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OPERATIONAL SAFETY OF THE ORNL GRAPHITE REACTOR USING ENRICHED-URANIUM FUEL

P. R. Kasten
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REACTOR EXPERIMENTAL ENGINEERING DIVISION

OPERATIONAL SAFETY OF THE ORNL GRAPHITE REACTOR USING ENRICHED-URANIUM FUEL

P. R. Kasten S. Jaye M. P. Lietzke

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PREFACE

This work was initiated in the summer of 1957 in connection with a review of the safety of the ORNL Graphite Reactor if enriched-uranium fuel elements were used. The study was completed in 1958, but the press of other problems delayed completion of this report. The actual conversion of the reactor to enriched fuel has not proceeded, and at this time it is not clear whether the conversion will be made. This report is being issued in the belief that many of the results will be of general interest.

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OPERATIONAL SAFETY OF THE ORNL GRAPHITE REACTOR USING ENRICHED-URANIUM FUEL

P. R. Kasten S. Jaye M. P. Lietzke

ABSTRACT

Safety studies were made pertaining to the use of highly-enriched-uranium fuel elements in the ORNL Graphite Reactor; these studies were made on the basis that permissible reactivity additions are reactivity additions which will not cause the fuel elements to melt. Appropriate equations covering power generation, heat transfer, and fluid flow were coded for the Oracle; numerical integration of the equations was performed for various instantaneous reactivity additions, safety-rod response times, and safety-rod reactivity worths. Both the present natural-uranium system and the proposed enriched system were considered. It was assumed that an instantaneous reactivity addition took place at time zero, and that the safety rods started to fall by gravity a specified time after the power level had exceeded its nominal level by 20%. The maximum reactivity addition which would not cause melting of fuel elements was calculated and was termed the maximum permissible reactivity addition.

The maximum permissible reactivity addition for the enriched-uranium system (operating nominally at 7 Mw) was found to be 2.4% Δk_e. This value is contingent on the following conditions: the safety rods are initially placed at the outer edge of the core; a 100-msec lag is associated with instrument response time (this time lag corresponds to the interval between the time at which the power level reaches 8.4 Mw and the time at which the safety rods start to drop by gravity); and the vertical safety rods which respond are worth 4% Δk_e (50% of total worth). With the present natural-uranium reactor, assuming that 50% of the total safety-rod worth is available, the maximum permissible reactivity addition is less than 0.7% Δk_e. It appears that the enriched system will be protected against fuel melting to at least the same extent as the present natural-uranium system so long as half the safety rods function and the total rod worth is increased to 8% Δk_e.

The sign of the metal temperature coefficient of reactivity in the enriched system is not known with certainty, but the magnitude is about 10^{-5} $\Delta k_e/^{\circ}C$; so long as the safety rods function, the sign of this coefficient does not significantly affect the maximum permissible reactivity addition. Neither does power failure in the fan circuit significantly affect the permissible reactivity addition so long as the safety rods function when the air flow drops below 80% of the normal value.

The countercurrent method of fueling, wherein fuel in adjacent channels is traveling in opposite directions, is recommended for high fuel burnups. With this scheme and 40% fuel burnup and 7-Mw operation, the maximum steady-state fuel temperature was 850°F; the corresponding temperature with 0% burnup was 800°F.

INTRODUCTION

Replacement of the solid, cylindrical natural-uranium fuel in the ORNL Graphite Reactor with tubular highly enriched fuel would permit attainment of a higher thermal-neutron flux at a given power level due to the removal of U^{238} (a neutron poison). Use of highly enriched fuel (alloyed with aluminum) should also permit operation with fewer slug failures (the alloyed highly enriched fuel elements have greater resistance to both thermal-cycle and radiation damage). In addition, since the area available for heat transfer with the proposed tubular enriched fuel elements will be about twice the area of the present cylindrical slugs (on a per-unit-length basis), the permissible power per unit length of fuel will be greater.

A primary factor in fuel replacement is the effect of such replacement on reactor safety, since there are possible difficulties associated with the use of highly enriched uranium which do not exist with the use of natural uranium. Although the fuel replacement will lead to a higher flux because of the smaller critical mass of U^{235}, the reactivity worth of experiment materials inserted into the reactor will be increased. Although the control rods also have more worth under these

conditions and the neutron lifetime is increased, it still would be necessary that the response of the control system be faster than the present response (designed for natural-uranium fuel) in order to maintain the same degree of reactor safety (in this study a period of 100 msec was associated with instrument delays in the enriched system, compared with 600 msec in the present system).

A factor to be considered in the required response of the control system is the heat capacity of the fuel elements. Because of the relatively low mass of fuel present, the energy required to raise the temperature of the enriched-uranium fuel elements to the melting point of aluminum is appreciably less than the corresponding value for natural-uranium fuel. Also, removal of the U^{238} from natural uranium results in a metal temperature coefficient of reactivity that is less negative than in the present reactor (and may be slightly positive), which increases the dependence of reactor safety upon the response of the control and safety system.

In the study reported here, the maximum reactivity addition which would not cause melting of the fuel elements was determined as a function of control-system response time. Temperature distributions for various fuel loadings and burnups were also obtained. In addition, for purposes of comparison, the maximum reactivity addition which would not cause melting of aluminum in the present natural-uranium reactor system was calculated.

REACTOR SYSTEM

The characteristics of the enriched-uranium graphite reactor will generally be unlike those of the present natural-uranium system. The size will be smaller, with the fuel loading being in the form of a right-circular cylinder 16 ft long and 17 ft in diameter. The fuel elements will be hollow cylinders 2 ft long, having an outside diameter of about 1.25 in. and a thickness of 60 mils. The elements will consist of a 20-mil-thick section of U^{235}-Al alloy clad with 20 mils of aluminum on each side and will contain about 5 g of U^{235} per linear foot. The elements will rest on the graphite in the same way that the present natural-uranium elements do. Cooling air will flow past the inside and outside surfaces. The cooling-system fans presently installed will be used, but the control and safety systems will be modified for more rapid response.

The power level will be about 4 Mw but may be increased later to 7 Mw. In this study the steady-state power was assumed to be 7 Mw, with 530 channels loaded with fuel and with cooling air entering the reactor at 100°F.

A summary of some pertinent reactor-system characteristics for both the natural- and enriched-uranium reactors is given in Table 1. Additional background information on the ORNL Graphite Reactor is given in Appendix A.

The information developed here is associated with the maximum permissible reactivity addition as a function of control-rod response. There are other aspects of reactor safety – for example, operation with enriched fuel at proposed power levels will increase the production rate of radioactive argon; also, operation at higher power densities, higher fuel burnups, and higher fuel-element temperatures requires knowledge of radiation-damage and temperature effects on possible dispersal of fission products. However, these aspects, which have been investigated by others,[1,2] will not be considered here. Particular mention should be made of the studies at Brookhaven National Laboratory (BNL) in connection with the conversion of their natural-uranium graphite reactor to enriched uranium; Appendix B summarizes the results of those studies. Many of the BNL results can be applied directly to the ORNL reactor.

MODEL USED

Nuclear and Temperature Calculations

The information desired was the maximum reactivity addition which would not cause melting of the fuel elements, as a function of control-system response time. The procedure followed in obtaining this information involved numerical integration of the appropriate equations associated with energy generation and removal. More specifically, a given reactivity addition was assumed to be introduced instantaneously with the reactor initially at 7 Mw, and the safety rods

[1] Personal communications from C. D. Cagle, J. A. Cox, and D. K. Holmes, ORNL, 1957–1958.

[2] N. Corngold *et al.*, *Summary Hazards Report on Enriched Fuel Element Loading for the BNL Research Reactor*, BNL Log No. C-9562 (May 24, 1956) (also issued as BNL-3146).

Table 1. Comparison of Reactor Characteristics*

Characteristic	Present Reactor	Enriched Reactor
General		
Moderator	Graphite	Graphite
Reflector	Graphite	Graphite
Fuel	Natural U	Enriched U
Loaded zone	Cubic (18 ft × 18 ft × 18 ft 8 in.)	Cylindrical (H = 16 ft; R = 8.5 ft)
Fuel channels in core	830	570
Critical loading (kg)		
U^{235}	193	29.5
U^{238}	27,216	3.0
Al	700	634
Operating loading (kg)		
U^{235}	343	32.8
U^{238}	47,630	3.3
Al	1240	705
Nuclear properties		
Buckling, B^2, cm^{-2}	92×10^{-6}	88.7×10^{-6}
Thermal utilization, f	0.890	0.563
Resonance escape probability, p	0.886	0.999
Material multiplication constant, k	1.067	1.194
Thermal diffusion length, L^2, cm^2		
Core	297	1101
Reflector	2400	2400
Prompt-neutron lifetime, sec	0.001	0.0038
$(\partial k_e/\partial T)_{graphite}$, $(°C)^{-1}$	-2.08×10^{-5}	-12.60×10^{-5}
$(\partial k_e/\partial T)_{fuel}$, $(°C)^{-1}$	-0.78×10^{-5}	~0
$(\partial k_e/\partial T)_{N_2}$, $(°C)^{-1}$	(Included in graphite)	$+1.18 \times 10^{-5}$
Reactivity values, Δk_e		
Operating reactivity requirements		
Temperature and spare for control	0.004	0.026
Xenon	0.002	0.008
Experiments, nonpermanent	0.001	0.003
Burnup, four weeks		0.002
Total	0.007	0.039
Reactivity in permanent experiments	0.005	0.015
Total excess reactivity (cold, clean reactor)	0.012	0.054

Table 1 (continued)

Characteristic	Present Reactor	Enriched Reactor
Reactivity worth of shutdown rods	0.027	0.077
Reactivity worth of regulating rods	0.0107	0.03
Maximum permissible excess reactivity with present shutdown rods**	0.019	0.039
Power and flux		
Reactor power level, Mw	3.5	4.0 (initially; 7 Mw design)
Average thermal-neutron flux, neutrons·cm^{-2}·sec^{-1}	4.5×10^{11}	2.6×10^{12}
Maximum thermal flux, neutrons·cm^{-2}·sec^{-1}	1.1×10^{12}	5.6×10^{12}
Fast flux/thermal flux	1	0.3
Heat transfer		
Maximum fuel temperature, °C	280	350
Typical cooling-air flow per loaded fuel channel, lb/min	6.5	10
Temperature of inlet air, °F	100	100

*Most of the values given here were obtained either from D. K. Holmes or C. D. Cagle, ORNL, personal communications; 1957–1958.

**Some question exists as to the dependability of the horizontal rods.

started to fall by gravity at a specified time after the power level had exceeded its nominal level by 20%. The net reactivity and power level were calculated as a function of time. As the power rose, the temperature of the fuel elements increased, as did the rate of cooling of the fuel, and the maximum fuel temperature was obtained as a function of time for various reactivity additions and safety-rod lag times. The maximum permissible reactivity addition was that which caused the maximum fuel-element temperature to reach but not exceed the melting temperature of aluminum.

The actual reactor has fuel spaced on 8-in. centers in a rectangular lattice cell, with diamond-shaped cooling channels. For practical purposes, this lattice cell was represented in cylindrical geometry, as illustrated in Fig. 1. The cross-sectional area of the cylindrical cell was equated to that of the actual 8- by 8-in. cell.

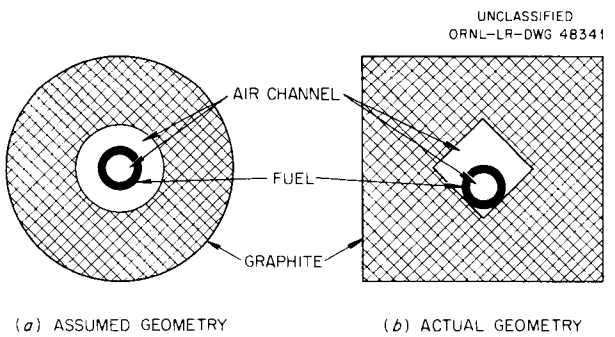

Fig. 1. Comparison of Cell Geometry Considered in Calculations vs Actual Geometry.

In calculating the fuel-element temperature, consideration was given to heat transfer from the fuel element to the graphite by radiative, convective, and conductive transfer (at contact surface). (Standard methods were used for obtaining the radiative and convective coefficients. The conductive coefficient was based on experimental

measurements;[3,4] the experiments were performed with cylindrical aluminum cans whose dimensions and weight per linear foot approximated those of the proposed fuel elements.) The steady-state heat transfer equations were:

for air contacting the inner surface of the fuel element,

$$-W_i c_i \frac{dT_i}{dx} + h_i p_i (T_f - T_i) = 0 \quad ; \tag{1}$$

for the fuel,

$$Q_f A_f - h_i p_i (T_f - T_i) - \\ - h_o p_o (T_f - T_o) - K(T_f - T_{cs}) = 0 \quad ; \tag{2}$$

for the air outside the fuel,

$$-W_o c_o \frac{dT_o}{dx} + h_o p_o (T_f - T_o) + \\ + h_o p_{cs} (T_{cs} - T_o) = 0 \quad ; \tag{3}$$

at the graphite surface,

$$Q_c A_c - h_o p_{cs} (T_{cs} - T_o) + \\ + K(T_f - T_{cs}) = 0 \quad ; \tag{4}$$

in the graphite,

$$\frac{k_c}{r} \frac{\partial}{\partial r} \left(r \frac{\partial T_c}{\partial r} \right) + Q_c = 0 \quad ; \tag{5}$$

and at the cell boundary,

$$\frac{\partial T_c}{\partial r} = 0 \quad . \tag{6}$$

The above system of equations was solved analytically to give the steady-state temperature distribution (the inlet coolant-air temperature was taken as 100°F). This distribution was used as the initial condition for the "transient" calculations.

The power distribution in the reactor core was assumed to be cosine in nature (the effects of deviations from this on safety are considered later); specifically, Q was represented by

$$Q_f = C_1(t) \sin \frac{\epsilon + x}{2\epsilon + L} \pi \tag{7}$$

and

$$Q_c = C_2(t) \sin \frac{\epsilon + x}{2\epsilon + L} \pi \quad . \tag{8}$$

During a power excursion, the spatial power distribution remained cosine in nature, but the magnitude changed.

The magnitude of the power excursion was obtained from the reactor kinetic equations, using five groups of delayed neutrons. Changes in reactivity were those associated with reactivity insertions, safety-rod movement, and temperature variations. The fuel and moderator temperature coefficients were included, but they had very little effect for the cases studied; the reactivity associated with fuel temperature change was obtained from the average fuel coefficient (uniform temperature change) weighted by the flux and temperature change. The specific kinetic equations used were:

$$\frac{dP}{dt} = \frac{k_e(1 - \beta) - 1}{l} P + \sum_{i=1}^{5} \lambda_i \mu_i \quad ; \tag{9}$$

$$\frac{d\mu_i}{dt} + \lambda_i \mu_i = k_e \frac{\beta_i}{l} P \quad ; \tag{10}$$

$$k_e = 1 + \Delta k_e - \Delta k_{sr} + \\ + \frac{\int_{V_r} \phi^2 \left[\Delta T_f \left(\frac{\partial k_e}{\partial T} \right)_f + \Delta T_c \left(\frac{\partial k_e}{\partial T} \right)_c \right] dV}{\int_{V_r} \phi^2 \, dV} \quad . \tag{11}$$

The transient heat transfer equations were:

for air inside the fuel annulus,

$$-W_i \frac{\partial T_i}{\partial x} + \frac{h_i p_i}{c_i} (T_f - T_i) = A_i \rho_i \frac{\partial T_i}{\partial t} \quad ; \tag{12}$$

[3] C. E. Shaw, J. Z. Ryan, and H. Specter, *Graphite Reactor Fuel Element Heat Transfer Coefficient*, KT-279 (Mar. 22, 1957).

[4] S. Kelinberg and M. A. St. John, *Graphite Reactor Fuel Element Heat Transfer Coefficient*, KT-292 (May 30, 1957).

for the fuel,

$$\frac{Q_f}{\rho_f c_f} - \frac{h_i p_i}{A_f \rho_f c_f}(T_f - T_i) - \frac{h_o \rho_o}{A_f \rho_f c_f}(T_f - T_o) - \\ - \frac{K}{A_f \rho_f c_f}(T_f - T_{cs}) = \frac{\partial T_f}{\partial t} \quad ; \qquad (13)$$

for air outside the fuel annulus,

$$-W_o \frac{\partial T_o}{\partial x} + \frac{h_o p_o}{c_o}(T_f - T_o) + \\ + \frac{h_o p_{cs}}{c_o}(T_{cs} - T_o) = A_o \rho_o \frac{\partial T_o}{\partial t} \quad ; \qquad (14)$$

for the graphite,

$$\frac{k_c}{r}\frac{\partial}{\partial r}\left(r \frac{\partial T_c}{\partial r}\right) + Q_c = \rho_c c_c \frac{\partial T_c}{\partial t} \quad ; \qquad (15)$$

and for the graphite boundary conditions,

$$K(T_f - T_{cs}) - h_o p_{cs}(T_{cs} - T_o) \\ = k_c A_c \left(\frac{\partial T_c}{\partial r}\right)_{cs} \quad , \qquad (16)$$

$$\left(\frac{\partial T_c}{\partial r}\right)_{\text{cell boundary}} = 0 \quad . \qquad (17)$$

The transient power and temperature equations were written in finite difference form, and programmed for the Oracle. For a given case, the calculation was terminated when any fuel temperature in the reactor exceeded 1220°F or if the highest fuel-element temperature began to decrease with time. When calculating the temperature distributions, extremely short time intervals had to be used to keep the numerical-integration procedure stable. In order to cut down machine time, the $\partial T_i/\partial t$ and $\partial T_o/\partial t$ terms were taken as zero, which permitted the temperatures of air inside and outside the fuel annulus to be explicitly related to the fuel temperature at each position and time. This is equivalent to assuming that the air is in constant equilibrium with the fuel. Since the air temperature during the important part of the transient would be lower than the equilibrium value, this approach leads to calculated fuel temperatures which are higher than would actually be encountered. However, such an effect would be small for the cases studied here.

The reactor was divided into five concentric cylindrical regions in such a way that

$$\int_{R_{j-1}}^{R_j} \phi^2(r)\, r\, dr$$

was a constant. The radial variation in the flux was taken to be $J_0(2.4r/R_1)$. Along a given cell, the length was broken up into 1-ft intervals for integration purposes; in the graphite, four equally spaced radial positions were considered. The temperature drop through the 60-mils-thick fuel element was considered negligible in comparison with the drop across the air film. It was assumed that 95% of the reactor power was generated in the fuel, and 5% in the graphite.

Utilizing the equations indicated above, the reactor power and temperatures were calculated as a function of time for various instantaneous reactivity additions; the safety rods were assumed to fall by gravity at a specified time (lag time) after the power had risen 20% above the nominal operating power level. Insertion of the safety rods was assumed to remove reactivity at various rates (many of the calculations considered reactivity removal at 5% Δk_e/sec for 1 sec). The associated temperature rise was calculated as a function of time (time increments were usually 40 msec) until the highest fuel temperature reached 1220°F or began decreasing before that temperature was reached; the reactivity addition associated with "turnover" in maximum fuel temperature was considered permissible, while attainment of 1220°F was considered unsafe.

Fuel Loading and Power Distribution

The flux and power distributions for various fuel loadings and burnups were based on the work of Corngold[5-8] (a detailed presentation of the

[5] N. Corngold, *Flux Distribution in the Modified BNL Reactor*, BNL Log No. C-8101 (June 10, 1954).

[6] N. Corngold, *Flux and Power Distribution and Critical Loading in the Modified BNL Reactor*, BNL Log No. C-8614 (Jan. 10, 1955).

[7] N. Corngold, *The Modified BNL Reactor III*, BNL Log No. C-9037 (July 5, 1955).

[8] N. Corngold, *The Modified BNL Reactor IV*, BNL Log No. C-9087 (Aug. 9, 1955).

calculational method is given by Prince[9]). The equations were essentially of the form

$$D \nabla^2 \phi(x) + (k_\infty - 1) \Sigma^T(x) \phi(x) = 0 \ , \quad (18)$$

$$v \frac{dN^{25}}{dx} = -\Sigma^{25}\phi \ , \quad (19)$$

$$v \frac{dN^p}{dx} = \Sigma_f^{25}\phi - \Sigma^p\phi \ . \quad (20)$$

The reactor was considered to be one-dimensional, and all neutrons were treated mathematically as "thermal" neutrons [the independent variable, x, was in the longitudinal direction; the radial flux distribution was considered to be given by $J_0(2.4r/R_1)$]. Steady-state conditions were assumed, with fuel continuously moving in a specified manner. Four loading schemes were considered: (1) new fuel introduced at the front face of the reactor and spent fuel removed at the rear face; (2) new fuel introduced at both the front and rear faces of the reactor (in adjacent channels – termed "countercurrent fueling") and spent fuel removed at both the rear and front faces; (3) new fuel introduced at both the front and rear faces of the reactor and spent fuel removed at the center of the reactor; and (4) new fuel introduced at the center of the reactor and spent fuel removed at both the front and rear faces.

The different loading schemes were considered in conjunction with various fuel burnups to give the flux and power distributions; combining the power-density distribution with Eqs. (1)–(6) gave the temperature distribution of the fuel in the longitudinal direction.

Loss of Coolant Flow

If an electrical power failure occurred, there would be a loss in coolant flow. A study was therefore made to determine whether the fuel temperature would become excessive if, at the time the flow dropped to 80% of normal, the safety rods were inserted into the reactor after specified lag times.

Following a loss of power, the air flow rate produced by the fans would decrease with time; under this circumstance the flow rate as a function of time was taken to be

$$W(t) = W(0) - \Delta W(1 - e^{-mt}) \ . \quad (21)$$

The values for $W(0)$, ΔW, and m were such that the calculated flow rate vs time after power failure agreed with experimental data.[10]

In calculating the fuel temperature rise, the heat transfer coefficient associated with the air flow was considered to vary as the 0.8 power of the air flow rate, while the reactivity and power were obtained from Eqs. (9)–(11).

Comparison Between Reactor with Natural-Uranium Fuel and Reactor with Enriched-Uranium Fuel

In order to obtain a better understanding of the safety of the reactor containing enriched fuel, calculations similar to the ones described in "Nuclear and Temperature Calculations" were also performed for the present ORNL reactor; the results provide a normalization point in terms of relative reactor safety.

RESULTS

Effect of Lag Time (Associated with Safety-Rod Motion) upon Permissible Reactivity Addition

Two primary factors which influence the safety of the reactor are the amount of reactivity which can be inserted into the system and the time required for safety rods to be inserted into the core region. Although the reactivity available in the enriched system will be greater than that available in the natural-uranium system, the controls will be altered so that the safety-rod response time will be less when enriched fuel is used. In the present natural-uranium reactor, the safety-rod response time is about 1000 msec, with 600 msec due to instrument delays and 400 msec due to rod travel time through low-flux regions. In the enriched system, the instrument delay time will be reduced to 100 msec; the time of rod travel in low-flux regions will be reduced to about 200 msec by positioning the rods so that their "out" position is at the edge of the core rather than in the reflector region. Although the heat capacity of the enriched-uranium fuel elements is appreciably lower than for the natural-uranium elements,

[9] B. E. Prince, *Permissible Exposures of U-233-Thorium Fuel Elements in Reactors Having Continuous Fuel Feed,* ORNL CF-57-9-95 (Sept. 30, 1957).

[10] Personal communication from J. A. Cox, ORNL, 1958.

the decrease in rod response time and the longer neutron lifetime permit relatively large amounts of reactivity to be safely added to the enriched system.

It is planned that conversion from a natural- to an enriched-uranium loading will be done in one step. Based on an enriched-fuel loading, the excess reactivity in the reactor system will be limited to 3.9% Δk_e relative to criticality in the clean, cold reactor. At operating temperature, 2.6% of the available excess is absorbed by the over-all negative temperature coefficient, which leaves only 1.3% available for an excursion. A large reactivity addition could conceivably occur during startup from a cold condition. However, since the reactor can become critical only after the vertical safety rods are completely withdrawn and after the horizontal shutdown rods are nearly withdrawn, a large positive Δk_e can be introduced only by the simultaneous withdrawal of the two regulating rods. At least 20 sec would be required for the withdrawal with the present withdrawal speed, and this situation can be handled safely. For example, starting at an initial power of 10^{-3} w (source power at criticality with a neutron source of 10^6 neutrons/sec corresponds to a power greater than 10^{-3} w), a rate of reactivity addition of 0.12% Δk_e/sec would cause the reactor power to reach 10 Mw in $\sim$12 sec, so that the safety rods would be tripped before $\sim$1.5% reactivity was added to the system, even if the power trip were the only safety device which operated.

In this study, several safety-rod lag times were considered; the results are in terms of the maximum reactivity addition which would not result in a maximum fuel temperature above 1220°F. Some of the results are given in Fig. 2 for the case of a total of either 5 or 10% reactivity worth in the safety rods. Positions lying above the appropriate curve correspond to maximum fuel temperatures greater than 1220°F; those below the curve correspond to maximum temperatures below 1220°F. As can be seen from these curves, the permissible reactivity addition is larger, and also more sensitive to lag time, when the lag time is relatively small.

The safety-rod lag time given in Fig. 2 is that delay which is associated with the electrical-mechanical safety systems and the initial movement of the rods prior to reducing reactivity. There is also a time required for the power to rise to 8.4 Mw (20% in excess of operating power) before the scram begins; this latter time has been included in the analysis and is not a part of the lag time given in Fig. 2. After the specified lag time, reactivity was considered to be ejected from the reactor at either 5 or 10% per second, for 1 sec (5% for the lower curve). Since the reactivity-position relationship is in fact not linear but S-shaped, the actual rate of reactivity compensation would be less than 5% (or 10%) per second initially, but would be greater than 5% (or 10%) per second during most of the travel. These points are considered below.

If the safety rods were initially at the top of the active region of the reactor, about ½ sec would be required before these rods start to have appreciable reactivity effect. Under these circumstances there would be an inherent time lag of 500 msec plus instrument and circuit delays. A reasonable instrument and circuit delay time appears to be 100 msec.[11] This would mean that the appropriate lag time to consider in Fig. 2 would be about 600 msec, and so the maximum permissible reactivity addition would be $\sim$2.2% Δk_e. However, if the rods were initially inserted $\sim$4 ft within the reactor, it would not take so long for the rods to drop into the effective region, and the appropriate time lag associated with safety-rod movement would be about 200 msec, or a total lag time of $\sim$300 msec; this would permit $\sim$3% Δk_e for the 5% rods. If the safety rods were spring-loaded, the effective lag time and rod-insertion time would be appreciably reduced. For example, an acceleration twice that of gravity would reduce the total lag time for the above cases to $\sim$300 and 200 msec, respectively, corresponding to permissible Δk_e's in excess of 2.9 and 3.3%.

Since the worth of the safety rods as a function of position is S-shaped, the results given in Fig. 2 are only approximations to the actual permissible reactivity additions. Also, it may be desirable to place the safety rods initially at a position other than at the surface of the active reactor lattice. Calculations were performed in which the rod worth as a function of position was considered to be cosine in nature, with the end

[11] Personal communication from E. P. Epler, ORNL, 1957.

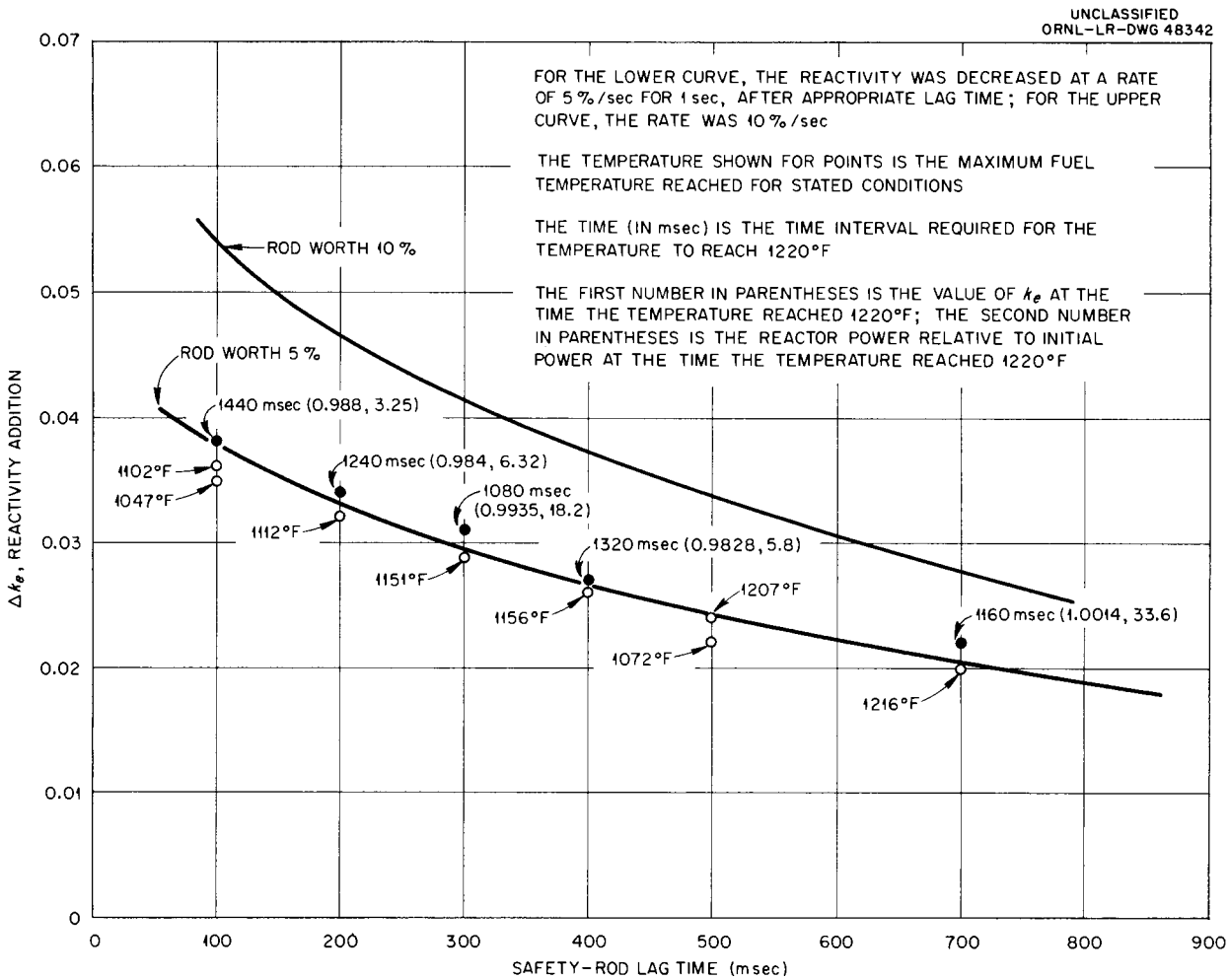

Fig. 2. Maximum Permissible Reactivity Addition as a Function of Rod Lag Time and Reactivity Worth of Safety Rod.

of the rod initially either at the core surface or partially inserted into the active lattice to a distance δ; the rod was assumed to drop by gravity after release. The equations considered were

$$\Delta k_{sr}(\theta) = \frac{\Delta k_{sr}^{T}}{2}\left[\cos\frac{\pi\delta}{L} + \sin\pi\left(\frac{\delta + \frac{1}{2}g\theta^2}{L} - \frac{1}{2}\right)\right] \qquad (22a)$$

for

$$0 \leqq \theta \leqq \sqrt{\frac{2(L-\delta)}{g}}$$

and

$$\Delta k_{sr} = \frac{\Delta k_{sr}^{T}}{2}\left(1 + \cos\frac{\pi\delta}{L}\right) \qquad (22b)$$

for

$$\theta > \sqrt{\frac{2(L-\delta)}{g}}\ ;$$

with θ equal to the time from the instant of rod release. Figure 3 gives the relative worth of the rod as a function of time after rod release for the case when the rod is initially at the edge of the core ($\delta = 0$) and also when it is inserted initially one-fourth of the way into the core ($\delta = L/4$).

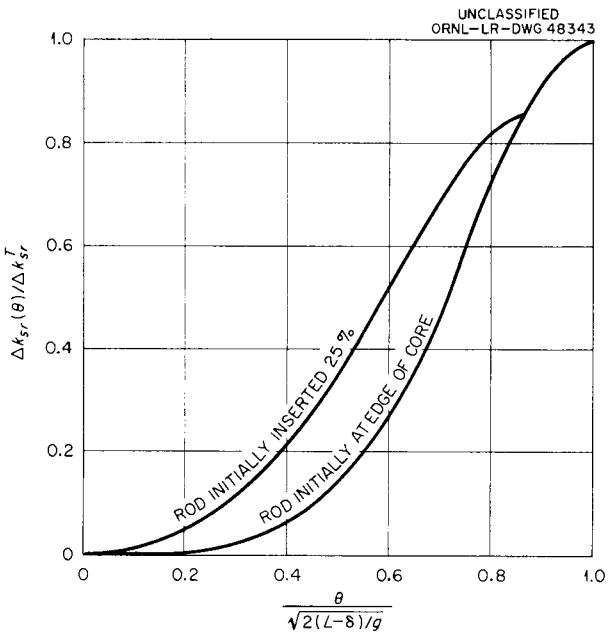

Fig. 3. Worth of Safety Rod as a Function of Initial Insertion and Time After Rod Release.

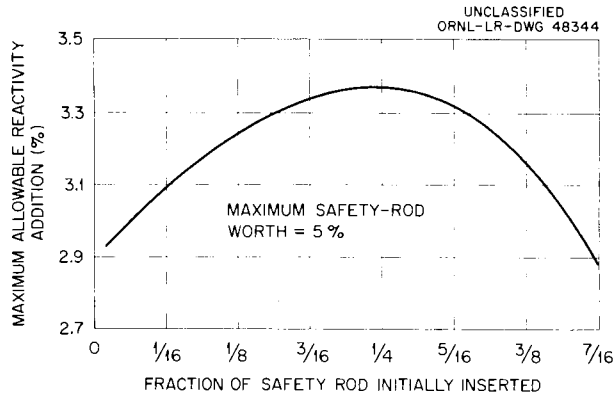

Fig. 4. Maximum Permissible Reactivity Addition vs Initial Safety-Rod Position.

Using Eqs. (22*a*) and (22*b*) and a maximum rod worth of 5%, the maximum permissible reactivity addition was obtained as a function of the initial rod insertion. The results are shown in Fig. 4; positions lying above the curve represent conditions leading to maximum fuel temperatures greater than 1220°F, while those below the curve correspond to conditions which do not cause fuel-element melting. The maximum permissible reactivity addition increases with initial rod insertion because the first movement of the rod has more reactivity effect if the rod is placed initially inside the active core; a maximum occurs in the curve of Fig. 4 since the effective total worth of the rod decreases as the initial rod position is moved toward the center of the reactor. An initial insertion of 25% is seen to maximize the permissible addition.

Although most of the calculations considered the worth of the safety rods to be 5%, this value may be too high. As given in Table 1, the reactivity worth of all present shutdown rods in the enriched reactor would be ~7.7%; of this, about 4.5% would be in the present three vertical safety rods. However, another safety rod could be added to the reactor by making use of an existing hole, which would increase the 4.5% value to about 6%. Additional rods can be placed in the reactor by drilling more holes in the graphite; thus a total worth of 10% Δk_e in the vertical safety rods is not unreasonable. If it is assumed that only 50% of the total reactivity worth of these rods is available in an emergency, the value of 5% used in these calculations is obtained. To indicate the maximum permissible reactivity addition as a function of rod worth, the results obtained in Fig. 2 are plotted in a different fashion in Fig. 5. Comparison of the results in Fig. 4 (with the rods placed initially at the edge of the core) with those in Fig. 5 indicates that the effective lag time, on the basis of the model used in obtaining the results in Fig. 2, is 300 msec for a rod worth 5% Δk_e.

If the rod worth which could be guaranteed were 4.0% Δk_e, then the results in Fig. 5 indicate that the maximum permissible reactivity addition would be ~2.5% Δk_e.

Calculations were also performed for the present natural-uranium reactor; in these, the model used was the same as that employed for the enriched system. The maximum permissible reactivity addition for the present reactor was found to be 1.2% Δk_e, on the basis of 3% reactivity worth in the safety rods and a safety-rod lag time of 1 sec. If the lag time were 1.5 sec, the maximum permissible reactivity addition would be 1.0%. Taking credit for only 50% of the safety-rod worth (~1.5% Δk_e), the permissible reactivity addition would be less than 0.7% Δk_e for the above lag times. The natural-uranium-fueled reactor is more sluggish (thermally) than the enriched one would be;

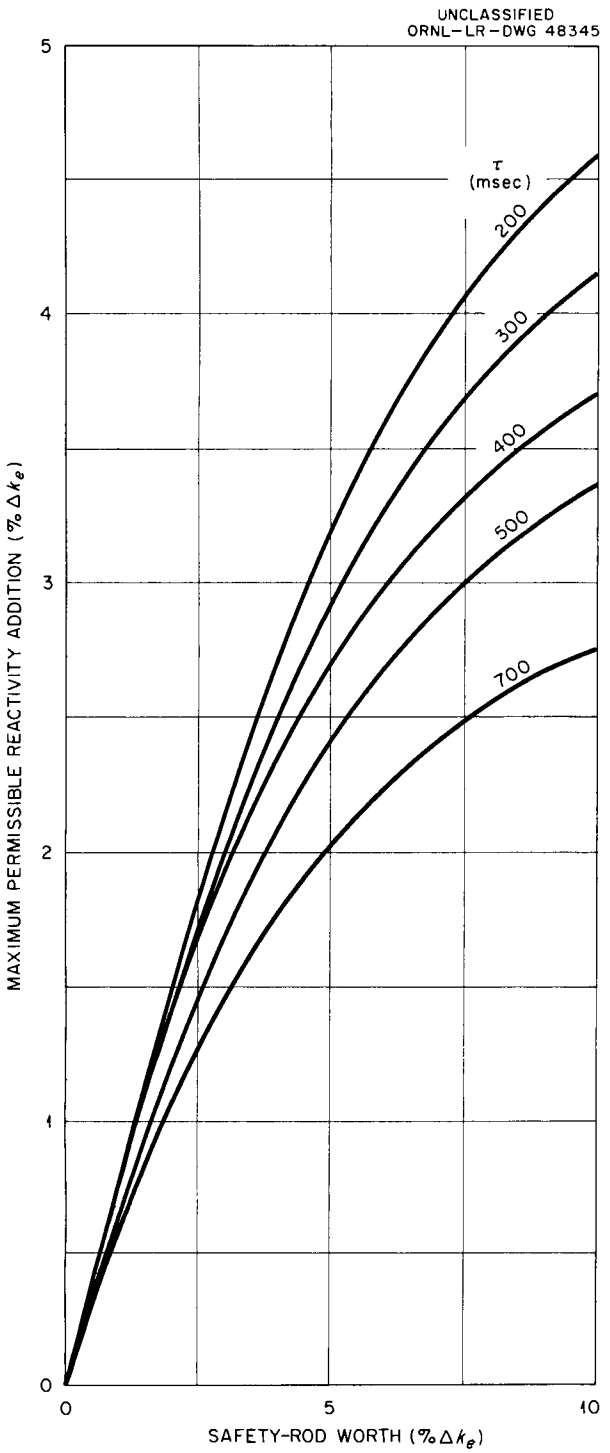

Fig. 5. Effect of Maximum Rod Worth on Maximum Permissible Reactivity Addition as a Function of Effective Lag Time.

however, due to its relatively short neutron lifetime, relatively weak safety rods, and sluggish control system, it can tolerate relatively small insertions of excess reactivity. However, smaller potential reactivity additions are associated with a natural-uranium system.

Effect of Metal Temperature Coefficient of Reactivity upon the Permissible Reactivity Addition

The relatively large negative temperature coefficient of reactivity associated with air-cooled graphite-moderated reactors is due primarily to the moderator coefficient and is associated with increasing the average energy of the neutron spectrum. However, because of the small heat capacity of the enriched fuel element and its inability to transfer heat rapidly to the graphite, the graphite temperature coefficient is ineffective during reactivity incidents. Further, the negative temperature coefficient associated with increased poisoning of the reactor due to Doppler broadening of U^{238} resonances is missing. There will be some Doppler broadening of the U^{235} resonances, which may or may not give a positive temperature coefficient of reactivity. Experimental studies at Hanford[12] indicate that the Doppler coefficient for U^{235} fuel elements is small ($\sim 10^{-5}\ \Delta k_e/°C$); however, the accuracy of the experiment was not sufficient to determine whether the coefficient was positive or negative. In these studies, two values for the metal temperature coefficient were used: $+5 \times 10^{-5}$ and -5×10^{-5}. These values had little effect on the maximum permissible reactivity addition (to within $0.002\ \Delta k_e$), so long as the safety rods functioned.

Effect of Neutron Lifetime on Permissible Reactivity Addition

The neutron lifetime in the modified ORNL Graphite Reactor has been estimated to be 3.8 msec, and this value was used in nearly all the calculations. However, in order to see what effect this value had on the maximum permissible reactivity addition, neutron lifetimes of 2 and 5 msec were used to find the maximum permissible reactivity addition for a representative case

[12] M. V. Davis, *Nuclear Sci. and Eng.* 3, 107 (1958).

(safety-rod lag time of 300 msec and rod worth of 5%) using the model on which the results in Fig. 2 are based. The results of these calculations are shown in Fig. 6.

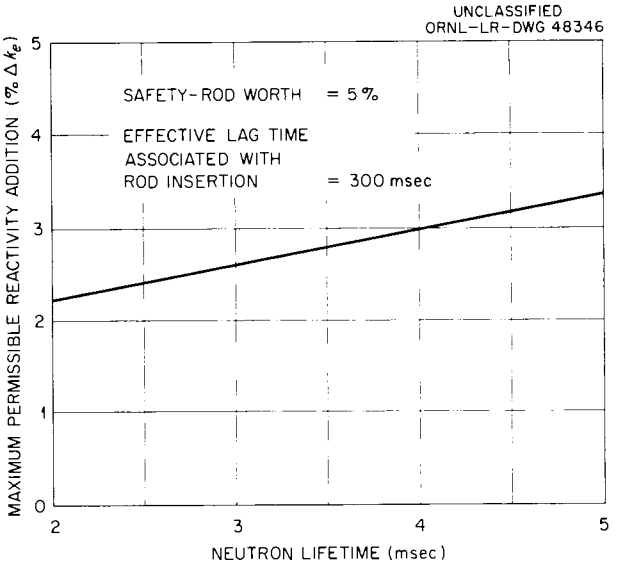

Fig. 6. Maximum Permissible Reactivity Addition as a Function of Prompt-Neutron Lifetime.

Effect of Power Failure

The combination of the large inertia of the fans and the natural convection of the air causes power failure to be relatively unimportant with respect to reactor safety. The safety rods were assumed to be interlocked with the fans in such a manner that a scram begins when the coolant flow falls below 80% of the normal operating value. The experimentally determined air flow rate[13] as a function of time following fan power failure is shown in Fig. 7; in calculating fuel temperatures, this relation was approximated by the equation

$$\frac{W(t)}{W(0)} = 1.0 - 0.9305\,(1 - e^{-0.0415t}) \quad . \qquad (23)$$

Approximately 6 sec is required for the flow to drop by 20% (from 3500 to 2800 cfm); after 8 sec the flow has dropped to 2700 cfm. Since the rods are fully inserted in less than 2 sec after initiation of rod drop, there is little decrease in cooling

[13] Intracompany (ORNL) correspondence from J. A. Cox to P. R. Kasten, 1958.

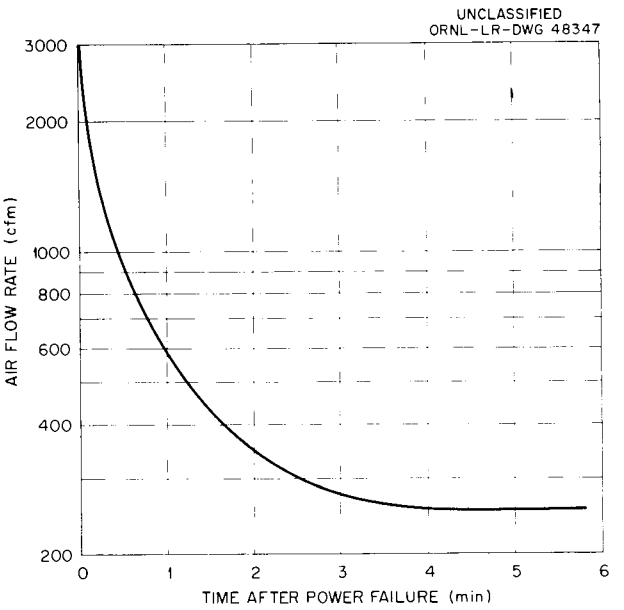

Fig. 7. Air Flow Rate After Fan Power Failure.

rate during the time of shutdown. Taking the safety-rod worth to be 5% and the time between power failure and safety-rod drop to be 6 sec, the maximum fuel temperature increased only 20°F before starting to decrease with time.

The natural-convection flow rate is 7% of the flow rate with the fans operating.[13] Using the Way-Wigner equation,[14] the decay heat generated after long fuel exposure and 4 min after a power failure is 0.134 Mw. Since this is less than 2% of rated power, natural convection would supply sufficient cooling.

The results of a loss of power to the cooling fans simultaneous with a reactivity addition are essentially no different from the results obtained with the fans operating. Due to the inertia associated with the fans, the safety rods are fully inserted before the coolant flow drops significantly.

Effect of Fuel Loading and Fuel Burnup on Maximum Fuel Temperature (Steady-State Conditions)

The maximum fuel temperature will be a function of fuel loading, fuel burnup, and power level. Because of the smaller reactor volume, the enriched reactor will operate at a higher power

[14] S. Glasstone, *Principles of Nuclear Reactor Engineering*, p 119, Van Nostrand, New York, 1955.

density than the natural-uranium reactor for a given power level, and will also have a higher maximum fuel temperature. Based on 0% burnup (cosine flux distribution), the maximum fuel-element temperatures were calculated [using Eqs. (1)–(6)] for the enriched- and natural-uranium reactor as a function of power level; the results are given in Fig. 8.

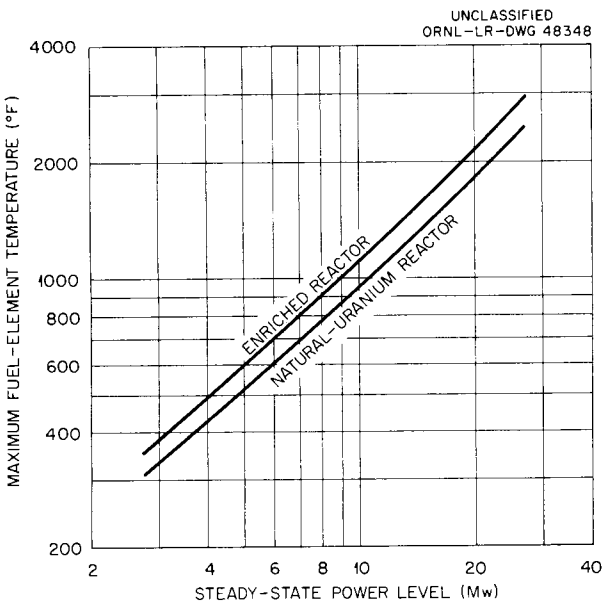

Fig. 8. Maximum Fuel-Element Surface Temperature vs Power Level (Steady-State Conditions, 0% Fuel Burnup).

It is planned that the enriched reactor will operate with fuel burnups of about 40%. Under these circumstances the power distribution will be a function of fuel loading, and the maximum steady-state fuel temperature as a function of power level will not necessarily correspond to that given in Fig. 8.

In order to achieve 40% burnup in all fuel elements, some scheme of fuel movement will be necessary. The simplest method would be to introduce new fuel at the front face of the reactor and remove the expended fuel from the rear face (push-through fueling, as in the present operative procedure). Another method would be to introduce fresh fuel at both the front and rear faces in adjacent channels and gradually push the fuel through the reactor (countercurrent fueling). A third method would introduce fresh fuel at both the front and rear faces (in the same fuel channel), with removal at the center of the reactor. A fourth would be the reverse of the third, with fresh fuel inserted in the center and removed at the ends. Using the methods of Corngold,[5–8] the flux and power distributions for 0 and 40% fuel burnup were obtained for the above loading schemes. Figure 9 gives the flux and power distributions for push-through fueling for 40% fuel burnup, and also the flux and power distribution for 0% burnup for any fueling scheme. So long as the burnup is not too great (40% was not too great), countercurrent fueling leads essentially to a cosine flux distribution, as indicated in Fig. 9. With countercurrent fueling, the average power distribution would be symmetrical but flatter than a cosine; the power distribution in a given channel, however, would be skewed as shown in Fig. 10, and it is this distribution which determines the maximum fuel temperature. Figure 11 gives the power distribution for 40% fuel burnup and two fueling methods: in one, fresh fuel enters the front and rear faces of the reactor and is discharged from the center of the reactor; in the other, fresh fuel enters the center and is discharged at both faces of the reactor. The latter fueling scheme leads to a peaked power and flux distribution and results in a relatively small reactor volume in which high thermal fluxes would be available; this fueling scheme was not considered further because of the above undesirable feature.

The power distributions for the various loading schemes were used to calculate the steady-state fuel-temperature distribution, with the reactor at 7 Mw and the fuel at 40% burnup at the time of removal from the reactor. These results are given in Fig. 12; the case of 0% burnup is shown for comparison. With a cosine power distribution (0% burnup), the maximum fuel-surface temperature was calculated to be 800°F; this peak value occurred 12 ft from the air-intake face (the incoming air temperature was 100°F, the exit air temperature was 610°F, and the air flow rate was 10 lb/min per channel). For 40% burnup, with new fuel added at one end and progressively pushed through the reactor, the maximum fuel-surface temperature was 950°F; this peak value occurred about midway through the reactor. Not

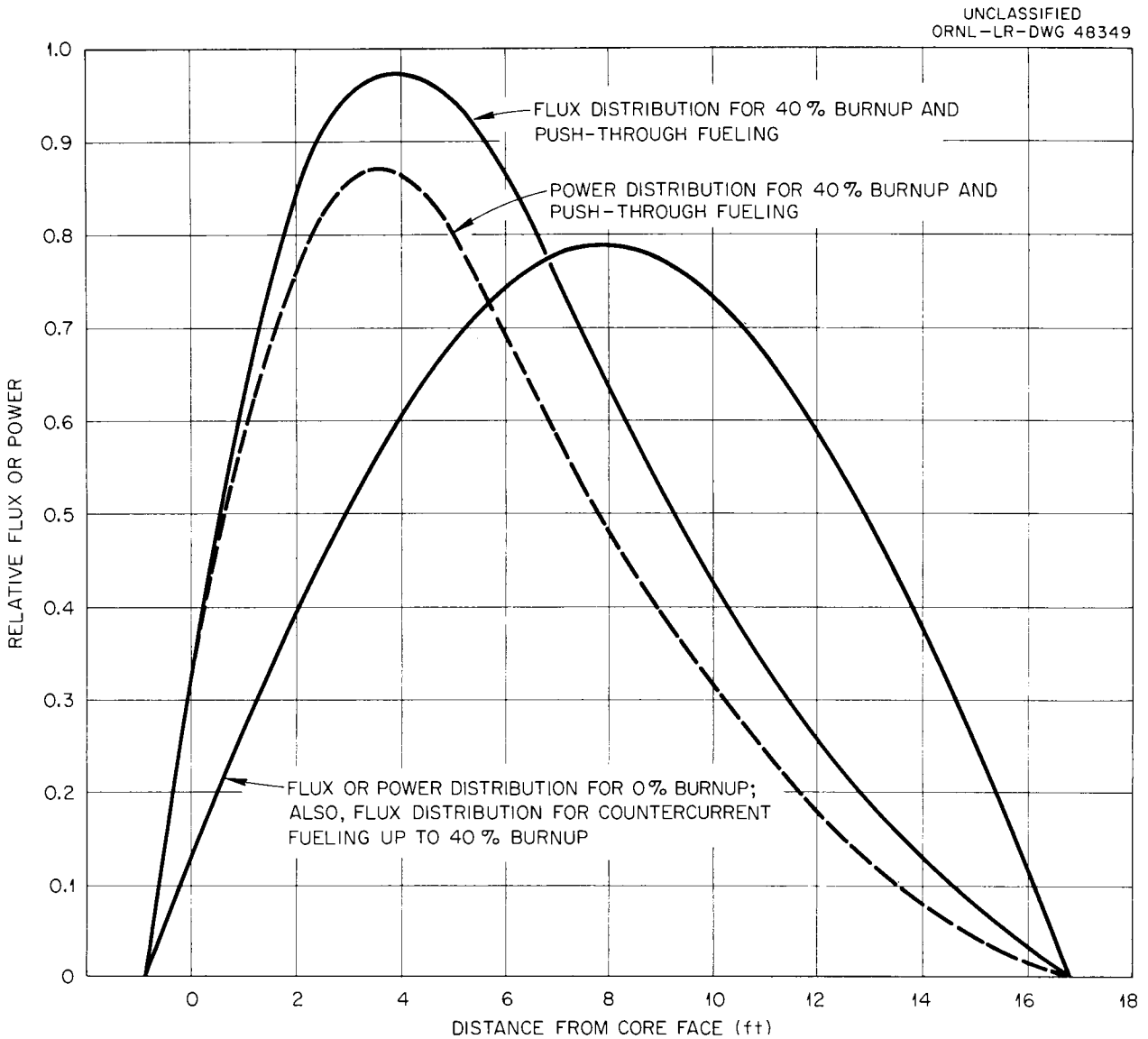

Fig. 9. Flux and Power Distributions for Push-Through and Countercurrent Fueling.

only does this type of loading increase the maximum fuel temperature, but also the maximum temperature during an excursion would increase more rapidly with time for a given Δk_e, relative to the rate of increase if the flux were cosine in distribution.

By using countercurrent fuel loading (adding new fuel at both ends of the reactor in alternate channels, and progressively pushing the fuel through the core), a cosine flux distribution is again attained. Even so, with burnup, there will be a greater power generation rate nearer the faces of the reactor relative to the generation rate for 0% burnup. For 40% burnup with counterfueling, the maximum fuel temperature was 850°F, and occurred 14 ft from the inlet-air face (this would be in the central channel, with fuel added to the face opposite the inlet-air face). If fresh fuel enters both faces of the reactor and is removed at the center after 40% burnup, there is a double temperature peak; as shown in Fig. 12, the maximum fuel temperature would be ~980°F.

For a power level of 4 Mw, the maximum steady-state fuel temperature would be ~490°F for 0% burnup. If 40% burnup took place and fuel were inserted into the reactor at the front face and

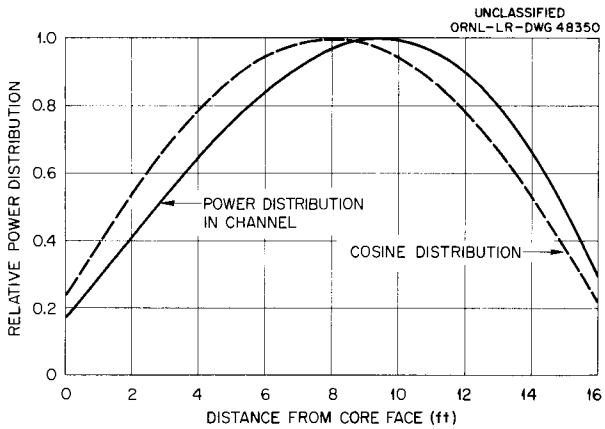

Fig. 10. Power Distribution in Channel Fueled from Rear Face of Reactor for Countercurrent Fueling with 40% Burnup.

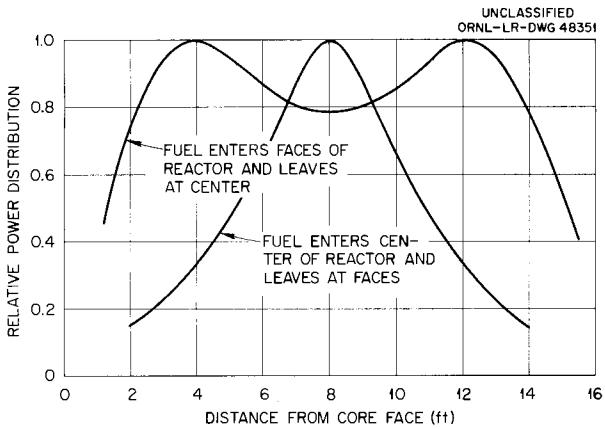

Fig. 11. Power Distribution for Two Fueling Schemes and 40% Burnup.

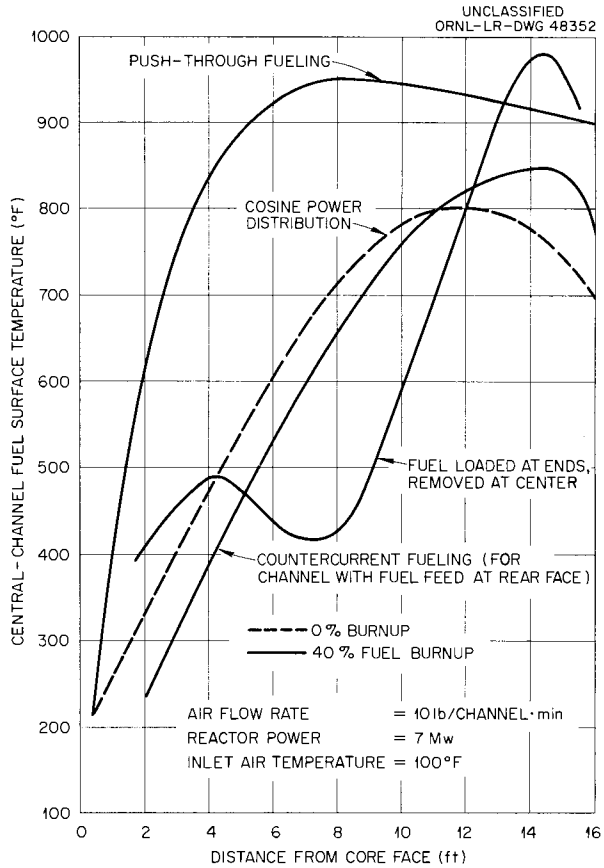

Fig. 12. Steady-State Temperature Distributions in Control Channel for Various Fuel-Loading Schemes (Reactor Power = 7 Mw, Inlet Air Temperature = 100°F, Air Flow Rate = 10 lb·channel^{-1}·min^{-1}).

removed at the rear, the maximum surface temperature would be 600 to 650°F. With fuel loaded countercurrently and removed after 40% burnup, the maximum surface temperature would be less than 550°F.

DISCUSSION

The results for the maximum permissible reactivity addition were obtained by using a cosine-shaped power distribution; with 40% burnup per fuel pass through the reactor, such a distribution is not readily attained. The results in Fig. 12 indicate that countercurrent fueling is the desirable method; with this type of fueling, the flux distribution essentially retains a cosine distribution, but the average power distribution is somewhat flatter than a cosine function. Under such circumstances the maximum permissible reactivity addition will not be quite as large as for the case of a cosine power distribution, since the maximum steady-state fuel temperature given in Fig. 12 would be 850 instead of 800°F. This is primarily an academic point under the conditions considered, as will be shown. First, the maximum fuel temperature during an excursion tends to occur at the point of maximum power density; thus in comparing the "cosine" and "countercurrent" curves in Fig. 12, the temperatures should be evaluated at the position at which the maximum temperature occurs during a reactivity excursion.

However, the 50°F difference has not increased at the point of maximum power generation for the two cases; thus, if 1220°F is taken as the maximum permissible fuel temperature with a cosine power distribution, the results based on a cosine

power distribution but a maximum permissible fuel temperature of 1170°F would apply to the countercurrent fueling method. With a 50°F difference, the maximum permissible reactivity addition does not change significantly. This is indicated in Fig. 2; with an effective lag time of 100 msec, the maximum temperature increased 55°F (from 1047 to 1102°F) when the reactivity addition increased from 3.5 to 3.6%; for a lag time of 500 msec, the maximum temperature increased 135°F (from 1072 to 1207°F) when the reactivity increased from 2.2 to 2.4%. Thus the use of countercurrent fueling decreases the maximum permissible reactivity addition less than 0.1% Δk_e in comparison with the 0% burnup case.

The reactivity worth of the rods which will respond during a reactivity accident cannot be specified with complete accuracy. However, by using additional safety rods, the total worth of safety rods could easily be more than 8% Δk_e (in the enriched reactor). Assuming that only 50% of their worth could be guaranteed under all circumstances, the results in Figs. 4 and 5 indicate a maximum permissible reactivity addition of 2.5% if the rods were initially at the edge of the core (placing the rods within the core would increase the maximum permissible reactivity addition but would decrease the number of neutrons available for experiments, and is therefore undesirable). Taking into consideration countercurrent fueling with 40% fuel burnup, a value of 2.4% is obtained.

In this study only instantaneous reactivity additions were considered, with the reactor initially at 7 Mw. If the power level were initially lower than 7 Mw, the maximum permissible reactivity addition would be slightly greater. At the lower initial powers, a greater fuel-temperature increase could be tolerated because the initial temperature would be lower. For circumstances in which reactivity is added as a rate function, the equivalent instantaneous reactivity addition can be obtained from the rate addition and the initial power of the reactor.

The maximum credible reactivity addition has not been evaluated here; it would be a function of the experiment materials present in the reactor. If all the enriched uranium were to melt and form a concentrated pool of uranium, large amounts of reactivity could be involved; however, this possibility appears to be very remote. If the fuel in several channels were to melt, the reactivity addition itself would not constitute a problem so long as the safety rods functioned, even though the uranium were to concentrate in the center of the reactor. Furthermore, so long as the fans are operating, any melted uranium would tend to be blown out of the reactor since the fuel which would melt first would be near the rear face of the reactor (countercurrent fueling, 40% burnup). The probability that more than a few fuel channels will be blocked to air flow without immediate reactor shutdown appears to be very low; blockage of a few channels could conceivably lead to release of fission fragments into the air stream, but would not involve ultimate reactor safety. Also, blockage of air flow would be no more probable with enriched fuel than it is at the present time. So long as the safety rods function as prescribed, the enriched system appears to be protected against fuel melting to at least the same extent as the present natural-uranium system, which has been operated safely for more than 15 years.

ACKNOWLEDGMENT

During the course of this work, assistance was given by several people; in particular, C. D. Cagle furnished information concerning reactor operation, D. K. Holmes provided results of nuclear calculations, and H. C. Claiborne, A. Fromhold, and B. E. Prince aided in formulating and checking out the equations and numerical procedures employed. Appreciation is also expressed to C. D. Cagle and W. K. Ergen for reviewing this report.

Appendix A

BACKGROUND INFORMATION ON NATURAL-URANIUM ORNL GRAPHITE REACTOR[15]

The ORNL Graphite Reactor consists of natural-uranium slugs in a 24-ft graphite cube enclosed in a concrete shield with inside dimensions of approximately 33 ft long, 24 ft wide, and 25 ft high. Between each side of the graphite-uranium matrix and the inside wall of the shield is a 1-in. space packed with asbestos rope. The space above the graphite is blocked by a steel beam. A 3-ft space is provided in front of the graphite as a chamber for the inlet air. This space is traversed by 1.25-in.-dia pipes. One end of each pipe fits into one of the 1252 holes through the front shielding, and the other end rests in one of the 1.75-in. square holes in the graphite. A 0.5- by 6-in. slot is cut in each of the pipes, and cooling air enters through the slots and also through the annuli around the 1.25-in. pipes in the 1.75-in. channels. The natural-uranium slugs are about 4 in. long and 1.1 in. in diameter and are jacketed with 2S aluminum (the jacketed slugs have an outside diameter of about 1.17 in. and a length of about 4.1 in.). The fuel channels are on a nominal 8-in. lattice spacing and are 24 ft long. A channel normally may contain from 30 to 68 slugs.

The reactor is cooled by pumping air through the 1.75-in. square channels. This is accomplished by two 70-in. static-suction-pressure fans, which are operated in parallel. The fans have individually rated capacities of 60,000 cfm at 150°F when drawing 55 in. of water suction. A small steam-engine-driven fan of 5000 cfm capacity is also available for operation under emergency conditions.

In 1944, it was calculated that the optimum pile loading would be from 40 to 45 tons of uranium. This loading would have 44 slugs in a channel and 710 loaded channels. With a maximum metal temperature of 200°C and inlet air at 75 to 80°C, this loading resulted in a reactor power of about 3000 kw. In June 1946, the average power was raised to 3400 kw. At that time, the reactor contained 49.1 tons of uranium, with 438 channels containing 50 slugs, 306 channels containing 44 slugs, and 90 channels containing about 35 slugs. The air inlet temperature was 25°C, and the outlet temperature was 90°C. The air rate into the pile was 6500 lb/min, while the air rate into the fans was 7200 lb/min; air leakage through the various openings in the shield and air leakage around the pile accounted for the difference between inlet and outlet flow rates. A 36 mm Hg pressure drop existed across the pile, and the total pressure drop in the system was 78 mm Hg. The maximum metal temperature was 245°C, the average metal temperature about 120°C, the maximum graphite temperature about 150°C, and the average graphite temperature about 90°C. By introducing a spray of cold water into the air before the fans, the air flow could be increased enough to raise the power 25% without increasing the maximum metal and air temperatures.

Presently, the reactor has a natural-uranium loading of about 54 tons; there are 830 loaded channels containing an average of about 45 slugs per channel (effective channel length is about 18 ft). The capacity of the two large fans is about 110,000 cfm, or 6.5 lb of air per channel per minute. The pressure drops across the various parts of the reactor system were measured in April of 1956, with the following results (in inches of water): static pressure measured at maximum fan suction, −53; pressure drop across reactor, 15.5; static pressure measured at bridge tube, −13; static pressure measured at rear plenum, −28.8; static pressure measured at discharge seal pit, +7.

The maximum slug temperature is limited to values less than those associated with a high rate of slug failures (temperatures less than ~250°C). For a given reactor power, the temperatures of the slugs are determined primarily by the effective heat transfer coefficients associated with the cooling fluid and the metal-graphite contacts (heat transfer in the longitudinal direction was found to be negligible). The metal-to-fluid heat transfer coefficient was calculated by conventional means, assuming that the effective heat transfer area was three-fourths that of the slug surface. At an air temperature of 58°C, an equivalent slug diameter of 0.0762 ft, and a free-flow area of 2 in.^{2} per channel, the coefficient

[15] L. B. Borst (ed.), *Graphite Uranium Production Piles*, PPR-4 (1951).

was found to be

$$h = 4.59G^{0.8} = 142W^{0.8} \quad , \qquad \text{(A-1)}$$

where

h = heat transfer coefficient, $\text{Btu}\cdot\text{hr}^{-1}\cdot\text{ft}^{-2}\cdot(^\circ\text{F})^{-1}$,

G = mass velocity, lb of air per square foot of flow area per second,

W = mass flow rate, lb of air per second.

Under normal flow conditions, the above formulas give h a value of about 26 $\text{Btu}\cdot\text{hr}^{-1}\cdot\text{ft}^{-2}\cdot(^\circ\text{F})^{-1}$. The value for h decreased with increasing air temperature by about 0.1% per degree C of air-temperature rise.

A considerable portion of the heat from the slug is conducted to the graphite through the metal-to-graphite contact area. The contact coefficient, k_c, was evaluated experimentally and found to be 4 $\text{Btu}\cdot\text{hr}^{-1}\cdot(^\circ\text{F})^{-1}\cdot\text{ft}^{-1}$ (this value apparently includes both radiation and conduction effects). On the same basis, the value for h, the convective coefficient, is equivalent to about 7 $\text{Btu}\cdot\text{hr}^{-1}\cdot(^\circ\text{F})^{-1}\cdot\text{ft}^{-1}$. Thus, under typical conditions, about one-third of the heat is transferred through the metal-to-graphite contact area, when the air velocity is about 150 fps in the metal-filled part of the pile. At higher air velocities the "contact" transfer becomes less important.

Reactor control and safety are maintained by regulating, shim, and safety rods. Originally, there were two horizontal regulating rods (R rods), four horizontal shim rods (S rods), four vertical rods (X rods), and two vertical safety tubes. Presently, there are two horizontal shim rods and three vertical safety rods, with the other controls remaining as stated above. Certain conditions have to be met before rods can be moved: R rods can be moved at any time; S rods can be moved out only after X rods are withdrawn.

The regulating rods are boron-steel bars (1.3% boron) $1\frac{3}{4}$ in. wide and 19 ft long. When they are fully inserted within the reactor, the far ends are 5 ft from the opposite face of the pile. A rod speed of 6.9 ips can be obtained by the No. 1 regulating rod. The No. 2 regulating rod has a speed of 6 ips on high speed and 0.06 ips on low speed. Since 1 inhour in the Graphite Reactor is equivalent to a Δk_e of 2.7×10^{-5}, the two regulating rods have maximum reactivity worths of about 0.0057 and 0.0050 Δk_e, respectively. The two regulating rods are placed 28 in. directly above and below the center of the pile. Based on a cosine flux distribution, a regulating rod should have 75% of the reactivity effect of a shim rod, which is in good agreement with experimental values.

The original four shim rods entered the pile on the north side at the corners of a 4-ft square centered on the horizontal axis of the pile. Presently, only two shim rods are used. The other two were found to be unnecessary and were removed to afford more experimental facilities. The two shim rods can be inserted in 4 sec under emergency conditions; they are inserted by hydraulic pressure if a power failure occurs. These rods also are made from boron steel (1.3% boron) and are bars $1\frac{3}{4}$ in. on a side. Normally, they are moved inward at a rate of 6 ips or outward at 3 ips. Only a single rod may be withdrawn at any one time. Each of the rods has a maximum reactivity worth of 235 inhours (0.0064 Δk_e based on 2.7×10^{-5} Δk_e per inhour); the total worth of the two rods is about 0.0128 Δk_e.

The original four vertical safety rods were made of boron steel (1.3% boron) and were cylinders 8 ft long and 1.5 in. in diameter. These rods had a reactivity worth of about 0.0027 Δk_e each, or a total worth of about 0.0108 Δk_e. Presently, there are three safety rods each constructed in the form of a 3.5-in.-square shell 0.25 in. thick. These rods contain cadmium, and each has a reactivity worth of about 185 inhours (0.005 Δk_e) – a total worth for all three safety rods of about 0.015 Δk_e. These rods are 8 ft long, and when inserted within the reactor, the midpoints are on the center line of the pile. These rods enter the pile through the action of gravitational force.

Safety rods are actuated when the power fails, when the neutron flux becomes too high, when the cooling air becomes too hot, when the metal temperature rises too high, or when the cooling air flow fails.

The two cylindrical safety tubes were made of aluminum and were $1\frac{3}{4}$ in. in diameter; they are no longer in use.

Also aiding ultimate reactor safety, in addition to the control rods, is the negative temperature

coefficient of reactivity. The temperature coefficient for the whole pile in vacuum was calculated to be −1.4 inhours/°C. A total temperature coefficient of −1.1 inhours/°C was observed experimentally (based on a reference temperature at the central or hottest slug). The metal temperature coefficient (based on central-slug temperatures) was −0.29 inhour/°C (-0.8×10^{-5} Δk_e/°C). The graphite temperature coefficient was −0.77 inhour/°C (-2.1×10^{-5} Δk_e/°C).

Appendix B

STUDIES BY BROOKHAVEN NATIONAL LABORATORY CONCERNING CHANGE OF FUEL FROM NATURAL TO ENRICHED URANIUM[16]

Brookhaven National Laboratory is in the process of changing the fuel in their graphite reactor from natural uranium to highly enriched uranium. This conversion is taking place in steps, but eventually all the fuel will be highly enriched in U^{235} (ref 16). Since many of the problems associated with conversion of their reactor are the same as, or similar to, those associated with conversion of the ORNL Graphite Reactor, a review of BNL's studies and results as presented in the referenced reports is given here (a summary hazards report on the enriched BNL reactor is given in ref 2).

The reactor consists of a 25-ft cube of graphite penetrated by a square array of 1369 channels. The channels have a circular cross section 2.67 in. in diameter and are spaced on 8-in. centers. A central air gap is used, which is about 8 cm wide. The pile has enough excess reactivity to permit sustained operation at a maximum flux of 5×10^{12}. The maximum fin temperature (old elements) is about 380°C, and the exit air temperature is about 170°C. Air flow is 1.25×10^6 lb/hr, with a reactor pressure drop of 55 mm Hg. The reactor has a heat output of 25 Mw under the above conditions with about 675 of the channels loaded.

It is anticipated that the use of enriched fuel will lead to higher neutron fluxes and fewer fuel failures. The enriched reactor will have about the same active volume as the natural-uranium system by loading 580 channels with 16 ft of fuel elements. At 40% burnup under steady-state conditions, each loaded channel will contain about 96 g of uranium for a total loading of 57 kg. Initially, a freshly loaded channel will contain 120 g of uranium. The fuel elements will be made of plates of the MTR type. The active fuel is a dilute alloy containing about 8 wt % uranium; the active-fuel thickness is 20 mils, with 20 mils of cladding on either side. Each plate contains 5.1 g of uranium, so that each three-plate fuel element (2 ft long) contains 15.3 g of uranium. It is expected that a flux of about 2.5×10^{13} will be obtained at 20 Mw. The new fuel element has about 70% more heat transfer area per unit length than the old element. Fuel temperature is not to exceed 350°C. Total air flow will be 1.1×10^6 lb/hr. Pressure drop will be 55 mm Hg, and the exit air temperature about 180°C.

With enriched fuel in the BNL reactor, a thermal-neutron flux of 2×10^{13} should be available at a reactor power of 20 thermal Mw.[17] The active volume would correspond to a 17- by 17-ft right-circular cylinder. It is anticipated that a burnup of 40% will be achieved; under this condition it is estimated that the radial flux distribution would be represented by a Bessel function and that the axial flux would be peaked in the north half of the reactor.[18] At 20 Mw, the inlet air would leave the reactor at 200°C if it entered at 38°C, while the maximum metal temperature would be less than 300°C if the air flow were

[16] Since the time of this writing in 1958, the BNL graphite reactor has been completely converted to highly enriched uranium fuel.

[17] T. V. Sheehan, *Proposed Changes in Operation of BNL Reactor*, BNL memo dated Feb. 3, 1954.

[18] J. Chernick, *Theoretical Problems of the Modified BNL Reactor*, BNL Log No. C-7988 (Apr. 20, 1954).

970,000 lb/hr.[17] Pressure drop across the pile would be 21.5 in. of water, and the required pumping power would be 1200 to 1500 hp.

A comparison of the natural-uranium BNL system with the enriched system is given in Table B-1.

The use of highly enriched uranium will change the uranium inventory and loading cost. In addition to the cost of uranium itself, there will be fabrication and fuel-reprocessing charges. It was estimated[19] that for fuel plates containing about 7 g of U^{235} per foot (U^{235} in an aluminum matrix 20 mils thick, clad with 20 mils of aluminum) fabrication costs would be about $10.50 per foot. Based on 40% fuel burnup and $6 per gram for U^{235} processing, processing costs would be about $30 per foot. The initial U^{235} inventory of about 60 kg of U^{235} would cost about $1,000,000 (at $17 per gram of U^{235}). Fuel fabrications of this inventory would cost about $100,000. At 20 Mw and 40% fuel burnup, about 18.9 kg of U^{235} would be required for the reactor per year; fabrication and chemical processing of this fuel would cost about $400,000 per year.

Since an enriched system would not be poisoned by large amounts of U^{238}, less fissionable fuel would be required in comparison to requirements in a natural-uranium system. Specifically, about 60 kg of uranium enriched to 93% U^{235} would be required if the working volume were equal to that of the natural-uranium loading.[20] The reactor would operate at 10 Mw with a maximum thermal flux somewhat greater than 10^{13}. With the enriched elements an air flow rate of 5000 lb/hr should be adequate at 10 Mw. However, at a flux of 10^{13}, highly enriched uranium systems rapidly lose reactivity unless sufficient reactivity control is present within the reactor system. To achieve 10% burnup in a cylindrical reactor with height equal to diameter, the equivalent diameter must increase from 13.5 to 17 ft in order to maintain criticality. For 20% burnup the uranium requirements for criticality are 12% more than for 10% burnup.[21]

As indicated above, the enriched reactor will have different nuclear properties than the natural-uranium-fueled system.[2,20] For example, the thermal diffusion length and the neutron lifetime will be longer. With highly enriched elements the reactor will be essentially homogeneous insofar as reactor calculations are concerned, since the fuel-element and the moderator disadvantage factors are practically unity. Accounting for voids, the Fermi age is about 410 cm^2 and the square of the thermal diffusion length is about 3700 cm^2 at operating temperature. The prompt-neutron lifetime is 1.6 msec in the natural-uranium reactor but will be 4 msec in the enriched system. For a maximum flux of 2 to 3×10^{13} there is no problem of xenon override after shutdown; the loss in reactivity due to equilibrium xenon will be about 0.014 Δk_e. Experiments were rated at 0.0075 Δk_e in the old reactor; in the

[19] D. H. Guerinsky, *Enriched Fuel Elements for BNL Pile*, BNL Log No. C-7867, discussions with E. Boyle, J. E. Cunningham, and J. H. Erwin, Feb. 25, 1954.

[20] J. Chernick, *Preliminary Study of Highly Enriched Fuel Elements for BNL Reactor*, BNL Log No. C-7796 (Feb. 3, 1954).

[21] J. Chernick, *Composition of Sample Enriched-Fuel Elements for a BNL Reactor*, BNL Log No. C-8562 (Dec. 21, 1954).

Table B-1. Comparison of BNL Reactor Characteristics*

Characteristic	Natural-Uranium System	Enriched-Uranium System
Fuel loading, kg of U^{235}	375	60
Neutron flux	5×10^{12}	2×10^{13}
Power level, Mw	28	20
Air flow, lb/hr	1.5×10^6	8×10^5
Pump power, hp	7500	1500
U^{236} production, kg/year		1.7

*M. Fox, *Proposal to Convert Fuel Loading in the BNL Reactor*, BNL Log No. C-8110 (June 21, 1954).

enriched reactor they will be worth about 0.023 Δk_e. In all, the enriched reactor will possess about 5.1% Δk_e in the clean cold state. In analogous fashion, the reactivity worth of the present control rods will be appreciably greater in the enriched reactor. In the natural-uranium reactor the control rods are estimated to be worth a total Δk_e of 0.0435; the worth of these rods in the enriched reactor was calculated to be 0.148 Δk_e.[22]

In the enriched reactor, the effects of temperature on the effective η of the fuel, the flux distribution, and the neutron leakage are the chief contributing factors[23] to the temperature coefficient of reactivity. Of these, the leakage effect is the most important. The value of $(1/\eta)(d\eta/dT)$ was assumed to be 1.0×10^{-5} per degree C. The flattening of the intracell flux distribution with increasing temperature changes the disadvantage factor so that the thermal utilization, f, changes with temperature. In the temperature range from 350 to 450°C, $(1/f)(df/dT)$ varied from 7 to 10×10^{-6} per degree C. The temperature variation of $B^2(dM^2/dT)$ for the enriched reactor varied from 1.75×10^{-4} per degree C at 350°C to 1.5×10^{-4} per degree C at 450°C. The total steady-state temperature coefficient of reactivity was calculated to be -1.52×10^{-4} per degree C at 350°C and 1.32×10^{-4} per degree C at 450°C.

During a power transient the associated temperature rise is restricted in large measure to the fuel element itself. Under these circumstances the metal temperature coefficient of reactivity (and also the temperature coefficient of η) responds promptly, but neutron leakage does not change immediately. The metal coefficient in the natural-uranium BNL reactor is about -2×10^{-5} Δk_e per degree C. With the enriched fuel elements the metal temperature coefficient appears to be practically zero. If some 240 g of U^{238} were added to the 2-ft fuel elements containing 13 g of U^{235}, the calculated metal coefficient[24] would be -0.3×10^{-5} per degree C.

The safety of a reactor is a function of the magnitude of a possible reactivity addition and the response of the reactor safety system. Studies at BNL have considered the reactivity changes which might be associated with rod movement and response in the enriched reactor.[25] Specifically, the reactor period (at the time of shutdown initiation) which could be tolerated without causing melting of the fuel elements was determined. The following events were assumed to take place: the reactor is brought to power by withdrawing control rods out of the reactor; as the reactor becomes critical, one or two rods are still in the reactor, and these are removed at the same rate as the previous rods, resulting in a positive reactor period; at a reactor power of 20 Mw, a safety device initiates a shutdown; it is assumed that all rods outside the reactor move into the reactor at time of shutdown; the control rods accelerate and reach their maximum velocity of 5.5 fps in 0.5 sec; in this first period of time the rods have moved 1.38 ft; they will move at 5.5 fps for the next 24 ft, causing the reactor to become subcritical.

During these events it was assumed that the power-removal capability of the coolant corresponded to 20-Mw operation. The results indicate that a period shorter than 0.57 sec (at automatic shutdown) could result in melting of the central fuel element. This period corresponds to an equivalent reactivity addition of 0.75% Δk_e. [The control rods in the enriched reactor will have a withdrawal speed of 0.1 fps. This withdrawal speed along with the above postulated series of events corresponds to a minimum period of about 3 sec if automatic shutdown is initiated at a power level of 30 Mw. Control-rod reactivity response (when inserting the rods) will be quicker if the rods are not fully withdrawn; in the enriched reactor it is planned to always insert the rods from an "advanced" position.]

The addition of water to the reactor could conceivably add reactivity; however, based on calculation and "experiments," it appears that such an event tends to shut the reactor down. Water has accidently been introduced into the

[22] T. Auerbach, *Effectiveness of Control Rods in the Modified BNL Reactor*, BNL Log No. C-8418 (Oct. 29, 1954).

[23] T. Auerbach, *Uniform Temperature Coefficient in Graphite of the Enriched Reactor*, BNL Log No. C-8506 (Dec. 2, 1954).

[24] N. Corngold, *The Metal Temperature Coefficient of the Modified BNL Reactor*, BNL Log No. C-8739 (Feb. 16, 1955).

[25] B. Mozer, *Transient Power of Rods in the Modified BNL Reactor and Control Rod Theory*, BNL Log No. C-8821 (Mar. 23, 1955); *Kinetic Behavior of the Modified BNL Reactor*, BNL Log No. C-8640 (Jan. 12, 1955).

natural-uranium reactor on several occasions, and its reactivity effect was strongly and unambiguously negative.

A limiting consideration in operating the enriched reactor is the temperature and radioactivity-retention ability of the fuel elements. The fuel concentration is closely related to the maximum permissible fuel temperature for a given reactor power; increasing the U^{235} concentration will normally increase the power density and the fuel temperature. The proposed enriched fuel elements initially contain about 7.5 g of U^{235} per foot of element, with each element being 2 ft long. At a reactor power of 20 Mw this corresponds to a power density such that the maximum metal temperature is not excessive, as verified by calculations and experiments.[26,27]

To clarify heat transfer limitations, the heat production rates and temperatures were calculated for elements containing one, three, and five times a fuel concentration of 7.5 g of U^{235} per foot. Considering 16 ft of fuel elements in the central channel, the respective heat-production rates for the differently loaded elements were computed to be 11.5, 38.4, and 66.6 $cal \cdot cm^{-1} \cdot sec^{-1}$, corresponding to maximum temperature differences between the metal and inlet air of 80, 260, and 450°C. In order to check these calculations, experimental measurements were made of the surface temperature of enriched elements placed in the natural-uranium reactor. Air temperature measurements were made at positions 78 and 130 in. from the gap (the latter position was near the plenum chamber). Fuel elements with 11 thermocouples brazed on the elements and containing 25.2 g of U^{235} per foot were loaded in a channel; these thermocouples were placed at various positions from 6 to 66 in. from the air gap. The elements were loaded in channel B04S, which is a fourth channel from the central axis. The maximum temperature was 324°C, located 66 in. from the gap. Placing fuel elements containing about 33 g of U^{235} per foot in channel A40S (near the central axis) resulted in a maximum fuel temperature of 413°C with the reactor at 24 Mw; the enriched fuel elements were tested under rather severe weather conditions inasmuch as the temperature and humidity of the inlet air were above normal values. While operating at the above fuel-surface temperatures, no excessive activity was noted in the exit air.

The above results can be used to check values calculated for the enriched reactor. At a power level of 28 Mw in the natural-uranium reactor, the heat production rate in the fuel elements placed in the B40S channel would be about equivalent to that in an enriched reactor (20 Mw) having a fuel loading of about 21 g of U^{235} per foot. Agreement between experimental and calculated values thus appears good; therefore a fuel loading as high as 21 g of U^{235} per foot does not appear to result in an excessive fuel-element temperature.

If a loss of coolant flow occurred during reactor operation, it may be possible that the temperature of enriched fuel elements would become excessively high due to their relatively low heat capacity. However, results obtained from an inadvertent "experiment" indicate that no fuel temperature increase follows loss of coolant flow; this is due to the associated reactor scram and convective air flow.[28] The "experiment" occurred with enriched fuel elements (loaded to about 21 g of U^{235} per foot) in the B40S channel and was caused by a power failure. The reactor was originally at 24 Mw and was automatically scrammed when the power failure occurred. The enriched-fuel temperature decreased sharply following the scram, reached a minimum value 2 to 3 min after shutdown, and then stabilized at slightly higher temperatures (10 to 15°C above the minimum) within 10 to 12 min after shutdown. The initial fuel temperature in the channel ranged from 175 to 325°C and dropped down to 50 to 110°C following the shutdown. There was no visual evidence of heat damage to the elements.

If melting of the enriched fuel elements occurred, fission gases may be released from the fuel into the air stream. In order to determine whether fission gases would be released, two

[26] N. Corngold, *Heat Production in Test Elements for the New BNL Reactor*, BNL Log No. C-8588 (Jan. 3, 1955).

[27] R. A. Meyer, *Temperature Measurements of the Enriched Fuel Elements in the BNL Pile*, BNL Metallurgy Memo No. 555 (Aug. 11, 1955).

[28] R. H. Engelken, *Enriched Fuel Element Temperature Measurements Following a Simultaneous Reactor Scram and Loss of Coolant Flow*, memo to R. W. Powell (Apr. 11, 1956).

types of fuel plates were tested at high temperatures.[29] One type was constructed by Sylvania and consisted of an active core of normal UO_2 dispersed in aluminum. The plates were 60 mils thick, including 20 mils of aluminum cladding on either side of the fuel. They were 16.5 in. long and 2.7 in. wide, and contained about 25 g of uranium per linear foot. Some ORNL-type plates were also tested; these consisted of an active core of depleted uranium alloyed with aluminum (20 mils thick), clad with 20 mils of aluminum. These plates were 25.5 in. long and 2.9 in. wide, and they corresponded to about 13 g of depleted uranium per linear foot.

The fuel plates were exposed in the BNL reactor and then heated in a furnace. The ORNL-type fuel melted at 655°C, while the Sylvania type melted at 653°C. With the ORNL plates it was found that the gamma background remained constant at 11 mr/hr until a temperature of about 600°C was attained [the control sample (at room temperature) indicated a radioactivity decay of 15% during this period]. The gas removed from the cold traps appeared to be Xe^{135}. Steady increases in activity release were noted from 670°C upward, with a particularly sharp increase at temperatures above 800°C. The ORNL experiments indicated that, upon fusion, about 1% or less of the rare fission gases present in the element were liberated in an hour of heating in a normal atmosphere. Of the 1% release, almost all occurred as the element melted; about 0.01 to 0.1% of the iodine was released.

A portion of the Sylvania-type core fuel was exposed to the air during heating tests. Under this condition, temperatures of 375 to 400°C resulted in gas samples having activities approximately six times the background of the counter. Sharp increases in filter activity were noted above 800°C; these were associated with significant releases of fission gases. Particulate releases became more severe as temperature increased.

Experiments were also undertaken to determine whether molten enriched fuel elements would penetrate graphite,[30] in these, the uranium-aluminum fuel plates contained about 2% uranium. The results indicate that UAl_3 or U_3O_8-Al can penetrate into graphite if the molten fuel and graphite are in a vacuum or a very good reducing atmosphere. Under such conditions the pressure difference required to force fuel into graphite is from 0.5 to 2 psi. However, any conditions that result in a monolayer of aluminum oxide will severely hinder penetration; with oxide formation, pressure differences greater than 40 psi were required to force the molten fuel into graphite. Because of air cooling and a maximum pressure differential of about 1 to 2 psi, it is expected that no fuel penetration would occur in the reactor.

Exposure of the fuel elements to high temperatures and thermal cycling may cause the elements to deteriorate. A number of sandwich-type fuel plates were exposed for long periods of time at temperatures of 340, 400, and 450°C in static air. Several of these plates were sectioned to expose the core material to the hot air. After 13,000 hr of exposure, no buckling or separation of core from cladding had occurred; neither was there any significant change in length nor measurable weight gain; there was no migration of the core material to the surface of the plate; and there was no powdering of the core material when the core was exposed. Fuel elements were thermal cycled between 420 and 270°C and also between 400 and 40°C with no ill effects.

To investigate the effects of radiation damage, enriched fuel elements were exposed in the reactor at temperatures as high as 420°C; the maximum burnup was 5.5% of the fuel. No observable pitting, blistering, or surface defects were observed. Some of the elements were slightly warped, but this effect could be attributed to handling. At no time in a year of testing was there any indication that gaseous fission products were escaping from the enriched elements.

In the enriched reactor, the neutron leakage will be relatively higher than in the natural-uranium reactor; however, since the total power will be 20 Mw (compared with 28 Mw), the absolute leakage flux will be less than for the natural-uranium system, and so the present shielding will be adequate. Increasing the flux by a factor of 5 without a corresponding decrease in reactor volume will increase the production rate of radioactive argon; however, the present stack

[29] R. Engelken, *High Temperature Activity Release Experiments with New Type Fuel Plate*, memo to R. W. Powell (Apr. 5, 1955).

[30] R. A. Meyer, *Uranium-Aluminum Penetration into AGOT Grade Graphite*, memo to D. H. Guerinsky (Oct. 18, 1955).

height appears sufficient (under usual meterological conditions) for the proposed operating conditions.

The burnup of fuel and the fuel loading pattern will have appreciable effects on the flux and power distribution in the reactor. To measure these effects, the flux and power distributions obtained for various loading patterns and fuel burnups have been calculated for steady-state conditions in the enriched BNL reactor.[5-8]

Initial calculations[5] were based on one-group one-region diffusion theory, and neglected poisoning effects; movement of fuel through the reactor was assumed to be in one direction. For an average neutron flux of 10^{13} the reactor power would be 6.3 Mw, and it would take 0.58 year for fuel to pass through the reactor and have a burnup of 10%. About 10.6 kg of U^{235} would have been fed into the reactor at the loading face during this time. Corresponding numbers for a reactor power of 6.3 Mw and various fuel burnups are: 1.23 years and 11.2 kg for 20% burnup; 2.82 years and 13.0 kg for 40% burnup.

Thermal flux and power distributions were also calculated on the basis that fuel enters both faces of the reactor and is removed at the central gap after 20, 30, and 40% fuel burnups.[6,7] Because of the burnup, there is more peaking in the power distribution than in the flux distribution. At 40% burnup and a power level of 20.7 Mw, the average central thermal flux was 3×10^{13}. With a reflector the flux distribution was somewhat flatter, with the average central flux being about 2×10^{13}.

NOMENCLATURE

a, A	Cross-sectional area, ft^2
c	Specific heat, $Btu \cdot lb^{-1} \cdot (°F)^{-1}$
C_1	Maximum heat generation rate in fuel, $Btu \cdot hr^{-1} \cdot ft^{-3}$
C_2	Maximum heat generation rate in graphite, $Btu \cdot hr^{-1} \cdot ft^{-3}$
D	Diffusion coefficient, cm
g	Local acceleration of gravity, ft/sec^2
h	Heat transfer coefficient, $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot (°F)^{-1}$
j	Index to specify radial position within reactor
J_0	Bessel function of first kind of zero order
k	Thermal conductivity, $Btu \cdot hr^{-1} \cdot ft^{-1} \cdot (°F)^{-1}$
k_∞	Material multiplication constant
Δk_{sr}	Total negative reactivity added to reactor by safety rods
$\Delta k_{sr}(\theta)$	Reactivity worth of safety rod as a function of time after rod release (rod falls by gravity)
Δk_{sr}^T	Total reactivity worth of safety rod when fully inserted (initially at edge of core)
k_e	Effective multiplication constant
Δk_e	Instantaneous change in effective multiplication constant; reactivity change
$\partial k_e/\partial T$	Average temperature coefficient of reactor (uniform temperature change), $(°F)^{-1}$ or $(°C)^{-1}$
K	Effective conductivity from fuel directly to graphite, $Btu \cdot hr^{-1} \cdot (°F)^{-1} \cdot ft^{-1}$
l	Prompt-neutron lifetime, sec or msec
L	Reactor length; total length of safety rod; ft
m	Parameter associated with decrease of air flow rate with time following power failure, hr^{-1}

N^{25}	Effective concentration of U^{235} in reactor cell, atoms/cc
N^p	Concentration of fission products in reactor cell, assuming one effective fission product nuclide per fission; atoms/cc
p	Perimeter distance, ft
P	Reactor power, Btu/hr
Q	Heat-generation rate, $Btu \cdot hr^{-1} \cdot ft^{-3}$
r, R	Radial position within a cell, ft
R_1	Effective radius of reactor core, ft
t	Time, sec
T	Temperature, °F
ΔT	Temperature increase, °F
v	Velocity of fuel in longitudinal direction, cm/sec
V	Volume, ft^3
W	Coolant flow rate, lb/hr
ΔW	Change in W from forced-convection steady-state conditions to natural-convection steady-state conditions, lb/hr
$W(0)$	W evaluated under initial conditions, lb/hr
x	Distance from front face of reactor core, along longitudinal axis; ft or cm
β	Effective number of delayed neutrons per fission neutron
β_i	Effective number of delayed neutrons in ith group per fission neutron
δ	Initial insertion distance of safety rod, measured from edge of core, ft
ϵ	Reactor extrapolation distance, ft
θ	Time after safety-rod release, sec
λ_i	Decay constant of delayed-neutron precursor in ith group, hr^{-1}
μ_i	Latent power from ith delayed-neutron group, Btu-hr/hr
ρ	Density, lb/ft^3
Σ^p	Effective absorption cross section of fission products, cm^{-1}
Σ^T	Total absorption cross section (includes fuel, moderator, cladding, and fission products cross sections), cm^{-1}
Σ^{25}	Absorption cross section of U^{235}, cm^{-1}
Σ_f^{25}	Fission cross section of U^{235}, cm^{-1}
τ	Effective lag time associated with rod insertion, msec
ϕ	Thermal-neutron flux, $neutrons \cdot cm^{-2} \cdot sec^{-1}$

Subscripts

c	Pertains to graphite
cs	Pertains to surface of graphite
f	Pertains to fuel

i	Pertains to air contacting inner surface of fuel; ith kind
o	Pertains to air contacting outer surface of fuel
r	Pertains to reactor
sr	Pertains to safety rod

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