1957, CEA-670(I). (In French.)

185. "Measurement of the Thermal Utilization Factor of the Reactor G1," F. ROULLIER and A. P. SCHMITT, 1957, CEA-670(J). (In French.)
186. "Measurement of the Temperature of the Neutrons in Reactor G1," V. RAJEVSKI and B. SAUTIEZ, 1957, CEA-670(K). (In French.)
187. "'Reheating' Experiment in the 35-ton Pile," J. CHEROT and Y. GIRAD, 1957, CEA-670(L). (In French.)
188. "High Power Experiments," F. DE LAAGE, H. RETEL, A. TESTE DU BAILLER, A. VEYSSIERE, and G. WANNER, 1957, CEA-670(M). (In French.)
189. "Observations on the Electronic Equipment Employed for Making Measurements on the Pile G1," J. AILLOUD, P. BLIN, A. MEUNIER, and A. TARABELLE, 1957, CEA-670(N). (In French.)
190. "Photoneutron Source for the Pile G1," J. LE-ROY and R. WUSTNER, 1957, CEA-670(O). (In French.)
201. "Treatment of the Fluid," P. DEQUENES and M. GARNIER, *Bull. inform. sci. et tech.* No. 20, 115-25, August 1958. (In French.)
202. "Control of the G-2 and G-3 Reactors," A. ERTAUD and A. PAGES, *Bull. inform. sci. et tech.* No. 20, 126-39, August 1958. (In French.)
203. "Electro-Mechanical Installations," M. LIGNIERES, J. BOBIGEAT, and A. DARRE, *Bull. inform. sci. et tech.* No. 20, 140-51, August 1958. (In French.)
204. "Corrosion Problems in Gas-Cooled Reactors," R. DARRAS, *Bull. inform. sci. et tech.* No. 21, 13-25, September 1958. (In French.)
205. "Annealing of the Graphite of the G-1," M. PASCAL, *Bull. inform. sci. et tech.* No. 23, 11-12, November 1958. (In French.)
206. "The Production of Electrical Energy of Nuclear Origin," AILLERET, p. 159-67 of *Journées d'Information sur l'Energie Nucleaire*, 14-15-16 Janvier 1957, *Documents*, Suppl. to No. 114, 1957. (In French.)

Journal Articles

191. "The Reactors at Marcoule," *Atomkernenergie* 2, 58-62, February 1957. (In German.)
192. "The French Dual-Purpose Reactors G-2 and G-3," K. YU, *Atomnaya Energiya* 3, 560-1, (1957).
193. "Neutron Study of the G-2 and G-3," A. TESTE DU BAILLER, *Bull. inform. sci. et tech.* No. 20, 1-9, August 1958. (In French.)
194. "Thermodynamic Study of the G-2 and G-3," J. CHABOSEAU and P. LAFERRIERE, *Bull. inform. sci. et tech.* No. 20, 10-27, August 1958. (In French.)
195. "Safety Devices of the G-2," C. GUIONNET, *Bull. inform. sci. et tech.* No. 20, 28-35, August 1958. (In French.)
196. "The Core of the G-2 and G-3 Reactors," G. DELEUZE, *Bull. inform. sci. et tech.* No. 20, 36-55, August 1958. (In French.)
197. "Civil Engineering of the Pressure Vessels," J. BELLIER and M. TOURASSE, *Bull. inform. sci. et tech.* No. 20, 56-68, August 1958. (In French.)
198. "Loading and Unloading," A. ERTAUD and G. DEROME, *Bull. inform. sci. et tech.* No. 20, 69-88, August 1958. (In French.)
199. "The Coolant Circuits," J. CHABOSEAU and P. LAFERRIERE, *Bull. inform. sci. et tech.* No. 20, 89-98, August 1958. (In French.)
200. "Installations for Energy Recovery," P. PAS-SERIEUX and R. SCALLIET, *Bull. inform. sci. et tech.* No. 20, 99-144, August 1958. (In French.)
207. "The French Center for Plutonium Production at Marcoule," *Energie nucléaire* 1, 141-4, July-September 1957. (In French.)
208. "The Cooling of the Reactors at Marcoule," J. CHABOSEAU, *Industries atomiques* No. 7 and 8, 67-72, June-July 1957. (In French.)
209. "Reactors and Power Production at Marcoule," *Nuclear Eng.* 1, 22-4, April 1956.
210. "Reactor G1 at Marcoule," *Nuclear Eng.* 3, 258, June 1958.
211. "Thermodynamics of Gas-Cooled Reactors," GILBERT B. MELESE, *Nucleonics* 16, No. 2, 72-6, February 1958.
212. "Safety of Magnesium Canning for CO₂-Cooled Reactors," MARC SALESSE, *Nucleonics* 16, No. 2, 123-4, February 1958.
213. "Completed and Projected Work of Electricité de France in the Nuclear Field," J. P. ROUX, *Rev. gen. elec.* 67, 351-64, July 1958. (In French.)
214. "Removal and Recovery of Heat from Nuclear Reactors," RICHARD ALAMI and PAUL AGERON, Paris, Dunod, 1958. (In French.)
215. "Temperature Measurements in French Nuclear Reactors," P. JOVER and P. BELIN, Presented at Nuclear Engineering and Science Conference, held at Chicago, March 17 to 21, 1958, Preprint 151, Session 15, New York, American Institute of Chemical Engineers, 1958.
216. "Instrumentation and Control of French Thermal Reactors," P. BONNAURE, Presented at

Nuclear Engineering and Science Conference, held at Chicago, March 17 to 21, 1958, Preprint 152, Session 21, New York, American Institute of Chemical Engineers, 1958.

217. "The Reactors of the French Program," JULES HOROWITZ, pp. 127-38 in "Atti del Congresso Scientifico, Volume I, Sezione Nucleare, Giugno 1957." (In French.)
218. "The Experimental Utilization of the EL-2 Reactor with Average Flux," M. SEGUIN, Presented at the European Atomic Energy Society Symposium on Research Reactors and their Applications, Noordwijk, Netherlands, Sept. 10-14, 1957. (In French.)

GERMANY

Reports

219. "Fuel Elements for a High-Temperature Reactor," A. BOETTCHER, and G. WIRTHS, and R. SCHULTEN, A/CONF.15/P/1005.
220. "The Pebble-Bed High-Temperature Reactor for 15 MW (Electr.)," R. SCHULTEN, A/CONF. 15/P/1054.
221. "Nuclear Power in the Federal Republic of Germany," SIEGFRIED BALKE, A/CONF.15/P/1806.
222. "Flux Fine-Structure in Gas-Cooled Reactor Cells," G. RIESCH, from *Atomkernenergie* 3, 15-21 (1958), AERE-Lib/Trans-811.
223. "Steam-Cooled Power Reactor," A. SCHRAUD, from *Atomkernenergie* 3, 133-6 (1958), IGR-T/R-89.

Journal Articles

224. "Basic Problems of Heat Transfer in Nuclear Reactors," HANS DIETER BAEHR, *AEÜ Mitt.* 48, 1-11, January 1958. (In German.)
225. "Nuclear Power Stations with Light-Water-Cooled or Gas-Cooled Reactors," ALFRED SCHULLER, *AEÜ Mitt.* 48, 47-57, January 1958. (In German.)
226. "Ship Propulsion by Nuclear Power, 1," K. ILLIES, *Atomics and Nuclear Energy* 9, 156-9, 184, May 1958.
227. "Dependence of Performance of Gas-Cooled Nuclear Reactors on the Cooling Gas Pressure and the Reactor Dimensions," ALFRED SCHRAUD, *Atomkernenergie* 2, 204-7, June 1957. (In German.)
228. "Nuclear Technology—Ship Propulsion Design with Gas Turbines," K. ILLIES, *Atomkernenergie* 2, 368-76, October 1957. (In German.)

229. "Ship Propulsion with Enclosed Gas Turbines and Gas-Cooled Reactors," K. BAMMERT, *Atomkernenergie* 2, 381-4, October 1957. (In German.)
230. "Special Nuclear Reactors for Ships," W. JUNKERMANN, *Atomkernenergie* 2, 385-90, October 1957. (In German.)
231. "Flux Fine Structure in Gas-Cooled Reactor Cells," GERHARD RIESCH, *Atomkernenergie* 3, 16-21, January 1958. (In German.)
232. "Calculation of the Optimum Coolant State in Gas-Cooled Reactors," H. BENZLER, *Atomkernenergie* 3, 331-6, August-September, 1958. (In German.)
233. "A Gas-Cooled Nuclear Reactor of High Thermal Efficiency," F. WINTERBERG, *Atomkernenergie* 4, 2-6, January 1959. (In German.)
234. "Heat Engineering Calculations of a Gas-Cooled Reactor," HANS DIETER BAEHR, *Atomkernenergie* 4, 7-11, January 1959. (In German.)
235. "Supplement to the Calculation of the Cooling Efficiency of Reactor Coolant Channels," H. BENZLER, *Atomkernenergie* 4, 12-13, January 1959. (In German.)
236. "Thermal Capacity in Reactors," M. LEDINEGG, *Atomkernenergie* 4, 14-22, January 1959. (In German.)
237. "Systematic Investigations on the Heat Transfer and Flow Resistance of Finned Tubes, A Contribution to the Application of Special Tubes in Gas-Cooled Reactors," GUNTER GRASS and FRANZ-PETER COENEN, *Atomkernenergie* 4, 41-8, February 1959. (In German.)
238. "Heat Engineering Calculations of Gas-Cooled Reactors, II," HANS DIETER BAEHR, *Atomkernenergie* 4, 49-55, February 1959. (In German.)
239. "Supplementary Control of Power Reactors by Enriched Shim Rods," D. SMIDT, *Atomkernenergie* 4, 129-31, April 1959. (In German.)
240. "Recent Projects on Reactors with Gaseous Coolant," W. KLIEFORTH, *Atomkernenergie* 4, 196-201, May 1959. (In German.)
241. "Nuclear Energy Propulsion Plant with Gas-Cooled Reactor for a 10,000 SHP Tanker," K. ILLIES, *Atomkernenergie* 4, 213-18, June 1959. (In German.)
242. "Explanation of the Gas Circuit of a 10,000 SHP Nuclear Energy Gas Turbine Ship Propulsion Plant," JURGEN BUSCH, *Atomkernenergie* 4, 218-22, June 1959. (In German.)
243. "Heat Exchangers of a Nuclear Energy Gas-Cooled Ship Propulsion Plant," HANS-G. PAUL, *Atomkernenergie* 4, 222-6, June 1959. (In German.)

244. "The Reactor, Its Construction and Control," WOLFGANG HUBSCHMANN, *Atomkernenergie* 4, 226-30, June 1959. (In German.)
245. "Installation of a Nuclear Energy Gas Turbine Plant in a Tanker," GEROLD, *Atomkernenergie* 4, 235-8, June 1959. (In German.)
246. "Selection of Structural Materials for Gas-Cooled Ship Reactor," W. JUNKERMANN, *Atomkernenergie* 4, 238-41, June 1959. (In German.)
247. "The Present State of Development of Gas-Cooled Reactors for High Temperatures," A. THIEL, *Atomkernenergie* 4, 241-4, June 1959. (In German.)
248. "Fuel Elements for Gas-Cooled Reactors," M. BECKER, *Atomkernenergie* 4, 244-7, June 1959. (In German.)
249. "When Will Atomic Power Plants be Economical? The Prospects of the Gas-Cooled Reactor," W. JUNKERMANN, *Atomwirtschaft* 3, 96-8, March 1958. (In German.)
250. "The 10 MW High-Temperature Reactor 'Dragon'," H. W. SCHMIDT, *Atomwirtschaft* 4, 211-16, May 1959. (In German.)

Patents

251. "Gas-Cooled Nuclear Reactor in Which the Moderating System is Separated from the Coolant System by a Barrier Layer," (To Siemens-Schuckertwerke, AG., Berlin u. Erlangen, Ger.), German Patent DAS 1031 901, *Atompraxis* 5, 42 (1959). (In German.)
252. "Gaseous Coolant for Nuclear Reactors Based on the Increased Heat Capacity of Dissociated Gases," (To Siemens-Schuckertwerke AG., Germany), German Patent DAS 1033 344, *Atompraxis* 4, 477 (1958). (In German.)
253. "Nuclear Reactor with Replacement Installations Between the Fuel Lattice," (To Siemens-Schuckertwerke AG., Berlin u. Erlangen, Ger.), German Patent DAS 1033 808, *Atompraxis* 5, 43 (1959). (In German.)

INDIA

Reports

254. "Natural Uranium Reactors for Small and Medium Power Outputs," M. DAYAL, S. R. PARANJTE, and M. R. SRINIVASAN, A/CONF.15/P/1641.
255. "Nuclear Study of a Low Power Beryllium Oxide Natural Uranium Power Reactor," R. D. JAIN and K. R. SRINIVASAN, A/CONF.15/P/1945.

ITALY

Reports

256. "Considerations on the Use of Thorium in Power Reactors," D. CIUFFA and L. SANI, A/CONF.15/P/1380.

Journal Articles

257. "Fuel Circulation in Natural Uranium and Graphite Reactors," G. GHILARDOTTI, *Energia nucleare (Milan)* 5, 95-103, February 1958. (In Italian.)
258. "Criteria for Choice of a Nuclear Moderator with Particular Reference to Graphite," G. GANDUSIO, P. PALMIERI, and P. M. STROCCHI, *Energia nucleare (Milan)* 5, 105-15, February 1958. (In Italian.)
259. "Control in Some Types of Nuclear Power Reactors," A. TESCARI, *Energia nucleare (Milan)* 5, 117-30, February 1958. (In Italian.)
260. "The Accident at Windscale and the Radiation Damage to Graphite," G. GANDUSIO and S. POLEZZO, *Energia nucleare (Milan)* 5, 289-96, May 1958. (In Italian.)
261. "The System CO₂-CO-C as Heat Exchange Medium in Nuclear Reactors," Ugo MIRANDA, *Ingegneria (Milan)*, 32, 489-92, June 1958. (In Italian.)
262. "Power Reactors: Actual Techniques," MARIO SILVESTRI, pp. 129-46 in "Proceedings of the Scientific Congress Nuclear Sessions, July 2-6, 1956, 3d., Electronic-Nuclear International Review," Rome, National Committee for Nuclear Research, 1956. (In Italian.)

NETHERLANDS

Books

263. Kernreactortechnologie, (Nuclear Reactor Technology.), RCN Mededeling Nr. 4, 1958, H. DE BRUYN, M. BUSTRAAN, J. J. WENT, H. WIJCKER, K. H. DENT, J. J. SYRETT, J. A. M. HAAGH, W. J. KAUFMAN, K. K. DE JONG, J. A. GOEDKOOP, P. MOSTERT, J. GOENS, and A. H. W. ATEN, s'Gravenhage, Netherlands, Reactor Centrum Nederland, 1958. (In English, Dutch, and French.)

SWEDEN

Reports

264. "Gas Cooled Natural Uranium Reactors for Large Power Stations, Three Designs," RAGNAR LILJEBLAD and D. MADSEN, A/CONF.15/P/2419.

265. "Optimization of Gas-Cooled Reactors with the Aid of Mathematical Computers," P. H. MARGEN, August 1958, AE-20.

Journal Articles

266. Swedish Power Reactor Proposal, RAGNAR LILJEBLAD, *Nucleonics* 15, No. 11, 165-9, November 1957.

SWITZERLAND

Reports

267. "Closed-Cycle-Gas-Turbine-Reactor-Power-Plant," W. WINKLER, Nov. 25, 1957, NP-6652.

Journal Articles

268. "Coolant Trap in Connection with a Closed-Cycle Gas Turbine for the Removal of Gaseous Fission Products from the Coolant (Helium) of a Nuclear Reactor," K. HINTERMANN and W. WINKLER, *Atomkernenergie* 3, 313-15, Aug.-Sept., 1958. (In German.)
269. "Circuits for the Gas-Cooled High-Temperature Reactor," W. SPILLMANN and R. TOGNONI, *Atomkernenergie* 4, 190-5, May 1959. (In German.)
270. "Thermodynamic Factors in the Selection of the Coolant for Nuclear Reactors," L. S. DZUNG, *Industries atomiques* 3, No. 3/4, 59-64, (1959). (In French.)

UNION OF SOVIET SOCIALIST REPUBLICS

Reports

271. "Rod Fuel Element for Gas-Cooled Heavy-Water Power Reactor," P. I. CHRISTENKO, P. A. PETROV, V. A. MEETROPOLEVSKY, K. D. SINELNIKOV, V. E. IVANOV, and V. F. ZELEN-SKY, A/CONF.15/P/2053.
272. "Experimental Reactor with Gaseous Fissionable Substance (UF_6)," I. K. KIKOIN, V. A. DMITRIEVSKII, Y. Y. GLAZKOV, I. S. GRIGORIEV, B. G. BUBOVSKII and S. V. KERSNOVSKII, A/CONF.15/P/2502.
273. "Perspectives of Nuclear Power in Closed Cycle Gas Turbines," A. I. MIKHAILOV, *Izvest. Akad. Nauk S.S.S.R., Otdel. Tekh. Nauk* No. 1, 113-19, January 1958. (In Russian.)
274. "A Gas-Cooled Heavy Water Power Reactor," A. I. ALIKHANOV, V. V. VLADIMIRSKII, P. A. PETROV, and P. I. CHRISTENKO, *Soviet J. Atomic Energy*, No. 1, 3-6. (1956.)

UNITED KINGDOM

Reports

275. "Graphite Moderated Thermal Reactor Calculations for the Calder Hall Reactors," P. W. MUMMEY, J. J. SYBETT, J. G. TYROB, and B. CUTTS, A/CONF.15/P/15.
276. "A Calculation of Resonance Escape Probability by Monte Carlo Methods," K. W. MORTON, A/CONF.15/P/19.
277. "The Derivation of Reactor Heat Transfer Transient Equations for Gas-Cooled Graphite Moderated Thermal Reactors," T. J. O'NEILL, A/CONF.15/P/21.
278. "Problems of Fueling in Industrial Calder Hall Type Reactors," R. R. GALLIE, H. H. GOTT, and F. C. W. COLMER, A/CONF.15/P/37.
279. "A Correlation of Heat Transfer for Finned Fuel Elements for Carbon Dioxide Reactors," D. F. HUGHES and M. R. SLACK, A/CONF.15/P/38.
280. "The Polyzonal Spiral Fuel Element," H. L. RITZ, A/CONF.15/P/48.
281. "Economics of Enrichment and of the Use of Plutonium and U-233," N. L. FRANKLIN, J. M. HILL, J. C. C. STEWART, and C. A. RENNIE, A/CONF.15/P/54.
282. "Kinetic and Control of the Berkeley Reactors," J. M. YELLOWLEES and P. R. J. FRENCH, A/CONF.15/P/55.
283. "The Detection of Burst Fuel Elements in Nuclear Reactors," K. W. CUNNINGHAM, G. V. HOUGH, and D. ALIAGA-KELLY, A/CONF.15/P/57.
284. "Current Re-Designs of Calder Hall," J. B. W. CUNNINGHAM, Includes Appendix: "Leakage of CO_2 from the Pressure Circuit of the Calder Hall Reactors," H. V. DISNEY, A/CONF.15/P/73.
285. "Hunterston Power Station," R. N. MILLAR, A/CONF.15/P/74.
286. "The Overall Control of Nuclear Power Stations of the Gas-Cooled Thermal Reactor Type," R. J. SMITH, A/CONF.15/P/85.
287. "Bradwell Nuclear Power Station," R. D. VAUGHAN and E. ANDERSON, A/CONF.15/P/263.
288. "The Berkeley Power Station," S. A. GHALIB and J. R. M. SOUTHWOOD, A/CONF.15/P/264.
289. "Nuclear Propulsion for Ships," R. V. MOORE and C. E. ILIFFE, A/CONF.15/P/266.
290. "Safety Aspects of the Calder Hall Reactor in Theory and Experiment," G. BROWN, H. KRONBERGER, F. M. LESLIE, J. MOORE, and P. W. MUMMEY, A/CONF.15/P/267.

291. "Some Methods of Calculating Nuclear Parameters for Heterogeneous Systems," S. E. BARDEN, D. BLACKBURN, A. Z. KELLER, and J. M. R. WATSON, A/CONF.15/P/272.
292. "Steam Cycles for Gas-Cooled Reactors," W. R. WOOTTON, A. J. TAYLOR and N. G. WORLEY, A/CONF.15/P/273.
293. "Chemical Studies of Carbon Dioxide and Graphite Under Reactor Conditions," A. R. ANDERSON and J. WRIGHT, H. W. DAVIDSON, R. LIND, D. R. STRANKS, and C. TYZACK, A/CONF.15/P/303.
294. "The Assessment and Testing of Fuel Elements," H. K. HARDY and H. LAWTON, A/CONF.15/P/306.
295. "A Monte Carlo Study of Thermal Utilization Factor and Diffusion Area: Gas-Cooled Graphite-Moderated Lattices," G. W. SCHAEFER and D. M. PARKYN, A/CONF.15/P/310.
296. "Theoretical Aspects of Fuel Cycling in Industrial Calder Hall Type Reactors," S. E. LEWIS and C. S. LOWTHIAN, A/CONF.15/P/311.
297. "Advances in the Design of Gas-Cooled Graphite-Moderated Power Reactors," R. V. MOORE, H. KRONBERGER, and L. GRAINGER, Includes Addendum 1, A/CONF.15/P/312.
298. "The Possibilities of Achieving High Temperatures in a Gas-Cooled Reactor," L. R. SHEPHERD, R. A. U. HUDDLE, L. A. HUSAIN, G. E. LOCKETT, F. STERRY, and D. V. WORDSWORTH, A/CONF.15/P/314.
299. "Reactor Physics Measurements During the Commissioning of the Calder Reactors," F. M. LESLIE, A/CONF.15/P/315.
300. "Development of Manufacturing Techniques for the Calder Hall Fuel Element," F. BUTLER, J. HARPER, I. H. MORRISON, and J. A. PARDOE, A/CONF.15/P/317.
301. "The High Temperature Zero Energy Reactor 'Zenith'," K. J. MITCHELL and R. A. GEARY, A/CONF.15/P/1463.
302. "Inspection and Maintenance of Large Reactors at Windscale Works and Calder Hall," T. TUOHY and G. D. IRELAND, A/CONF.15/P/1518.
303. "Operating Experience at Calder Hall," H. G. DAVEY, J. GAWTHROP, and T. N. MARSHAM, A/CONF.15/P/1522.
304. "The Development of the Hunterston Fuel Element," K. J. WOOTTON and W. E. DENNIS, A/CONF.15/P/1523.
305. "BEPO Wigner Energy Release," J. L. DICKSON, G. H. KINCHIN, R. F. JACKSON, W. M. LOMER, and J. H. W. SIMMONDS, A/CONF.15/P/1805.
306. "Evaluation of Finned Tubes for Steam Generators in Gas-Cooled Power Reactor Systems," F. CARR, C. ROUNTHWAITE, and P. A. YOUNG, A/CONF.15/P/2327.
307. "Safety Considerations for Gas-Cooled Thermal Reactors of the Calder Hall Type," F. R. FARMER, P. T. FLETCHER, and T. M. FRY, A/CONF.15/P/2331.
308. "Theory of Annealing Kinetics Applied to the Release of Stored Energy From Irradiated Graphite in Air-Cooled Reactors," A. H. COTTELL, J. C. BELL, G. B. GREENOUGH, W. M. LOMER, and J. H. W. SIMMONS, A/CONF.15/P/2485.
309. "The Measurement of Impurities in Helium. Part I. The Katharometer as a Continuous Analyser for Total Impurities," M. WILKENS and J. D. WILSON, Feb. 1959, AERE-C/R-2808.
310. "The Use of Induced N¹⁶ Activity in a Reactor Coolant for the Determination of Neutron Flux Distribution," A. GOODINGS, 1958, AERE-EL/M-105.
311. "Irradiation Damage and its Significance in Reactor Design," R. S. BARNES, Sec. 2 [of] Irradiation Damage, Papers Presented at the First Colloquium of the U.K.A.E.A. Diffraction Analysis Conference held at A.W.R.E. on April 2, 1958, AERE-M and C/R-2751 (Sect. 2).
312. "Stored Energy in BEPO Graphite," G. H. KINCHIN, Feb. 1958, AERE-R/M-169.
313. "Neutron Streaming in Gas-Cooled Reactors," I. S. GRANT, Oct. 1958, AERE-R/R-2568.
314. "Exponential Experiments with Natural and Slightly Enriched Uranium, Graphite Moderated Systems," D. J. LOCKEY, H. PEARSON, A. L. POPE, and G. M. WELLS, Sept. 1956, AERE-R/R-2597.
315. "Nuclear Reactor Physics for the Hyperboloid Reactor," D. A. NEWMARCH, April 1959, AERE-R/R-2753.
316. "Measurements of the Change in Cross Section of Irradiated Uranium Made by Modulating the Power of a Nuclear Reactor," (thesis), D. J. LITTLER, Nov. 6, 1956, AERE-RS/R-2092.
317. "Gas Flow in the Main Circuits and Reactor Channels," G. PACKMAN and G. H. INGLIS, Oct. 11, 1955, IGC-TRDC/P-116.
318. "Analysis of Load Deflection Test on the Supporting Feet of Calder Hall Pressure Vessel (Reactor 2)," W. H. IRVINE, Feb. 21, 1956, IGE-TM-01.
319. "Calder Works: The Radiological Design of the Discharge Plant," K. H. DENT and H. C. KNIGHTS, 1958, IGE-R-7.

320. "Monitor for Water Vapour in Carbon Dioxide," C. D. REID, K. W. PAGET, and J. W. HILL, Dec. 1957, IGR-R/CA-201(D).
321. "CO₂-Cooled Thermal Reactors—Pressure Vessels; Factors in the Selection of Steel," A. B. McINTOSH, Nov. 4, 1955, IGR-TM/C-013.
322. "Calculations on the Bowling Characteristics of Various Pippa Fuel Elements," R. O. HUGHES, April 6, 1956, Igr-TM/S-080.
323. "Deflection of Fins in Calder Reactors," W. B. HALL, May 16, 1956, IGR-TM/W-057.
324. "Possible Mechanisms of Failure of Carbon Dioxide-Cooled Reactor Pressure Vessels," T. J. HEAL, B. WATKINS, and A. COWAN, Oct. 1957, IGR-TN/C-632.
325. "Revised Lattice Constants for the Calder Hall Thermal Reactors," B. CUTTS, Oct. 1955, IGR-TN/R-254.
326. "Coolant Gags for the Calder Reactor," J. T. WILSON and M. LONG, March 28, 1956, IGR-TN/W-319.
327. "The Commissioning and Operational Experience of the Calder Hall Reactors," T. N. MARSHAM, Oct. 1957, IGT-R-20.
328. "Accident at Windscale No. 1 Pile on October 10, 1957," Nov. 1957, NP-6539.
329. "Stress Relieving on the Calder Heat Exchangers, Report No. 5256," Sept. 8, 1955, NP-6633.
330. "Conventions for Fluxes and Cross-Sections for Pippa Calculations," B. CUTTS, R. R. GALLIE, and P. W. MUMMERY, May 1956, NPCC/RPWP/P-17.
331. "A Survey of Lattice Calculations for Graphite Moderated Gas-Cooled Thermal Reactors," B. CUTTS (ed.), Sept. 1955, RDB(R)/TN-55.
332. Channel Ratings for the Calder Hall Reactors Including the Effect of Xenon Poisoning on the Radial Flux Distribution," J. M. B. LANG, Sept. 1, 1955, RDB(R)/TN-56.
337. "Nuclear Power and Gas Turbines-I," *Atomics* 7, 431-36, December 1956.
338. "World's Largest Atomic Power Station for Scotland," *Atomics* 8, 5-6, January 1957.
339. "Nuclear Power for Scotland, Details of the Hunterston, Ayrshire Reactor," *Atomics* 8, 93-6, March 1957.
340. "Coolant Gas Circuit and Steam Raising Plant," R. H. BIRD, *Atomics* 8, 245-50, July 1957.
341. "The Future for Nuclear Power," CHRISTOPHER HINTON, *Atomics* 8, 258-61, July 1957.
342. "Hinkley Point, Advanced Design of World's First 500,000 KW Nuclear Station," *Atomics* 8, 377-80, October 1957.
343. "Economics of Power Reactors," W. OSWALD, *Atomics* 8, 395-8, October 1957.
344. "Selecting the Reactor Coolant, Part II. Specific Materials," J. F. ABLITT, *Atomics and Nuclear Energy* 9, 355-9, October 1958.
345. "Reactor Fuel Life—The Reactivity Limits," P. J. GRANT, *Atomic World* 10, 132-5, April 1959.
346. "The Dragon H.T.G.R. Project. A Preliminary Design Study," *Atomic World* 10, 178-9, 185, May 1959.
347. "Building a Nuclear Power Station," *Special Suppl. to British Engineering International*, May 1958.
348. The Economics of Reactors of the Calder Hall Type, P. H. W. WOLFE, *Elec. J.* 160, 1729, June 6, 1958.
349. "Nuclear Power Stations in the United Kingdom," OSCAR JIMENEZ REYNALDO, *Energia nuclear (Madrid)* 2, No. 5, 44-53, Jan.-March 1958. (In Spanish.)
350. "The Role of Thorium in Nuclear Energy," D. J. O'CONNOR, *Energie nucléaire* 1, 5-12, Jan.-April 1959. (In French.)
351. "Heat Transfer," K. J. WOOTTON, *G.E.C. Atomic Energy Rev.* 1, 23-6, March 1957.
352. "Fuel Element Design," P. A. LINDLEY, *G.E.C. Atomic Energy Rev.* 1, 13-15, March 1957.
353. "Selection of Fuel-Canning Materials for Carbon-Dioxide-Cooled Nuclear Reactors," B. C. WOODFINE, *G.E.C. Atomic Energy Rev.* 1, 91-6, September 1957.
354. "A Fission-Product Detection System," J. B. B. MILLS, P. F. HARTSHORN, and R. D. BRYDEN, *G.E.C. Atomic Energy Rev.* 1, 97-104, September 1957.
355. "Heat Transfer Within Reactor Fuel Elements," R. G. S. SKIPPER, *G.E.C. Atomic Energy Rev.* 1, 141-6, March 1958.

Journal Articles

333. "The Methods of Using Fuels in a Nuclear Power Programme Including Some Notes on Reactor Types and Uses," J. V. DUNWORTH, *Acta Phys. Austriaca* 11, 452-70 (1958).
334. "The Economic Problems of Nuclear Power," J. A. JUKES, *Atom* No. 24, 17-28, October 1958.
335. "Nuclear Power Reactors—Some Inspection Problems," S. W. JOSLIN, *Atomics* 7, 233-6, 259, July 1956.
336. "Calder Hall—Mechanical Engineering Aspects," *Atomics* 7, 354-65, October 1956.

356. "Gas-Circulation Problems in Reactor Development," J. R. RIDLEY, *G.E.C. Atomic Energy Rev.* 1, 160-5, March 1958.
357. "The Specification of Fault Conditions for Gas-Cooled Graphite-Moderated Reactor Systems," T. J. O'NIELL, *G.E.C. Atomic Energy* 1, 166-73, March 1958.
358. "The High-Temperature Gas-Cooled Reactor," *G.E.C. Atomic Energy Rev.* 1, 115, September 1958.
359. "The Structure of the Graphite Moderator at Hunterston," I. A. B. GAUNT, *G.E.C. Atomic Energy Review* 1, 182-7, September 1958.
360. "The Determination of Thermal Stresses in Concrete Shields," I. W. HANNAH, *G.E.C. Atomic Energy Review* 1, 209-16, September 1958.
361. "Some Corrosion Problems in Gas-Cooled Reactors," M. W. DAVIES, *G.E.C. Atomic Energy Rev.* 2, 11-17, March 1959.
362. "Some Remarks of Nuclear Reactor Instrumentation," J. WALKER, *I.R.E. Trans. Nuclear Sci.* NS-3, No. 1, 21-5, February 1956.
363. "The Atmospheric Diffusion of Gases Discharged from the Chimney of the Harwell Reactor BEPO," N. G. STEWART, H. J. GALE, and R. N. CROOKS, *Intern. J. Air Pollution* 1, 87-102, October 1958.
364. "Some Problems in the Maintenance of Nuclear Reactors," H. G. DAVEY, *J. Brit. Nuclear Energy Conf.* 1, 173-88, October 1956.
365. "Heat Removal from Nuclear Power Reactors," JACK DIAMOND and W. B. HALL, *J. Brit. Nuclear Energy Conf.* 1, 227-40, July 1956.
366. "Symposium: Calder Works Nuclear Power Plant," *J. Brit. Nuclear Energy Conf.* 2, 41-232, April 1957.
367. "Symposium on Calder Works Nuclear Power Plant," *J. Brit. Nuclear Energy Conf.* 2, 233-91, July 1957.
368. "Future Engineering Development of the Gas-Cooled Nuclear Reactor," J. M. KAY, *J. Brit. Nuclear Energy Conf.* 3, 96-100, April 1958.
369. "Commissioning and Operation of 'A' Station, Calder Works," H. G. DAVEY, *J. Brit. Nuclear Energy Conf.* 3, 101-8, April 1958.
370. "Operational Experience at Calder Hall," K. L. STRETCH, *J. Brit. Nuclear Energy Conf.* 3, 109-15, April 1958.
371. "Temperature Transients in Gas-Cooled Thermal Nuclear Reactors," J. H. BOWEN and E. F. O. MASTERS, *J. Brit. Nuclear Energy Conf.* 3, 188-200, July 1958.
372. "Advanced Reactors," J. V. DUNWORTH, *J. Brit. Nuclear Energy Conf.* 4, 61-77, January 1959.
373. "The Nuclear Propulsion of Ships," CHRISTOPHER HINTON and R. V. MOORE, *J. Joint Panel Nuclear Marine Propulsion* 2, 1-25, April 1958.
374. "Flux Changes with Long-Term Irradiation in a Reactor of the Calder Hall Type," A. HITCHCOCK, V. E. PRICE, and Y. SHENTON, *J. Nuclear Energy* 6, 66-78 (1957).
375. "Metallurgical Aspects of Calder Hall," TOM BISHOP, *Metal Progress* 71, No. 6, 65-71, June 1957.
376. "Creep Tests on Materials for Gas-Cooled Reactors. New Research Equipment at G.E.C. Atomic Laboratories," *Metallurgia* 56, 313-14, December 1957.
377. "Nuclear Power and the Engineer," CHRISTOPHER HINTON, *Nature* 182, 1047-51, October 1958.
378. "The Future Development of Gas-Cooled Reactors," J. R. M. SOUTHWOOD, *New Scientist* 3, No. 71, 24-7, March 27, 1958.
379. "How They are Building Nuclear Britain," NIGEL CALDER, *New Scientist* 4, 413-14, 16, 18, July 17, 1958.
380. "Leak Detection in Nuclear Energy Heat Exchangers," P. H. CRUMLEY, *Nuclear Energy Engr.* 12, 448-9, December 1958.
381. "The Advanced Gas-Cooled Reactor," *Nuclear Energy Eng.* 13, 96-7, February 1959.
382. "Engineering Features of Calder Hall," *Nuclear Eng.* 1, 155, July 1956.
383. "Calder Hall," *Nuclear Eng.* 1, 266-303, October 1956.
384. "The World's Reactors. No. 6. Calder Hall," *Nuclear Eng.* 1, 388a, December 1956.
385. "Comparison of the Industrial Designs," *Nuclear Eng.* 2, 5-9, January 1957.
386. "Berkeley Nuclear Power Station," *Nuclear Eng.* 2, 96-102, March 1957.
387. "Site Organization at Calder Hall," J. F. VARGOE, *Nuclear Eng.* 2, 135-8, April 1957.
388. "The New Stations. A Detailed Comparison," *Nuclear Eng.* 2, 139, April 1957.
389. "Bradwell-On-Sea Nuclear Power Station. 1. Optimization of the N.P.P.C. Design. 2. General Description of the Bradwell Station," R. D. VAUGHAN, *Nuclear Eng.* 2, 140-5, April 1957.
390. "Instrumentation of Gas and Steam Circuits," *Nuclear Eng.* 2, 186-9, May 1957.
391. "Chemical Problems in Coolant Gases," W. E. DENNIS, *Nuclear Eng.* 2, 321-3, August 1957.
392. "Hinkley Point. Anticipated 500 MW Output," *Nuclear Eng.* 2, 423-6, October 1957.
393. "Windscale," *Nuclear Eng.* 2, 510-12, December 1957.

394. "Gas Monitoring Equipment for Berkeley Power Station," F. HARLEN, *Nuclear Eng. 3*, 158-62, April 1958.
395. "The World's Reactors. No. 19. Hinkley Point Power Station," *Nuclear Eng. 3*, 286-6a, July 1958.
396. "Nuclear Generation in the United Kingdom, The Programme and the Power Stations," *Nuclear Eng. 3*, 375-8, September 1958.
397. "Steam Cooling, The Vickers Nuclear Engineering Ship Propulsion Proposal," *Nuclear Eng. 4*, 246-9, June 1959.
398. "Chapelcross," *Nuclear Eng. 4*, 250-2, June 1959.
399. "Calder Hall—The First Nuclear Power Station in the World—How It Was Built—How It Works," *Nuclear Power 1*, 240-58, October 1956.
400. "Calder Hall," *Nuclear Power 1*, 276-94, November 1956.
401. "Some Engineering Problems of Nuclear Power Stations," CLAUDE D. GIBB, *Nuclear Power 2*, 85-90, March 1957.
402. "Data Processing for Burst Slug Systems," D. D. BOWEN, *Nuclear Eng. 2*, 91-5, March 1957.
403. "Some Characteristics of Reactor Gas Cooling Systems," P. FORTESCUE, *Nuclear Power 2*, 188-92, May 1957.
404. "Civil Engineering at Bradwell," A. HOUGHTON BROWN, *Nuclear Power 2*, 403-6, October 1957.
405. "Bradwell Reactor Vessels. Design, Fabrication and Erection Problems, R. F. BISHOP, *Nuclear Power 2*, 406-9, October 1957.
406. "Bradwell Nuclear Power Station," *Nuclear Power 2*, 410-13, October 1957.
407. "Nuclear Physics at Bradwell, Some Design Aspects," S. E. LEWIS, *Nuclear Power 2*, 414-15, October 1957.
408. "Bradwell Boilers. The Design Problems and Some Solutions," F. CARR, *Nuclear Power 2*, 416-19, October 1957.
409. "Extended Surface Boiler Tubing, Development of Integral Finned Steel Pipe for Bradwell and Other Power Stations," F. HOWARTH, *Nuclear Power 2*, 420-1, October 1957.
410. "Bradwell Turbo-Alternators, Auxiliary Turbine Sets and Gas Circulators," *Nuclear Power 2*, 422-3, October 1957.
411. "Layout and Services. Some Bradwell Considerations," T. ANDREWS, *Nuclear Power 2*, 424-5, October 1957.
412. "Boilermaking for Bradwell. Unorthodox Transport Arrangements," *Nuclear Power 2*, 426, October 1957.
413. "Special Lubricants for Bradwell," *Nuclear Power 2*, 427-8, October 1957.
414. "Bradwell, Industry's Contribution," *Nuclear Power 2*, 429-37, October 1957.
415. "Hinkley Point, The World's Biggest Nuclear Power Station," *Nuclear Power 2*, 450-4, November 1957.
416. "Power Station Instrumentation," J. C. NUTTER and J. DOWSING, *Nuclear Power 2*, 455-62, November 1957.
417. "Ships: Gas Cooled Reactor is the Choice," CHRISTOPHER HINTON and R. V. MOORE, *Nuclear Power 3*, 11-16, January 1958.
418. "The Calder Hall Graphite Structure," E. LONG, *Nuclear Power 3*, 58-63, February 1958.
419. "Harwell Wigner Release, First Since Windscale," *Nuclear Power 3*, 230, May 1958.
420. "Chapelcross Power Station," P. KNOWLES, *Nuclear Power 3*, 304-12, July 1958.
421. "Designing Pumps and Blowers for Nuclear Power Plants," R. A. STRUB, *Nuclear Power 3*, 592-4, December 1958.
422. "Some Design Considerations in Gas-Cooled Reactors," H. WEST and A. L. SHAW, *Nuclear Power 3*, 598-9, December 1958.
423. "Bellows Expansion Joints for Gas Ducting," *Nuclear Power 3*, 607-8, December 1958.
424. "Steps in Designing a Gas-Cooled Reactor," K. J. WOOTTON, *Nuclear Power 4*, 64-73, February 1959.
425. "Dragon, Details of the 10 MW HTGR," *Nuclear Power 4*, 102-3, April 1959.
426. "Dust Removal in Gas-Cooled Reactors," R. A. LE PAGE, *Nuclear Power 4*, 104-7, April 1959.
427. "Chapelcross: Commissioning the First Reactor," D. R. R. FAIR, *Nuclear Power 4*, 104-11, June 1959.
428. "Calder Hall Conference," *Nucleonics 14*, No. 12, S10-S32, December 1956.
429. "Gas-Cooled Reactor for the South of Scotland Electricity Board," P. J. GRANT, *Nucleonics 16*, No. 5, 108-13, May 1958.
430. "The Use of Electronic Digital Computers in a Study of High-Temperature Gas-Cooled Reactors," D. V. WORDSWORTH, *Reactor Technol. 1*, 25-34, April 1959.

Books

431. "Nuclear Reactor Control and Instrumentation, Nuclear Engineering Monographs," J. H. BOWEN and E. F. O. MASTERS, London, Temple Press, Ltd., 1959.
432. "Irradiation Facilities at Windscale Works," G. B. GREENOUGH and J. MOORE, Presented at the European Atomic Energy Society Symposium on Research Reactors and their Applications, Noordwijk, Netherlands, Sept. 10-14, 1957.

433. "Experimental Uses of the Medium Flux Reactors at Harwell," H. R. MCK. HYDER, Presented at the European Atomic Energy Society Symposium on Research Reactors and their Applications, Noordwijk, Netherlands, Sept. 10-14, 1957.
434. "Steam Cycles for Nuclear Power Plant, Nuclear Engineering Monographs," W. R. WOORTON, London, Temple Press, Ltd., 1958.

Patents

435. "Improvements in or Relating to Fuel Element Charging/Discharging Gear for Nuclear Reactors," E. LONG (to United Kingdom Atomic Energy Authority), British Patent 772,702, Apr. 17, 1957, *Nuclear Power* 2, 394, September 1957.
436. "Gas-Cooled Atomic Reactor Equipment" (to Allmanna Svenska Elektriska AB) [Sweden], British Patent 775,602, *Nuclear Eng.* 2, 452, October 1957.
437. "Improvements in or Relating to Nuclear Reactors," F. W. FENNING and R. F. JACKSON (to United Kingdom Atomic Energy Authority), British Patent 779,134, July 17, 1957, *Nuclear Power* 2, 394, September 1957.
438. "Improvements in or Relating to Nuclear Reactors," R. V. MOORE and G. PACKMAN (to United Kingdom Atomic Energy Authority), British Patent 779,408, July 17, 1957, *Nuclear Power* 2, 394, September 1957.
439. "Improvements in or Relating to Nuclear Reactors," R. V. MOORE, K. H. DENT, and J. H. BOWEN (to United Kingdom Atomic Energy Authority), British Patent 781,648, Aug. 21, 1957, *Nuclear Power* 2, 486, November 1957.
440. "Improvements in or Relating to Support Devices for Use in Nuclear Reactors," F. C. GREENHALGH and E. LONG (to United Kingdom Atomic Energy Authority), British Patent 783,094, July 28, 1954, *Nuclear Power* 3, 92, February 1958.
441. "Improvements in or Relating to Fuel Elements for Nuclear Reactors," K. H. DENT (to United Kingdom Atomic Energy Authority), British Patent 784,890, Aug. 25, 1955, *Nuclear Power* 3, 92, February 1958.
442. "Improvements in Nuclear Reactors Including Horizontal Graphite Bars" to Commissariat a l'Energie Atomique, Paris), British Patent 785,928, June 14, 1957, *Nuclear Power* 3, 44, January 1958.
443. "Improvements in or Relating to Pressure Vessels," R. N. MILLAR and K. J. MITCHELL (to the General Electric Co., Ltd.), British Patent 785,945, May 31, 1955, *Nuclear Power* 3, 92, February 1958.
444. "Improvements in Devices for Discharging Rods of Fissionable Material from Nuclear Reactors," SEBASTIEN FOGLIA (to Commissariat a l'Energie Atomique, Paris), British Patent 787,971, June 16, 1956, *Nuclear Power* 3, 92, February 1958.
445. "Improvements in or Relating to Nuclear Reactors" (to Stichting Reactor Centrum, Nederland), British Patent 788,284, July 16, 1956, *Nuclear Power* 3, 142, March 1958.
446. "Improvements in or Relating to Fuel Elements for Nuclear Reactors," W. B. HALL and T. DAVIS (to United Kingdom Atomic Energy Authority), British Patent 789,257, Nov. 5, 1955, *Nuclear Power* 3, 142, March 1958.
447. "Nuclear Reactor Arrangements," P. A. LINDLEY (to General Electric Co., Ltd.), British Patent 789,482, *Nuclear Eng.* 3, 360, August 1958.
448. "Heat Transfer Arrangements," D. A. BOYLAND (to General Electric Co., Ltd.), British Patent 790,688, *Nuclear Eng.* 3, 414, September 1958.
449. "Nuclear Reactors," W. E. DENNIS and P. A. LINDLEY (to General Electric Co., Ltd.), British Patent 791,011, *Nuclear Eng.* 3, 414, September 1958.
450. "Improvements in or Relating to Nuclear Reactors," J. LE FOLL and G. MELESE (to Commissariat a l'Energie Atomique), British Patent 792,171, March 19, 1958, *Nuclear Power* 3, 254, May 1958.
451. "Improvements in Nuclear Reactors and an Improved Method of Operating Nuclear Reactors" (to Babcock and Wilcox, Ltd.), British Patent 794,329, April 30, 1958, *Nuclear Power* 3, 348, July 1958.
452. "Retaining Devices for Fuel Elements in Gas-Cooled Nuclear Reactors," G. E. ROBINSON (to United Kingdom Atomic Energy Authority), British Patent 794,891, *Nuclear Eng.* 3, 556, December 1958.
453. "Improvements in or Relating to Retaining Devices for Fuel Elements in Gas-Cooled Nuclear Reactors" (to United Kingdom Atomic Energy Authority), British Patent 794,981, May 14, 1958, *Nuclear Power* 3, 348, July 1958.
454. "Improvements in or Relating to Nuclear Reactors," D. W. T. ANGWIN (to General Electric Co., Ltd.), British Patent 795,826, May 28, 1958, *Nuclear Power* 3, 348, July 1958.

455. "Improvements in or Relating to Nuclear Reactors," P. FORTESCUE and J. P. MIDDLETON (to United Kingdom Atomic Energy Authority), British Patent 799,212, Aug. 6, 1958, *Nuclear Power* 3, 524, October 1958.
456. "Nuclear Reactor," P. FORTESCUE and J. P. MIDDLETON (to U. K. Atomic Energy Authority), British Patent 799,212, *Nuclear Eng.* 4, 146, March 1959.
457. "Vapour Generators," W. R. WOOTTON (to Babcock and Wilcox, Ltd.), British Patent 803,382, *Nuclear Eng.* 4, 284, June 1959.
458. "Process for Recovering Gaseous Fission Products for Nuclear Reactor" (to Deutsche Gold-Und Silber-Scheideanstalt Vormals Roessler), British Patent 809,586, Feb. 25, 1959.
459. "Improvements in or Relating to Heat Transfer Arrangements," DONALD ARTHUR BOYLAND (to General Electric Co., Ltd.), British Patent 811,597, April 8, 1959.
460. "Improvements in or Relating to Heterogeneous Nuclear Reactors" (to Brown, Boveri & Cie, Aktiengesellschaft), British Patent 812,809, April 29, 1959.

UNITED NATIONS

Books

461. "Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Held in Geneva, August 8-20, 1955, Volume 2, Physics: Research Reactors," New York, United Nations, 1956.
462. "Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Held in Geneva, August 8-20, 1955, Volume 3, Power Reactors," New York, United Nations, 1955.
463. "Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Held in Geneva, August 8-20, 1955, Volume 5, Physics of Reactor Design," New York, United Nations, 1956.

YUGOSLAVIA

Reports

464. "Packed Bed Fuel Assembly for a Gas-Cooled Heterogeneous Reactor," MILORAD RISTIĆ and ZORAN ZARIĆ, A/CONF.15/P/492.

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Appendix A

LATTICE CALCULATIONS AND LIFETIME CONSIDERATIONS

The significance of the physics results can be made apparent only when they are integrated with the heat-transfer calculations and other design considerations, since the final choice of a lattice configuration reduces to a question of minimum power cost. Certain preliminary conclusions can be drawn, however, without considering the economics of the overall plant.

One important decision to be made in selecting core parameters is the choice of the diameter of the individual fuel rods in the cluster. It can be rather confidently predicted that the best rod size to use is the one that permits extracting the greatest amount of heat per unit volume of core. It is intuitively obvious that the smaller diameter fuel rods will permit the greater heat extraction per unit volume of fuel. The lattice reactivity calculations, however, show that larger rods permit a greater volume of fuel per unit volume of core. It appears that, from the point of view of power generation per unit volume of core, there exists a broad optimum range with respect to fuel rod diameter that has a maximum in the neighborhood of 0.75 inch. The multiplication factor vs. volume fraction of UO_2 (fuel volume/core volume) for 0.75-inch-diameter rods in 7-rod clusters is shown in Figure A.1. (Figures A.1 through A.12 appear on pp. 66-73.) The ratios of UO_2 volume fractions at constant k for the different rod diameters are nearly the same over quite a wide range of k . Table A.1 (see p. 000) lists in detail the values of η , ϵ , p , f , k , and R_e for the 7-rod cluster lattice of UO_2 fuel elements clad in stainless steel that were analyzed in the ORNL GCR-2 design study using the methods described in Section 2.1 of this report. The table is divided into two sections: (1) mod-

erator at room temperature, neutron temperature = 294°K and (2) moderator temperature = 750°F , neutron temperature = 814°K .

Loss of Neutrons by Leakage

In calculating the effect of neutron leakage on the multiplication factor, the standard relation

$$k_{eff} = \frac{k_{\infty} e^{-B^2 \tau}}{1 + L^2 B^2}$$

is used as a starting point. The presence of the voids in the cooling channels increases both the age and the diffusion length for the lattice. In the United States, the void correction factors applied to the age and diffusion length computations are based on the Behren's¹ streaming model. In the United Kingdom, the Grant streaming model appears to yield results in better agreement with their experiments.

The sensitivity of the leakage from the core to such factors as lattice pitch, fuel enrichment, cooling-channel diameter, and core size is shown in Figures A.2, A.3, and A.4. Figure A.2 shows the leakage as a function of core dimensions. The effects of fuel enrichment and lattice pitch are shown in Figure A.3. The leakage increases for large lattice spacings because of the increase in diffusion length associated with the lower concentration of fuel and cladding. The leakage increases for small lattice spacings because of the marked increase in fast leakage that results from crowding the cooling channels together. This effect is exhibited in Figure A.4, which presents the void correction factors for fast and slow neutrons.

¹ D. J. Behrens, *Proc. Phys. Soc. (London)* 62A, 607 (1949).

Reactivity Limitations on Fuel Exposure

It is helpful in discussing the problem of reactivity lifetime to recognize three distinct modes of fuel irradiation which will yield different average fuel lifetimes. Case 1 is the "batch loading" program in which all the fuel is charged to the reactor at the same time and is all discharged at the time when the reactivity of the fuel has fallen to the point where the chain reaction can no longer be maintained under the desired operating conditions. For this case the power density is assumed to be uniform throughout the reactor. Case 2 is the same as Case 1, except that the power distribution is not assumed to be uniform. In Case 3, the excess neutrons available with fresh fuel are used to extend the irradiation of old fuel that has a reactivity too low to sustain the reactor by itself. Fresh fuel is added to the core and the most highly irradiated fuel is removed, more or less continuously, in such a way that a constant equilibrium composition of fuel is maintained at all times in all parts of the reactor. This is the case referred to by Spinrad² as the case of uniformly graded fuel irradiation.

The results of the lifetime calculations are shown in Figures A.5 through A.12. A number of approximations have been made in these calculations, of which the following are the most important.

1. The slow neutron flux distribution within a cell is assumed to be unchanged during the irradiation; i.e., the cladding and moderator disadvantage factors were assumed to be constant. This assumption produces very little error; since only about 4 percent of the neutrons are absorbed in graphite, the moderator disadvantage factor varies slowly with fuel cross section, and the fuel cross section varies slowly with exposure (Figure A.9). The cladding disadvantage factor is always very close to unity.

2. The self-shielding of the 1-ev resonance

of Pu 240 has been neglected. For the long exposures reached for some of the lattices studied, the effective cross section of PU 240 could drop by a factor of 2 or more. While the effect of such a reduction in cross section could, in principle, be quite important, it is believed that the resulting errors in the present lifetime estimates are not great. The direction of this error is to cause an underestimate of the lifetime.

3. The effective cross sections have not been changed from one lattice to another. The lattices that are of the greatest interest have a larger average absorption cross section and, therefore, have relatively more epithermal neutron absorptions than in the case for which the effective cross sections have been calculated. The cross section most strongly affected is that of Pu 240, which has a greater relative importance for the more intermediate (epithermal) spectra. Thus, this effect tends to compensate for the neglect of self-shielding.

The immediate result of these calculations is the fuel lifetime corresponding to Case 1 (batch loading, uniform exposure). Values of the lifetime in Mwd/MT are given in Figure A.5 as functions of lattice pitch and fuel enrichment for clusters of seven 0.75-inch-diameter UO₂ rods, with 0.020-inch stainless steel cladding, at an average neutron temperature of 814° K (~541° C). For this fuel element, lifetimes of several thousand megawatt days per metric ton will certainly require enrichment to two or three times the U 235 concentration of natural uranium.

It is rather interesting to present the same results in other ways. Figure A.6 shows the burnup, in megawatt days per gram of U 235 contained in fresh fuel, as a function of the initial value of k_{eff} and the initial conversion ratio. Figure A.7 shows the same function for various enrichments of the fresh fuel.

The isotopic composition of the fuel as a function of exposure level is shown in Figure A.8 for a lattice very close to that of the EGCR design (8.2-inch pitch instead of 8 inches). It may be seen that the Pu 239 almost

²B. I. Spinrad *et al.*, "Reactivity Changes and Reactivity Lifetimes of Fixed-Fuel Elements in Thermal Reactors," *Proc. of Geneva Conference*, P/835.

reaches an equilibrium concentration, whereas the Pu 240 and Pu 241 continue to build up throughout the irradiation. Figure A.9 shows the variation of the neutron yield cross section,

$$\sum_i \nu_i N_i \sigma_{fi},$$

and of the fuel absorption cross section. Figure A.10 shows the variation of the conversion ratio (rate of production of Pu 239 and Pu 241 divided by the rate of consumption of U 235, Pu 239, and Pu 241). The rise in R_c reveals the importance of Pu 240 as a fertile material for long irradiations.

The variation of k_{eff} with exposure level depends upon the initial conversion ratio and on the fuel enrichment. For natural uranium feed, the reactivity increases at first (following Sm^{149} saturation), but the tendency with 2 percent enriched fuel is monotonically downward. Figure A.11 shows the dependence of k_{eff} on exposure for the same case as the previous figures (essentially the EGCR). Also shown is the function

$$\bar{k}(X) = \frac{1}{X} \int_0^X k(X') dX',$$

where X is the exposure in Mwd/MT. The exposure for which $\bar{k}(X) = 1$ is rather close to the lifetime which can be achieved by continuous fuel loading (Case 3) and illustrates the benefits to be gained by absorbing the excess neutrons from fresh fuel in highly irradiated fuel rather than in control rods. (Actually, the lifetime attainable by continuous loading is a few percent less than $\bar{k}$, as defined above.) It may be seen that continuous loading can extend the lifetime of the fuel by roughly a factor of 2, relative to Case 1. For an actual batch-loaded exposure (Case 2), Spinrad has estimated that the lifetime would be less than that of Case 1 by a factor roughly equal to the average-to-maximum power density ratio, for example, 0.5 to 0.7. Thus the benefits of continuous loading are substantial.^{2 3}

² W. B. Lewis, *Low Cost Fueling Without Recycle*, DR-39 (December, 1956).

In fact, the lifetime indicated for the design which embodies a lattice with an 8-inch pitch and 2 percent enrichment corresponds to a residence time of fuel in the reactor that is a large fraction of the probable 20-year life of the reactor. The Case 1 lifetime, Figure A.5, is about 7,350 Mwd/MT. For continuous fueling, this should be increased to 14,000 to 15,000 Mwd/MT. Since the specific power is about 5 Mwd/MT, this would yield an average residence time of about 10½ years (load factor of 0.80). Since the devices that would permit continuous onstream loading of the reactor pose quite serious engineering problems, it is desirable to investigate the possibility of approximating the continuous loading cycle by reloading a rather substantial fraction of the total number of fuel channels at a time, at appropriate intervals. Figure A.12 shows fuel lifetime (in Mwd/MT) as a function of the number of intervals into which the loading cycle might be divided. For example, for $N = 10$, the 10 percent of the fuel elements that had received the greatest exposure would be removed from the core at one time and replaced with fresh fuel. Thus, for an average residence time of 10 years, it would be necessary to replace 10 percent of the core at roughly 1-year intervals. After each refueling operation, the core would contain equal numbers of fuel elements that had been in the core for 9, 8, 7, 2, 1, and 0 years, with each age group being uniformly distributed through the core. It may be seen from Figure A.12 that 10 steps will give nearly as long a fuel exposure as continuous loading. The effective multiplication factor of the core after each refueling operation, which indicates the amount of reactivity the control rods must take care of with each fresh fuel charge, is also shown in Figure A.12. It is recognized that with such long residence times as those mentioned above, an equilibrium fueling cycle might never be established in the lifetime of the reactor.

FIGURE A.1.—Multiplication factor vs. UO_2 volume fraction at a fuel enrichment of 2%.

FIGURE A.2.—Neutron leakage vs. core height and diameter.

FIGURE A.3.—Neutron leakage vs. lattice pitch and enrichment.

FIGURE A.4.—Effect of cooling channel size on Fermi age and diffusion length.

FIGURE A.5.—Fuel lifetime vs. lattice pitch and enrichment.

FIGURE A.6.—Fuel lifetime vs. multiplication factor and conversion ratio.

UNCLASSIFIED
ORNL-LR-DWG 27536R

FIGURE A.7.—Fuel lifetime vs. multiplication factor and enrichment.

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ORNL-LR-DWG 27539R

FIGURE A.8.—Isotopic composition of fuel vs. exposure.

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ORNL-LR-DWG 27538R

FIGURE A 9 —Neutron yield, fuel absorption, and fission cross sections vs. exposure.

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ORNL-LR-DWG 27540R

FIGURE A.10.—Conversion ratio vs. exposure.

FIGURE A.11.—Multiplication factor vs. exposure.

FIGURE A.12.—Effect of staggered loading of fuel lifetime.

TABLE A 1.—RESULTS OF CALCULATIONS OF K_{∞} AND CONVERSION RATIO FOR SEVEN-ROD CLUSTERS OF UO_2 FUEL RODS CLAD WITH STAINLESS STEEL

Case No.	Rod diameter (in.)	Cladding thickness (in.)	Enrichment (at %)	Cooling hole diameter (in.)	Lattice pitch (in.)	η	ϵ	p	f	k_{∞}	R_{∞}
Moderator at Room Temperature, Neutron Temperature=294°K											
181 ----	0 50	0 000	0 714	3 25	5 0	1 339	1 016	0 6126	0 9692	0 808	1 358
182 ---	0 50	0 000	0 714	3 25	6 0	1 339	1 016	0 7413	0 9442	0 952	1 092
183 ---	0 50	0 000	0 714	3 25	7 0	1 339	1 016	0 8147	0 9195	1 019	0 940
184 -	0 50	0 000	0 714	3 25	8 0	1 339	1 016	0 8605	0 8945	1 047	0 845
185 ----	0 50	0 000	0 80	3 25	5 0	1 392	1 016	0 6126	0 9709	0 841	1 298
186 ----	0 50	0 000	0 80	3 25	6 0	1 392	1 016	0 7413	0 9470	0 993	1 032
187 ----	0 50	0 000	0 80	3 25	7 0	1 392	1 016	0 8147	0 9232	1 064	0 880
188 - ----	0 50	0 000	0 80	3 25	8 0	1 392	1 016	0 8605	0 8994	1 095	0 785
189 ----	0 50	0 020	1 25	3 25	5 0	1 580	1 016	0 6046	0 8603	0 835	1 118
190 ---	0 50	0 020	1 25	3 25	6 0	1 580	1 016	0 7349	0 8453	0 997	0 852
191 ---	0 50	0 020	1 25	3 25	7 0	1 580	1 016	0 8096	0 8294	1 078	0 700
192 ---	0 50	0 020	1 25	3 25	8 0	1 580	1 016	0 8565	0 8120	1 116	0 605
276 ----	0 50	0 020	1 25	2 50	5 0	1 580	1 016	0 6749	0 8509	0 922	0 970
277 ----	0 50	0 020	1 25	2 50	6 0	1 580	1 016	0 7734	0 8342	1 035	0 771
278 ----	0 50	0 020	1 25	2 50	7 0	1 580	1 016	0 8334	0 8160	1 091	0 650
279 ---	0 50	0 020	1 25	2 50	8 0	1 580	1 016	0 8724	0 7960	1 114	0 572
280 ----	0 50	0 020	1 25	2 50	9 0	1 580	1 016	0 8992	0 7754	1 119	0 518
281 ---	0 50	0 020	1 25	2 50	10 0	1 580	1 016	0 9183	0 7532	1 110	0 480
282 ----	0 50	0 020	1 50	2 50	5 0	1 646	1 016	0 6749	0 8665	0 978	0 916
283 ----	0 50	0 020	1 50	2 50	6 0	1 646	1 016	0 7734	0 8500	1 099	0 717
284 ----	0 50	0 020	1 50	2 50	7 0	1 646	1 016	0 8334	0 8323	1 160	0 596
285 ----	0 50	0 020	1 50	2 50	8 0	1 646	1 016	0 8724	0 8135	1 187	0 518
286 ----	0 50	0 020	1 50	2 50	9 0	1 646	1 016	0 8992	0 7932	1 192	0 464
287 ----	0 50	0 020	1 50	2 50	10 0	1 646	1 016	0 9183	0 7723	1 186	0 426
288 ----	0 50	0 020	2 00	2 50	5 0	1 737	1 016	0 6749	0 8891	1 059	0 850
289 - -	0 50	0 020	2 00	2 50	6 0	1 737	1 016	0 7734	0 8740	1 193	0 651
290 ----	0 50	0 020	2 00	2 50	7 0	1 737	1 016	0 8334	0 8573	1 261	0 530
291 - -	0 50	0 020	2 00	2 50	8 0	1 737	1 016	0 8724	0 8395	1 293	0 452
292 ---	0 50	0 020	2 00	2 50	9 0	1 737	1 016	0 8992	0 8202	1 302	0 398
293 --	0 50	0 020	2 00	2 50	10 0	1 737	1 016	0 9183	0 8003	1 297	0 360
153 - -	0 75	0 000	0 714	3 25	6 0	1 339	1 024	0 6064	0 9716	0 808	1 371
154 ---	0 75	0 000	0 714	3 25	7 0	1 339	1 024	0 7077	0 9500	0 926	1 161
155 ----	0 75	0 000	0 714	3 25	8 0	1 339	1 024	0 7755	0 9360	0 995	1 021
156 ----	0 75	0 000	0 714	3 25	9 0	1 339	1 024	0 8222	0 9186	1 036	0 924
157 ---	0 75	0 000	0 80	3 25	6 0	1 392	1 024	0 6064	0 9728	0 841	1 311
158 --	0 75	0 000	0 80	3 25	7 0	1 392	1 024	0 7077	0 9556	0 964	1 101
159 ---	0 75	0 000	0 80	3 25	8 0	1 392	1 024	0 7755	0 9386	1 038	0 961
160 ----	0 75	0 000	0 80	3 25	9 0	1 392	1 024	0 8222	0 9216	1 080	0 864
161 ----	0 75	0 000	1 25	3 25	6 0	1 580	1 024	0 6064	0 9774	0 959	1 131
162 ----	0 75	0 000	1 25	3 25	7 0	1 580	1 024	0 7077	0 9642	1 104	0 921
163 -	0 75	0 000	1 25	3 25	8 0	1 580	1 024	0 7755	0 9500	1 192	0 781
164 -	0 75	0 000	1 25	3 25	9 0	1 580	1 024	0 8222	0 9325	1 241	0 684
165 ---	0 75	0 000	1 50	3 25	6 0	1 646	1 024	0 6064	0 9790	1 001	1 077
166 ----	0 75	0 000	1 50	3 25	7 0	1 646	1 024	0 7077	0 9675	1 154	0 867
167 ----	0 75	0 000	1 50	3 25	8 0	1 646	1 024	0 7755	0 9550	1 248	0 727
168 ----	0 75	0 000	1 50	3 25	9 0	1 646	1 024	0 8222	0 9385	1 301	0 630
211 ----	0 75	0 020	1 00	3 25	6 0	1 491	1 024	0 5995	0 8780	0 804	1 211
212 ---	0 75	0 020	1 00	3 25	7 0	1 491	1 024	0 7021	0 8650	0 927	1 001
213 ---	0 75	0 020	1 00	3 25	8 0	1 491	1 024	0 7704	0 8517	1 002	0 861
214 - -	0 75	0 020	1 00	3 25	9 0	1 491	1 024	0 8179	0 8372	1 046	0 764
215 ----	0 75	0 020	1 00	3 25	10 0	1 491	1 024	0 8522	0 8206	1 068	0 693
216 ----	0 75	0 020	1 00	3 25	11 0	1 491	1 024	0 8777	0 8000	1 072	0 648
169 ----	0 75	0 020	1 25	3 25	6 0	1 580	1 024	0 5995	0 8947	0 868	1 131
170 ---	0 75	0 020	1 25	3 25	7 0	1 580	1 024	0 7021	0 8836	1 004	0 921
171 ----	0 75	0 020	1 25	3 25	8 0	1 580	1 024	0 7704	0 8705	1 085	0 781
172 ----	0 75	0 020	1 25	3 25	9 0	1 580	1 024	0 8179	0 8551	1 132	0 684
209a ---	0 75	0 020	1 25	3 25	10 0	1 580	1 024	0 8522	0 8370	1 154	0 613
209b --	0 75	0 020	1 25	3 25	11 0	1 580	1 024	0 8777	0 8175	1 161	0 568
173 ----	0 75	0 020	1 50	3 25	6 0	1 646	1 024	0 5995	0 9053	0 915	1 077
174 ----	0 75	0 020	1 50	3 25	7 0	1 646	1 024	0 7021	0 8942	1 058	0 867
175 ----	0 75	0 020	1 50	3 25	8 0	1 646	1 024	0 7704	0 8820	1 145	0 727

TABLE A.1.—RESULTS OF CALCULATIONS OF K_{∞} AND CONVERSION RATIO FOR SEVEN-ROD CLUSTERS OF UO_2 FUEL RODS CLAD WITH STAINLESS STEEL—Continued

Case No	Rod diameter (in)	Cladding thickness (in)	Enrichment (at %)	Cooling hole diameter (in)	Lattice pitch (in)	η	ϵ	p		k_{∞}	R_o
Moderator at Room Temperature, Neutron Temperature=294°K—Continued											
176-----	0 75	0 020	1 50	3 25	9 0	1 646	1 024	0 8179	0 8675	1 196	0 630
210a-----	0 75	0 020	1 50	3 25	10 0	1 646	1 024	0 8522	0 8500	1 221	0 559
210b-----	0 75	0 020	1 50	3 25	11 0	1 646	1 024	0 8777	0 8308	1 229	0 514
270-----	0 75	0 020	2 00	3 25	6 0	1 737	1 024	0 5995	0 9203	0 981	1 011
271-----	0 75	0 020	2 00	3 25	7 0	1 737	1 024	0 7021	0 9091	1 135	0 801
272-----	0 75	0 020	2 00	3 25	8 0	1 737	1 024	0 7704	0 8960	1 228	0 661
273-----	0 75	0 020	2 00	3 25	9 0	1 737	1 024	0 8179	0 8817	1 283	0 564
274-----	0 75	0 020	2 00	3 25	10 0	1 737	1 024	0 8522	0 8662	1 313	0 493
275-----	0 75	0 020	2 00	3 25	11 0	1 737	1 024	0 8777	0 8497	1 326	0 448
294-----	1 00	0 020	1 25	4 00	8 0	1 580	1 028	0 6428	0 9039	0 944	1 055
295-----	1 00	0 020	1 25	4 00	9 0	1 580	1 028	0 7147	0 8915	1 035	0 907
296-----	1 00	0 020	1 25	4 00	10 0	1 580	1 028	0 7674	0 8776	1 094	0 797
297-----	1 00	0 020	1 25	4 00	11 0	1 580	1 028	0 8068	0 8625	1 130	0 716
298-----	1 00	0 020	1 25	4 00	12 0	1 580	1 028	0 8371	0 8465	1 151	0 653
299-----	1 00	0 020	1 25	4 00	13 0	1 580	1 028	0 8609	0 8300	1 160	0 604
300-----	1 00	0 020	1 50	4 00	8 0	1 646	1 028	0 6428	0 9125	0 992	1 001
301-----	1 00	0 020	1 50	4 00	9 0	1 646	1 028	0 7147	0 9012	1 090	0 853
302-----	1 00	0 020	1 50	4 00	10 0	1 646	1 028	0 7674	0 8882	1 153	0 743
303-----	1 00	0 020	1 50	4 00	11 0	1 646	1 028	0 8068	0 8737	1 192	0 662
304-----	1 00	0 020	1 50	4 00	12 0	1 646	1 028	0 8371	0 8577	1 215	0 599
305-----	1 00	0 020	1 50	4 00	13 0	1 646	1 028	0 8609	0 8404	1 224	0 556
306-----	1 00	0 020	2 00	4 00	8 0	1 737	1 028	0 6428	0 9252	1 062	0 935
307-----	1 00	0 020	2 00	4 00	9 0	1 737	1 028	0 7147	0 9142	1 167	0 787
308-----	1 00	0 020	2 00	4 00	10 0	1 737	1 028	0 7674	0 9020	1 236	0 677
309-----	1 00	0 020	2 00	4 00	11 0	1 737	1 028	0 8068	0 8885	1 280	0 596
310-----	1 00	0 020	2 00	4 00	12 0	1 737	1 028	0 8371	0 8732	1 306	0 533
311-----	1 00	0 020	2 00	4 00	13 0	1 737	1 028	0 8609	0 8570	1 318	0 484

Moderator Temperature=750° F, Neutron Temperature=814° K

342-----	0 50	0 018	1 25	2 50	5 0	1 539	1 016	0 6531	0 8639	0 882	1 064
343-----	0 50	0 018	1 25	2 50	6 0	1 539	1 016	0 7570	0 8502	1 006	0 849
344-----	0 50	0 018	1 25	2 50	7 0	1 539	1 016	0 8208	0 8356	1 072	0 717
345-----	0 50	0 018	1 25	2 50	8 0	1 539	1 016	0 8625	0 8200	1 106	0 630
346-----	0 50	0 018	1 25	2 50	9 0	1 539	1 016	0 8912	0 8028	1 119	0 571
347-----	0 50	0 018	1 25	2 50	10 0	1 539	1 016	0 9118	0 7847	1 119	0 529
348-----	0 50	0 018	1 50	2 50	5 0	1 609	1 016	0 6531	0 8783	0 938	1 006
349-----	0 50	0 018	1 50	2 50	6 0	1 609	1 016	0 7570	0 8655	1 071	0 791
350-----	0 50	0 018	1 50	2 50	7 0	1 609	1 016	0 8208	0 8518	1 143	0 659
351-----	0 50	0 018	1 50	2 50	8 0	1 609	1 016	0 8625	0 8370	1 180	0 572
352-----	0 50	0 018	1 50	2 50	9 0	1 609	1 016	0 8912	0 8204	1 195	0 513
353-----	0 50	0 018	1 50	2 50	10 0	1 609	1 016	0 9118	0 8025	1 196	0 471
354-----	0 50	0 018	2 00	2 50	5 0	1 706	1 016	0 6531	0 8994	1 018	0 933
355-----	0 50	0 018	2 00	2 50	6 0	1 706	1 016	0 7570	0 8875	1 165	0 718
356-----	0 50	0 018	2 00	2 50	7 0	1 706	1 016	0 8208	0 8746	1 244	0 586
357-----	0 50	0 018	2 00	2 50	8 0	1 706	1 016	0 8625	0 8610	1 287	0 499
358-----	0 50	0 018	2 00	2 50	9 0	1 706	1 016	0 8912	0 8456	1 306	0 440
359-----	0 50	0 018	2 00	2 50	10 0	1 706	1 016	0 9118	0 8289	1 310	0 398
384-----	0 50	0 018	2 50	2 50	5 0	1 770	1 016	0 6531	0 9140	1 074	0 889
385-----	0 50	0 018	2 50	2 50	6 0	1 770	1 016	0 7570	0 9033	1 230	0 674
386-----	0 50	0 018	2 50	2 50	7 0	1 770	1 016	0 8208	0 8914	1 316	0 542
387-----	0 50	0 018	2 50	2 50	8 0	1 770	1 016	0 8625	0 8780	1 362	0 455
388-----	0 50	0 018	2 50	2 50	9 0	1 770	1 016	0 8912	0 8635	1 384	0 396
389-----	0 50	0 018	2 50	2 50	10 0	1 770	1 016	0 9118	0 8478	1 390	0 354
248-----	0 75	0 018	1 20	3 25	6 0	1 522	1 024	0 5657	0 9037	0 797	1 260
249-----	0 75	0 018	1 20	3 25	7 0	1 522	1 024	0 6744	0 8936	0 940	1 035
250-----	0 75	0 018	1 20	3 25	8 0	1 522	1 024	0 7486	0 8837	1 031	0 881
251-----	0 75	0 018	1 20	3 25	9 0	1 522	1 024	0 8001	0 8723	1 088	0 775
252-----	0 75	0 018	1 20	3 25	10 0	1 522	1 024	0 8377	0 8591	1 122	0 697
253-----	0 75	0 018	1 20	3 25	11 0	1 522	1 024	0 8655	0 8438	1 138	0 639

TABLE A.1.—RESULTS OF CALCULATIONS OF K_{∞} AND CONVERSION RATIO FOR SEVEN-ROD CLUSTERS OF UO_2 FUEL RODS CLAD WITH STAINLESS STEEL—Continued

Case No	Rod diameter (in.)	Cladding thickness (in.)	Enrichment (at %)	Cooling hole diameter (in.)	Lattice pitch (in.)	η	ϵ	p	f	k_{∞}	R_{∞}
Moderator Temperature=750° F, Neutron Temperature=814° K—Continued											
230	0.75	0.018	1.25	3.25	6.0	1.539	1.024	0.5657	0.9059	0.808	1.245
231	0.75	0.018	1.25	3.25	7.0	1.539	1.024	0.6744	0.8961	0.953	1.020
232	0.75	0.018	1.25	3.25	8.0	1.539	1.024	0.7486	0.8858	1.045	0.866
233	0.75	0.018	1.25	3.25	9.0	1.539	1.024	0.8001	0.8748	1.103	0.760
234	0.75	0.018	1.25	3.25	10.0	1.539	1.024	0.8377	0.8622	1.138	0.682
235	0.75	0.018	1.25	3.25	11.0	1.539	1.024	0.8655	0.8466	1.155	0.624
236	0.75	0.018	1.375	3.25	6.0	1.576	1.024	0.5657	0.9109	0.832	1.214
237	0.75	0.018	1.375	3.25	7.0	1.576	1.024	0.6744	0.9016	0.982	0.989
238	0.75	0.018	1.375	3.25	8.0	1.576	1.024	0.7486	0.8913	1.077	0.835
239	0.75	0.018	1.375	3.25	9.0	1.576	1.024	0.8001	0.8797	1.135	0.729
240	0.75	0.018	1.375	3.25	10.0	1.576	1.024	0.8377	0.8667	1.172	0.651
241	0.75	0.018	1.375	3.25	11.0	1.576	1.024	0.8655	0.8530	1.192	0.593
242	0.75	0.018	1.50	3.25	6.0	1.609	1.024	0.5657	0.9160	0.854	1.187
243	0.75	0.018	1.50	3.25	7.0	1.609	1.024	0.6744	0.9067	1.008	0.962
244	0.75	0.018	1.50	3.25	8.0	1.609	1.024	0.7486	0.8972	1.107	0.808
245	0.75	0.018	1.50	3.25	9.0	1.609	1.024	0.8001	0.8866	1.169	0.702
246	0.75	0.018	1.50	3.25	10.0	1.609	1.024	0.8377	0.8744	1.207	0.624
247	0.75	0.018	1.50	3.25	11.0	1.609	1.024	0.8655	0.8595	1.226	0.566
336	0.75	0.018	2.00	3.25	6.0	1.706	1.024	0.5657	0.9290	0.918	1.114
337	0.75	0.018	2.00	3.25	7.0	1.706	1.024	0.6744	0.9205	1.084	0.889
338	0.75	0.018	2.00	3.25	8.0	1.706	1.024	0.7486	0.9115	1.192	0.735
339	0.75	0.018	2.00	3.25	9.0	1.706	1.024	0.8001	0.9015	1.261	0.629
340	0.75	0.018	2.00	3.25	10.0	1.706	1.024	0.8377	0.8900	1.302	0.551
341	0.75	0.018	2.00	3.25	11.0	1.706	1.024	0.8655	0.8771	1.326	0.493
378	0.75	0.018	2.00	3.25	6.0	1.770	1.024	0.5657	0.9391	0.963	1.070
379	0.75	0.018	2.00	3.25	7.0	1.770	1.024	0.6744	0.9312	1.138	0.845
380	0.75	0.018	2.00	3.25	8.0	1.770	1.024	0.7486	0.9226	1.251	0.691
381	0.75	0.018	2.00	3.25	9.0	1.770	1.024	0.8001	0.9133	1.325	0.585
382	0.75	0.018	2.00	3.25	10.0	1.770	1.024	0.8377	0.9025	1.370	0.507
383	0.75	0.018	2.00	3.25	11.0	1.770	1.024	0.8655	0.8901	1.396	0.449
360	1.00	0.018	1.25	4.00	8.0	1.539	1.028	0.6091	0.9174	0.884	1.155
361	1.00	0.018	1.25	4.00	9.0	1.539	1.028	0.6861	0.9087	0.986	0.996
362	1.00	0.018	1.25	4.00	10.0	1.539	1.028	0.7430	0.8993	1.057	0.878
363	1.00	0.018	1.25	4.00	11.0	1.539	1.028	0.7860	0.8890	1.106	0.789
364	1.00	0.018	1.25	4.00	12.0	1.539	1.028	0.8192	0.8770	1.137	0.720
365	1.00	0.018	1.25	4.00	13.0	1.539	1.028	0.8454	0.8634	1.155	0.666
366	1.00	0.018	1.50	4.00	8.0	1.609	1.028	0.6091	0.9255	0.932	1.097
367	1.00	0.018	1.50	4.00	9.0	1.609	1.028	0.6861	0.9170	1.041	0.938
368	1.00	0.018	1.50	4.00	10.0	1.609	1.028	0.7430	0.9082	1.116	0.820
369	1.00	0.018	1.50	4.00	11.0	1.609	1.028	0.7860	0.8980	1.168	0.731
370	1.00	0.018	1.50	4.00	12.0	1.609	1.028	0.8192	0.8866	1.201	0.662
371	1.00	0.018	1.50	4.00	13.0	1.609	1.028	0.8454	0.8731	1.221	0.608
372	1.00	0.018	2.00	4.00	8.0	1.706	1.028	0.6091	0.9369	1.001	1.024
373	1.00	0.018	2.00	4.00	9.0	1.706	1.028	0.6861	0.9288	1.118	0.865
374	1.00	0.018	2.00	4.00	10.0	1.706	1.028	0.7430	0.9200	1.199	0.747
375	1.00	0.018	2.00	4.00	11.0	1.706	1.028	0.7860	0.9100	1.254	0.658
376	1.00	0.018	2.00	4.00	12.0	1.706	1.028	0.8192	0.8991	1.292	0.589
377	1.00	0.018	2.00	4.00	13.0	1.706	1.028	0.8454	0.8867	1.315	0.535
390	1.00	0.018	2.50	4.00	8.0	1.770	1.028	0.6091	0.9439	1.046	0.980
391	1.00	0.018	2.50	4.00	9.0	1.770	1.028	0.6861	0.9361	1.169	0.821
392	1.00	0.018	2.50	4.00	10.0	1.770	1.028	0.7430	0.9275	1.254	0.703
393	1.00	0.018	2.50	4.00	11.0	1.770	1.028	0.7860	0.9178	1.313	0.614
394	1.00	0.018	2.50	4.00	12.0	1.770	1.028	0.8192	0.9066	1.351	0.545
395	1.00	0.018	2.50	4.00	13.0	1.770	1.028	0.8454	0.8930	1.374	0.491

Appendix B

DETAILED TABLES OF REACTOR CHARACTERISTICS

The principal characteristics of reactors that are typical of the slightly enriched gas-cooled reactors, the heavy-water-moderated reactors, and the semihomogeneous graphite-moderated reactors are presented in the tables below:

- B.1.—Characteristics of Oak Ridge National Laboratory Gas-Cooled Reactor Design (GCR-2).
- B.2.—Characteristics of Kaiser Engineers—Allis-Chalmers Gas-Cooled Reactor Design (GCPR).
- B.3.—Characteristics of Kaiser Engineers—Allis-Chalmers Experimental Gas-Cooled Reactor (EGCR).
- B.4.—Characteristics of British Advanced Gas-Cooled Reactor (AGR) Windscale Prototype.
- B.5.—Characteristics of British Full-Scale Advanced Gas-Cooled Reactor (AGR) Design.
- B.6.—Characteristics of General Nuclear Engineering Corporation Gas-Cooled Reactor Prototype Design.
- B.7.—Characteristics of General Nuclear Engineering Corporation Gas-Cooled Reactor Large-Scale Design.
- B.8.—Characteristics of the General Atomic Gas-Cooled Reactor Prototype Design.
- B.9.—Characteristics of the General Atomic Large-Scale Gas-Cooled Reactor Design.

TABLE B.1.—CHARACTERISTICS OF OAK RIDGE NATIONAL LABORATORY GAS-COOLED REACTOR DESIGN (GCR-2)

General:		
Thermal power	-----	687 Mw.
Electrical power:		
Gross	-----	252 Mw.
Net	-----	225 Mw.
Power cycle	-----	Steam cycle, single pressure, once-through.
Steam conditions	-----	950° F, 950 psia.
Turbine heat rate	-----	9,458 Btu/kwhr.
Efficiency:		
Gross	-----	36.7%.
Net	-----	32.8%.
Active core dimensions		30 ft dia x 20 ft high.
Reflector thickness	---	2.5 ft radial, 2.5 ft axial.

Moderator:		
Material	-----	Graphite.
Inventory	-----	2.24 x 10 ⁶ lb, including reflector.
Coolant:		
Material	-----	Helium.
Pressure	-----	300 psia.
Temperature:		
Inlet	-----	460° F.
Outlet	-----	1,000° F.
Fuel and physics data:		
Fuel composition	----	UO ₂ .
Fuel enrichment	----	2% U 235.
Fuel geometry	-----	Seven UO ₂ rods, 0.75 in. OD x 0.32 in. ID.
Fuel cladding	-----	0.020-in.-thick type 304 stainless steel.
Lattice pitch	-----	8.0 in. square.
Ratio of moderator volume to fuel volume.		18.1.
Maximum fuel surface temperature:		
Design	-----	1,200° F.
Absolute	-----	~1,500° F.
Specific power of fuel.		5.01 Mw/MT of uranium, 250 kw/kg of U ²³⁵ .
Average specific power of fuel.		3.28 kw/ft of capsule, 11,190 Btu/hr·ft.
Fuel surface heat flux:		
Average	-----	53,430 Btu/hr·ft ² , 17 w/cm ² .
Maximum	-----	106,000 Btu/hr·ft ² , 33 w/cm ² .
Average core power density.		1.72 kw/liter, 48.6 kw/ft ³ .
Axial peak-to-average power ratio.		1.32.
Gross radial peak-to-average power ratio.		1.50.
Fuel management	----	Staggered fuel loading, effectively continuous.
Fuel exposure:		
Average	-----	7,500 Mwd/MT, about 0.4 fissions per initial fissile atom over 20-year reactor life.
Maximum	-----	15,000 Mwd/MT, 0.8 fissions per initial fissile atom.

Fuel exposure—Continued	
Maximum excess k, δk_{eff} .	0.10 (cold, clean).
Maximum operating δk_{eff} .	0.04.
Control:	
Rod material	Silver.
Rod geometry	Cylindrical tubes, 2 in. OD x 1.25 in. ID.
Number of rods	61.
Pressure vessel:	
Shape	Sphere.
Dimensions	50 ft OD x 3.25 in. thick.
Material	Type SA-212 steel, grade B.
Containment vessel	None.

TABLE B.2.—CHARACTERISTICS OF KAISER ENGINEERS-ALLIS-CHALMERS GAS-COOLED REACTOR DESIGN (GCPR)

General:	
Thermal power	600 Mw.
Electrical power:	
Gross	247 Mw.
Net	215 Mw.
Steam conditions	1,000° F, 1,300 psia.
Efficiency:	
Gross	41.2%.
Net	35.8%.
Moderator:	
Material	Graphite.
Inventory	1.66 x 10 ⁶ lb, including reflector.
Coolant:	
Material	Helium.
Pressure	315 psia.
Temperature:	
Inlet	510° F.
Outlet	1,050° F.
Fuel and physics data:	
Fuel composition	UO ₂ .
Fuel enrichment	2% U 235.
Fuel geometry	Seven UO ₂ rods, 0.705 in. OD x 0.323 in. ID.
Fuel cladding	0.020-in.-thick type 304 stainless steel.
Lattice pitch	8.6 in. square.
Maximum fuel surface temperature:	
Design	1,300° F.
Absolute	~1,600° F.
Specific power of fuel	7.55 Mw/MT of uranium, 378 kw/kg of U 235.
Fuel management	Continuous loading.

Fuel and physics data—Continued	
Average fuel exposure	10,000 Mwd/MT, 0.53 fissions per initial fissile atom.
Control rods:	
Material	B ₄ C, stainless-steel jacketed.
Geometry	Cylindrical tubes.
Number	106.
Pressure vessel:	
Shape	Cylinder.
Dimensions	33 ft ID x 64 ft high x 4.0 in. thick.
Material	Carbon steel.

TABLE B.3.—CHARACTERISTICS OF KAISER ENGINEERS-ALLIS-CHALMERS EXPERIMENTAL GAS-COOLED REACTOR (EGCR)

General:	
Thermal power	84.3 Mw.
Electrical power:	
Gross	29.4 Mw.
Net	24.1 Mw.
Power cycle	Steam, single pressure, no reheat.
Steam conditions	902° F, 1,300 psia.
Efficiency:	
Gross	34.9%.
Net	28.6%.
Active core dimensions	11.84 ft dia x 14.5 ft high.
Reflector thickness	2.0 ft radial, 2.5 ft axial.
Moderator material	Graphite.
Coolant:	
Material	Helium.
Pressure	315 psia.
Temperature:	
Inlet	510° F.
Outlet	1,050° F.
Fuel and physics data:	
Fuel composition	UO ₂ .
Fuel enrichment	2.55%.
Fuel geometry	Seven UO ₂ rods, 0.705 in. OD x 0.323 in. ID.
Fuel cladding	0.020-in.-thick type 304 stainless steel.
Lattice pitch	8.0 in. square.
Maximum fuel surface temperature:	
Design	1,455° F.
Absolute	1,575° F.
Maximum fuel central temperature.	3,300° F.
Specific power of fuel	7.55 Mw/MT of uranium, 300 kw/kg of U ²³⁵ .

Fuel and physics data—		Reactor core—Continued	
Continued		Lattice pitch, one channel on core center line.	10.75 in. triangular.
Average specific power of fuel.	4.2 kw/ft of capsule, 14,300 Btu/hr·ft.	Total number of channels.	253.
Fuel surface heat flux:		Gas inlet temperature to reactor core.	250° C (reference value) rising to 325° C.
Average -----	68,000 Btu/hr·ft ² , 21 w/cm ² .	Bulk gas outlet temperature.	500° C (reference value) rising to 575° C.
Maximum -----	155,000 Btu/hr·ft ² , 49 w/cm ² .	Nominal maximum fuel element can surface temperature.	600° C.
Average core power density.	2.1 kw/liter, 60.4 kw/ft ³ .	Total gas mass flow (250° C reactor inlet, 500° C reactor outlet) for 100 Mw(th) reactor output.	793 lb/sec.
Axial peak-to-average power ratio.	1.66 (control rods inserted 7.5 in.).	Graphite type -----	"A" grade throughout.
Gross radial peak-to-average power ratio.	1.37.	Coolant -----	CO ₂ with up to 2% CO.
Fuel management ----	Continuous fueling.	Fuel and physics data:	
Average fuel exposure	10,000 Mwd/MT, 0.42 fissions per initial fissile atom.	Total weight of fuel (UO ₂) in core (218 fuel channels).	12.35 MT.
Control rods:		Reactor rating:	
Material -----	B ₄ C, stainless-steel jacketed.	Mean -----	9.2 Mw/MT of uranium.
Geometry -----	Cylindrical tube, 3.25 in. OD.	Peak -----	21.2 Mw/MT of uranium.
Pressure vessel:		Feed enrichment at 6,000 Mwd/MT burnup (equilibrium once-through cycle).*	1.2%.
Shape -----	Cylinder.	Averaging factor:	
Dimensions -----	20 ft ID x 46 ft high x 2.75 in. thick.	Axial -----	0.715.
Material -----	Type SA-212 steel, grade B.	Radial -----	0.604.
Containment vessel:		Thermal flux:	
Shape -----	Cylinder.	Average -----	2.68 x 10 ¹⁸ neutrons/cm ² ·sec.
Dimensions -----	114 ft dia x 200 ft high.	Peak -----	6.2 x 10 ¹⁸ neutrons/cm ² ·sec.
<hr/>		Fuel canning material—	
TABLE B.4.—CHARACTERISTICS OF BRITISH ADVANCED GAS-COOLED REACTOR (AGR) WINDSCALE PROTOTYPE		Control:	
General:		Number of rods (in channels on same lattice as fuel channels).	14 (nominally).
Thermal power -----	100 Mw.	Diameter of rods -----	3.5 in. over the inserts.
Total gas circulator power (electrical).	~4.5 Mw.	Neutron-absorbing material.	4% cast boron steel inserts.
Net electrical output--	~28.5 Mw.	Total reactivity absorbed.	9% (nominally).
Reactor core:		*The enrichment level has recently increased. It is estimated the core will contain 10–20% stainless steel elements and the enrichment will be $\geq 1.75\%$.	
Core dimensions:			
Length -----	14 ft.		
Diameter (nominal) --	15 ft.		
Reflector thickness:			
Radial (mean) -----	1 ft 8 in.		
Axial -----	2 ft.		
Graphite dimensions:			
Overall diameter (distance across corners).	18 ft 4 in.		
Overall length -----	18 ft.		

Pressure vessel:

Internal diameter ----	21 ft.
Barrel plate thickness:	
Plain strakes -----	2 $\frac{7}{8}$ in.
Support strakes ----	4 $\frac{3}{8}$ and 3 $\frac{1}{4}$ in.
Dome thickness:	
Unreinforced -----	1 $\frac{3}{4}$ in.
Reinforced -----	3 $\frac{7}{8}$ in.
Over-all height -----	53 ft 9 in.
Length over parallel portion.	32 ft 3 in.
Weight of shell -----	~250 MT.
Weight of diagrid ----	~12 MT.
Weight of graphite structure.	~240 MT.
Number of refueling branches.	253.
Weight of refueling branches (including shield plugs).	~41 MT.
Weight of biological shield, complete.	~6000 MT.
Weight of neutron shield.	~130 MT.
Weight of thermal shield.	~200 MT.

Ducts:

Number of ducts ----	4 concentric pairs.
Internal diameter of shell:	
Inner duct -----	1 ft 11 in. inside lagging.
Outer duct -----	2 ft 10 in.
Wall thickness -----	$\frac{1}{2}$ in. (nominal).
Material -----	Carbon steel.

Heat exchangers:

Number of exchangers	4.
Internal diameter of shell.	11 ft.
Shell thickness -----	1 $\frac{1}{2}$ in.
Overall length -----	~65 ft.
Weight of shell -----	~72 MT.
Weight of tubes, headers, and internal supports.	~70 MT.

Superheater outlet pressure.	660 psig.
Superheater outlet temperature.	460° C.

Gas circulators:

Number of circulators.	4.
Method of grouping --	In parallel.
Gas mass flow per circulator.	198 lb/sec.
Static pressure rise --	~28 psig.
Type of circulator ---	Centrifugal.
Temperature at outlet.	250-325° C.

Gas circulators—Con.

Power at circulator coupling, (at 98% mechanical efficiency).	~1,500 hp.
Shaft speed (maximum).	3,000 rpm.
Type of drive -----	Squirrel-cage electric motor.
Method of flow control	Vane control and throttling.

Main turboalternator:

Number installed ----	1.
Rating at 0.8 power factor.	33 Mw.
Shaft speed -----	3,000 rpm.
Frequency -----	50 cycles.
Generated voltage ----	11 kv.
Turbine stop-valve pressure.	650 psig.
Condenser vacuum ---	28.5 in. Hg.

TABLE B 5—CHARACTERISTICS OF BRITISH FULL-SCALE ADVANCED GAS-COOLED REACTOR (AGR) DESIGN*

	AGR with beryllium-canned fuel	AGR with stainless-steel-canned fuel
General:		
Reactor heat output (flat-tened).	1,797 Mw.	1,797 Mw.
Heat generation:		
Inner zone.	1,080 Mw.	1,080 Mw.
Outer zone.	717 Mw.	717 Mw.
Net electrical output.	562 Mw.	562 Mw.
Station net efficiency.	31.3%.	31.3%.
Core size.	34.3 ft dia, 24.3 ft high.	34.3 ft dia, 24.3 ft high.
Coolant pressure.	216 p s i.	216 p s i.
Coolant terminal temperatures.	T ₁ =260° C. T ₂ =480° C.	T ₁ =260° C. T ₂ =480° C.
Pressure vessel.		
Diameter.	60-ft-dia sphere.	60-ft-dia sphere.
Thickness.	3 in.	3 in.
Design stress.	13,000 p s i.	13,000 p s i.
Fuel and physics data:		
Material.	UO ₂	UO ₂ .
Outside diameter.	Equivalent to thirty-five 0.4-in.-dia solid rods.	Equivalent to forty-five 0.35-in.-dia solid rods.
Canning material.	Beryllium.	Stainless steel
Canning thickness.	0.04 in.	0.012 in.
Fuel average rating:		
Inner zone.	10.7 Mw/MT of uranium.	10.7 Mw/MT of uranium.
Outer zone.	6.9 Mw/MT of uranium.	6.9 Mw/MT of uranium.
Fuel maximum rating.	15 Mw/MT of uranium.	15 Mw/MT of uranium.
Maximum heat flux.	33 w/cm ² .	33 w/cm ² .

See footnote at end of table.

TABLE B.5.—CHARACTERISTICS OF BRITISH FULL-SCALE ADVANCED GAS-COOLED REACTOR (AGR) DESIGN—Con.

	AGR with beryllium-canned fuel	AGR with stainless-steel-canned fuel
Fuel and physics data—Con.		
Maximum temperature:		
Can.....	600° C.....	600° C.
Fuel.....	1,750° C.....	1,750° C.
Weight of fuel:		
Inner zone.....	101 MT of uranium.	101 MT of uranium.
Outer zone.....	105 MT of uranium.	105 MT of uranium.
Flattened radius.....	12 ft.....	12 ft.
Lattice pitch.....	11.1 in.....	11.1 in.
Ratio of moderator volume to fuel volume.	44.9.....	44.9.
Weight of free U ²³⁵ for first charge.	734 kg.....	1,886 kg.
Average irradiation level....	6,000 Mwd/MT of uranium.	6,000 Mwd/MT of uranium.
Critical enrichment:		
Inner zone.....	1.2%.....	1.74%.
Outer zone.....	1.6%.....	2.36%.
Initial enrichment:		
Inner zone.....	1.2%.....	1.85%.
Outer zone.....	1.78%.....	2.68%.
Initial conversion factor:		
Inner zone.....	0.79.....	
Outer zone.....	0.66.....	

* Table B.5 relates to a preliminary design study carried out by the United Kingdom Atomic Energy Authority in 1958–1959. Further work by the Authority in consultation with the Central Electricity Generating Board has shown that a substantially higher performance should be achieved.

TABLE B.6.—CHARACTERISTICS OF GENERAL NUCLEAR ENGINEERING CORPORATION GAS-COOLED REACTOR PROTOTYPE DESIGN

General:

Thermal power (including heat loss).	153.4 Mw.
Station net electrical rating.	50 Mw.
Net plant efficiency ---	32.6%.
Steam conditions at turbine:	
Primary	950° F, 1,465 psia.
Reheat	950° F, 215 psia.
Fuel material	Slightly enriched UO ₂ , 95% theoretical density.
Initial enrichment ---	1.854 at. % U 235.
Containment enclosure..	Cylindrical steel enclosure of primary system.

Fuel and core:

Fuel material	UO ₂ , 95% theoretical density.
---------------------	--

Fuel and core—Con.

Fuel pellet diameter..	0.500 in.
Cladding and structural material.	Type 347 stainless steel.
Cladding thickness ---	0.015 in.
Number of fuel rods per bundle.	19.
Number of bundles per pressure tube.	Undetermined (6 to 12).
Pressure-tube lattice --	Square.
Tube centerline pitch..	8 in.
Pressure-tube material	Zircaloy-2.
Inside tube diameter..	4.75 in.
Wall thickness:	
Nominal	0.141 in.
Minimum	0.125 in.
Insulation material --	Stagnant CO ₂ gas and radiation barrier.
Insulation thickness --	0.200 in.
Number of pressure tubes.	208.
Total UO ₂ load	20.3 short tons.
Equivalent core diameter.	10.83 ft.
Core height or length --	12.0 ft.
Reflector thickness ---	1.0 ft (nominal).
Moderator:	
Moderator material --	D ₂ O.
Weight in tank, cold --	45 tons (nominal).
Operating pressure ---	50 psig.
Temperature:	
Average	231° F.
Outlet	265° F.
Saturation temperature at operating pressure.	297.5° F.
Control rods:	
Number	33.
Shape	Blades.
Size	10 in. wide x 1.5 in. thick.
Material (neutron absorbing).	Cadmium.
Reactor heat transfer:	
Heat flux:	
Average	76,600 Btu/hr·ft ² .
Maximum	190,000 Btu/hr·ft ² .
Maximum clad surface temperature (including all hot spot factors).	1,500° F.
Maximum UO ₂ fuel temperature (including all hot spot factors).	3,800° F.

Reactor heat transfer— Continued		Moderator:	
Reactor rating -----	7.57 Mw/ton of UO ₂ , 8.67 Mw/ton of uranium, 3.43 Mw/ton of D ₂ O in tank.	Moderator material --	D ₂ O.
Fuel exposure -----	10,000 Mwd/ton of ura- nium.	Weight in tank -----	355 tons (includes end shield tanks).
Throughput in 1 year at 0.80 load factor.	5.08 tons of uranium.	Operating pressure ---	50 psig.
Gas coolant system:		Average temperature -	237° F.
Gas -----	CO ₂ .	Control rods:	
Pressure -----	500 psia (nominal).	Number -----	100.
Temperature:		Shape -----	Blades.
Inlet -----	550° F.	Size -----	12 in. wide.
Outlet -----	1,050° F.	Reactor heat transfer:	
Coolant flow -----	1,705,000 lb/hr.	Heat flux across UO ₂ :	
Number of blowers ---	2 half-size.	Average -----	95,000 Btu/hr·ft ² .
		Maximum -----	228,000 Btu/hr·ft ² .
		Maximum fuel temper- ature:	
		Surface -----	1,140° F.
		Central -----	3,500° F.
		Reactor rating -----	8.98 Mw/ton of UO ₂ , 10.2 Mw/ton of uranium, 2.73 Mw/ton of D ₂ O in tank.
		Fuel exposure -----	4,300 Mw/ton of uranium (reactivity limited).
		Throughput per year at 0.80 load factor.	75 tons of UO ₂ .
		Gas coolant system:	
		Gas -----	CO ₂ .
		Pressure -----	500 psi.
		Temperature:	
		Inlet -----	500° F.
		Outlet -----	1,000° F.
		Coolant flow -----	22 x 10 ⁶ lb/hr.
		Number of blowers ---	12.

TABLE B.7.—CHARACTERISTICS OF GENERAL NUCLEAR ENGINEERING CORPORATION GAS-COOLED REACTOR LARGE-SCALE DESIGN

General:	
Thermal power -----	970 Mw.
Net electrical output..	325 Mw.
Net plant efficiency ---	33.5%.
Fuel material -----	UO ₂ , 95% theoretical den- sity.
Enrichment:	
Initial -----	Natural uranium.
Final:	
U 235 -----	0.34 at. %.
Pu -----	0.29 at. %.
Containment enclosure..	185-ft-dia sphere.
Fuel and Core:	
Fuel material -----	UO ₂ , 95% theoretical den- sity.
Fuel diameter -----	0.80 in.
Fuel cladding -----	0.035-in.-thick Be. with 16 fins.
Number of fuel rods per bundle.	19.
Pressure-tube lattice..	Square.
Tube centerline pitch..	10 in.
Pressure-tube material	Zircaloy-2
Inside tube diameter..	5.5 in.
Wall thickness -----	0.164 in.
Insulation material --	Stagnant CO ₂ (gaps).
Number of pressure tubes.	736.
Total UO ₂ load -----	108 short tons.
Core height -----	18 ft.
Reflector thickness ---	16 in.

TABLE B.8.—CHARACTERISTICS OF THE GENERAL ATOMIC GAS-COOLED REACTOR PROTOTYPE DESIGN

General:	
Thermal power -----	115 Mw.
Electrical power:	
Gross -----	46 Mw.
Net -----	40 Mw.
Power cycle -----	Steam, single pressure.
Steam conditions -----	1,000° F, 1,450 psia (at the turbine).
Efficiency:	
Gross -----	40.0%.
Net -----	34.8%.
Active core dimensions..	9.0 ft dia x 7.5 ft high.
Reflector thickness ---	2.0 ft radial, 2.0 ft axial.
Moderator:	
Material -----	Graphite.
Inventory -----	18.5 MT moderator, 47.7 MT reflector.

Coolant:	
Material -----	Helium.
Pressure -----	300 psia.
Temperature:	
Inlet -----	662° F.
Outlet -----	1,382° F.
Fuel and physics data:	
Fuel composition -----	Homogeneous mixture of graphite, UC, and ThC plus additional graphite in center.
Fuel enrichment -----	93 wt % U 235.
Fuel geometry -----	Single rods, 3.125 in. dia x 90 in. long (small central hole in fuel) clad with 0.05-in. graphite plus 0.125-in. sleeves 4.15 in. OD x 145 in. long.
Fuel cladding -----	Graphite.
Lattice pitch -----	4.15 in. triangular array.
U 235:Th:C atomic ratio in core.	1:6.35:1,900.
Maximum fuel surface temperature.	2,400° F.
Average specific power of fuel.	83.4 Mw/MT of U + Th, 606 kw/kg of U 235.
Average fuel surface heat flux.	87,000 Btu/hr·ft ² , 28 w/cm ² .
Average core power density.	8.53 kw/liter, 241 kw/ft ³ .
Gross radial peak-to-average power ratio.	1.51.
Fuel management -----	Batch loading.
Average fuel exposure.	80,900 Mwd/MT of U + Th, 0.63 fissions per initial fissile atom.
Maximum excess k, δk_{eff} .	0.13 (cold, clean).
Maximum operating δk_{eff} .	0.04.
Control rod:	
Rod material -----	B ₄ C dispersed in graphite, probably clad with either graphite or stainless steel.
Rod geometry -----	Hollow cylinders, <4 in. OD, slipped over central support.
Number of rods -----	46.
Pressure vessel:	
Shape -----	Cylinder.
Dimensions -----	14 ft ID x 34.5 ft high (over-all) with flanged heads 2 in. thick.
Material -----	Type SA-212 steel, grade B.
Containment vessel -----	Welded vertical cylinder.

TABLE B.9.—CHARACTERISTICS OF THE GENERAL ATOMIC LARGE-SCALE GAS-COOLED REACTOR DESIGN

General:	
Thermal power -----	805 Mw.
Electrical power:	
Gross -----	365 Mw.
Net -----	325 Mw.
Power cycle -----	Steam.
Efficiency:	
Gross -----	46.4%.
Net -----	40.4%.
Active core dimensions	21.3 ft dia x 17.4 ft high.
Reflector thickness:	
Radial -----	3.0 ft.
Axial -----	3.0 ft.
Moderator:	
Material -----	Graphite.
Inventory -----	253 MT moderator, 387 MT reflector.
Coolant:	
Material -----	Helium.
Pressure -----	300 psia.
Temperature:	
Inlet -----	662° F.
Outlet -----	1,382° F.
Fuel and physics data:	
Fuel composition -----	Homogeneous mixture of graphite, UC, and ThC.
Fuel enrichment -----	93 wt % U 235.
Fuel geometry -----	Single rods, 8.35 in. dia over-all, 17.4 ft long.
Fuel cladding -----	Graphite.
Lattice pitch -----	8.35 in. triangular array.
U ²³⁵ :Th:C atomic ratio in core.	1:24.2:16,000.
Maximum fuel surface temperature.	2,400° F.
Average specific power of fuel.	40 Mw/MT of U + Th, 1,000 kw/kg of U 235.
Average fuel surface heat flux.	94,500 Btu/hr·ft ² .
Average core power density.	4.35 kw/liter.
Average fuel exposure	75,000 Mwd/MT of U + Th, 2.0 fissions per initial fissile atom.
Control rod:	
Rod material -----	B ₄ C in graphite.
Rod geometry -----	Hollow cylinders.
Number of rods -----	60.
Pressure vessel:	
Shape -----	Cylinder.
Dimensions -----	28.3 ft ID x 60 ft high (over-all), 4 in. thick.