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UNITED STATES ATOMIC ENERGY COMMISSION

**SUMMARY HAZARDS REPORT ON ENRICHED
FUEL ELEMENT LOADING FOR THE BNL
RESEARCH REACTOR**

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May 24, 1956

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Summary Hazards Report on Enriched Fuel Element
Loading for the BNL Research Reactor

May 24, 1956

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Summary Hazards Report on Enriched Fuel Element
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I. Introduction

The Reactor Department of the Brookhaven National Laboratory has proposed⁽¹⁾ that the thermal neutron flux of the Brookhaven research pile be increased by replacing the present natural uranium fuel elements by new elements highly-enriched in U²³⁵. Theoretical analysis has shown that, as a result of the removal of the U²³⁸, the thermal neutron flux can be increased by a factor of four or five over the present flux; in addition, heat can be removed more efficiently from the new fuel element, and the increased thermal neutron flux can be obtained without increasing the cost of operating the pile*. Finally, the present helium leak detection system and its instrumentation can be eliminated.

The usefulness of the pile as a research tool would be greatly enhanced by the proposed change and the staff of the Laboratory considers the change to be highly desirable. At the present time, at least one-third of the entire Laboratory research program depends upon the use of the reactor facilities. In addition, a number of outside organizations are dependent on the BNL reactor in their research. Examples are the Westinghouse and General Electric Corporations - under contract to the U. S. Navy. In all, about a dozen industrial and academic research organizations are doing research at the reactor site. Also, some 150 shipments of irradiated objects are made to off-site institutions, each month.

Both the quality and the quantity of the research data produced depend,

*The report of reference (1) was in error with respect to power costs. It is now estimated that the savings at the presently proposed flux (3×10^{13}) will be far less than \$400,000 there estimated.

in a large measure, on the neutron flux available. The newer research reactors, such as MTR, the Argonne CP-5, and the Los Alamos and ORR types, all have fluxes in the range of 3 to 4×10^{13} and upwards as compared to the present BNL flux of 5×10^{12} . In the performance of most experiments, the time devoted to data taking is inversely proportional to the flux. For some experiments, there is an inverse square dependence upon the flux; e.g., (1) the determination of the cross sections of a radioactive nuclide where the flux is involved in both the production of the nuclide and the measurements, and (2) the production of radioactive nuclides involving second order effects such as successive neutron captures. No less important is the fact that some experiments which are impossible or marginal at low fluxes become possible of accomplishment at the higher fluxes. Thus, the relatively low flux of the Brookhaven reactor serves to handicap the research program here in comparison to programs at other sites. There is an impressive array of expensive research equipment at the Brookhaven reactor which would be advanced to a much more favorable position by an increase of the available flux.

The proposed conversion involves significant changes in the nuclear properties of the pile as well as a new use of the sandwich-type fuel element. It has, therefore, been necessary to study the possible hazards involved in the operation of the pile under the new conditions. This report is a summary of the studies, both theoretical and experimental, which have been made so far of the nuclear properties of the "new" pile, of the stability of the new fuel elements, and of the safety and practicability of operation of the "new" pile. Preliminary results of these studies were reported to a subcommittee (Osborne, Stratton, Brooks, with Russell) of the Advisory

Committee on Reactor Safeguards at a recent meeting.

The results obtained from these studies are sufficiently encouraging, as will be shown in the body of the report, so that a more extensive test of the new loading is now being planned. This test, if successful, could also serve as the first step in the conversion from the old fuel elements to the new ones. The old fuel elements from thirty-six central channels of the pile are to be replaced. About 300 new fuel elements will be used in this test and it will be possible to study the changes in the nuclear parameters of the pile, the metallurgical properties of the new elements, and the effect of the elements on the operation of the pile with enough elements to yield results with adequate statistics. Thirty-six channels should give these results without causing too great a perturbation in the operation of the pile, that is, without introducing any hazard problems. The conversion of 36 channels, however, represents a significant change from the present routine operation of the reactor and it is thought that the Advisory Committee on Reactor Safeguards should be consulted. This report will also serve, therefore, to present to the Committee the information it needs for its consideration of the 36-channel test.

II. Brief Description of the Brookhaven Pile and the Changes Proposed

A. Brief Description of the BNL Pile

The Brookhaven research pile has been described in detail, with emphasis on those aspects having to do with the control and safety of the pile, in the original report to the Advisory Committee on Reactor Safeguards^(2,3). In the following, we shall refer to that report by its number, BNL-18. For convenience, some of the details will be repeated here. The pile consists of a 25-foot cube of graphite penetrated by a square ar-

ray of 1369 channels (37 on a side) capable of containing fuel elements. The channels have a circular cross-section area of 36 cm^2 (2.67" inner diameter) and are spaced on 8-inch centers. The graphite cube is built in two equal parts separated by a central air gap, originally 8 cm wide. The gap permits the cooling air to enter each channel at its center so that hot air emerges from each end. This design permits cold air to strike the region generating the greatest heat flux, and to cool the graphite at the center of the structure.

The fuel element assemblies contain natural uranium slugs 1.1 inches in diameter and 4.0 inches long. Thirty-three of these are contained in a finned aluminum cartridge 11 feet long. One such cartridge lies in the air channel on either side of the gap, and each loaded fuel channel contains two 11-foot long elements so placed that they do not interfere with the passage of air through the gap. To improve the heat transfer efficiency, each aluminum cartridge has 6 longitudinal fins.

The pile has enough excess reactivity to permit sustained operation at a maximum flux of about 5×10^{12} neutrons per cm^2 per sec. The maximum fin temperature is in the neighborhood of 380°C and the exit air temperature is about 170°C . An air flow rate of 1.25×10^6 lbs/hr is required with a pressure drop through the graphite of 55 mm Hg. Under these conditions, the reactor has a heat output of 25 MW. The flux and power level are obtained under the conditions cited with about 675 channels loaded with uranium.

B. Proposed Change of the Fuel Loading

In order for the pile to continue to support its present large research program, the enriched fuel must be spread over approximately the same volume of the graphite as the old fuel. This can be achieved by loading each of

580 channels with 16 feet of fuel element, 8 feet on each side of the central gap. In steady-state, at 40 percent burn-up (see ahead), each loaded channel will contain about 96 grams of uranium highly enriched in U^{235} so that there will be about 57 kg of uranium in the pile. In the initial phase of operation, a freshly loaded channel will contain about 120 gm of uranium.

The fuel element will be made of plates of the MTR-type, the "meat" of which is a dilute alloy, about 8 percent by weight, of uranium and aluminum. The plates will be $24\frac{1}{8}$ inches long, about $3\frac{1}{8}$ inches wide and 0.060 inch thick; the "meat" of the plate is 0.020 inch thick and is covered on both sides by aluminum 0.020 inch thick. Three of these plates are formed into a single fuel element as shown in Fig. 2. Each plate contains 5.1 gm of uranium so that each element, 2 feet long, contains 15.3 gm of uranium. The total loading of uranium will require about 15000 flat plates from which 5000 fuel elements would be assembled.

The new fuel element has about 70 percent more heat transfer area per unit length than the old element, and it is expected that a flux of about 2.5×10^{13} neutrons per cm^2 per sec will be obtained at a power level of 20 MW. The design has been based on a fuel temperature not to exceed $350^\circ C$. The total air flow through the pile will be 1.1×10^6 lbs/hr. and the pressure drop through the graphite will be 55 in.Hg. The exit air temperature will be about $180^\circ C$.

The enriched pile contains considerably less U^{235} than the present natural-U pile, yet operates at about the same power level. Hence fuel burn-up with a consequent loss in reactivity must be considered. Periodically, fresh fuel elements will be loaded into the reactor at the plenum face of the core, and spent elements removed from the interior. The effects of this movement of fuel on flux distributions, handling procedures, etc. will be considered below.

C. Advantages of the Proposed Loading

The main advantage of the new loading has already been mentioned, namely the enhanced usefulness of the pile as a research tool because of the 4-5 fold increase in the thermal neutron flux. There are several additional advantages which result from the use of the new fuel element and attendant changes in some of the operating conditions. The use of plates containing a dilute uranium-aluminum alloy clad in aluminum makes the possible hazard due to oxidation of uranium very small; holes in fuel elements will no longer be a serious problem. The helium leak-detection system can therefore either be dispensed with or replaced by a much simpler system, thereby easing a persistent operating problem. Also, a difficult and hazardous situation in the canal, associated with the storing and shipping of the natural-U fuel will be greatly improved.

Some of the problems associated with the graphite will also be eased. For example, the oxidation of graphite in the presence of both radiation and high temperature will be reduced. The rate of oxidation is highly sensitive to temperature in the temperature range at which the pile is operated. The maximum graphite temperature will be reduced from its present value of about 300°C to about 250°C. In spite of the increase in neutron flux, the rate of oxidation should be reduced by a significant amount. In addition, there should be a decrease in the damage to the graphite caused by fast neutrons. If the pile is run at about 20 megawatts instead of 25-28 megawatts, the fast neutron flux will be reduced to about 70 or 80 percent of its present value. Since the fast neutrons are responsible for the graphite damage, rather than the thermal neutrons, the damage will be reduced, while the thermal neutron flux is increased. It is possible that this desirable effect

will be offset to some extent by the change in graphite temperature distribution in going over to the enriched reactor. The change in distribution can be altered favorably by decreasing the amount of fuel burn-up permitted.

III. The Nuclear Physics of the Enriched Reactor

The essential nature of the enriched core with regard to its reproductive properties is summarized by the value of $k_{\infty} = \eta \epsilon p f$. Since we deal with uranium highly enriched in U^{235} , we may take $\epsilon = 1.000..$ and $p = 1.000..$ and write $k_{\infty} = \eta f$. For uranium highly enriched in U^{235} , $\eta = 2.09$.

In calculating f and, consequently, k_{∞} , we shall distinguish between two cases. The first case involves a test-core composed of some thirty-six adjoining central channels in the natural-U reactor, each filled with 16 feet of enriched elements. A typical cell here will contain, as was noted in the Introduction, unburnt fuel, aluminum cladding and graphite. These give a thermal utilization of 0.610 and a $k_{\infty} = 1.27$ for a "clean" array at operating temperature. The k_{∞} for the remainder of the reactor core - which is composed of some 600 rods of natural uranium in a graphite matrix - is, in the same "clean", "hot" condition, $k_{\infty} = 1.07$.

The second case is that of the reactor fully converted to enriched fuel, and operating in its steady-state condition. Here, due to burn-up, we find k_{∞} a function of position, ranging from 1.27 to 1.09 for fuel that is 40 percent burnt. These values refer to a hot, clean reactor. The small thermal utilization suggests that in both cases there will be considerable absorption in graphite.

A comparison of neutron economy in the present and enriched BNL reactors follows:

	<u>Present</u>	<u>Enriched</u>
For each neutron produced via slow neutron fission of U^{235} the average number of neutrons slowing down past the fission threshold of U^{238} is ..	1.03	1.00
The average number that escape from the reactor during the slowing-down process is	0.039	0.036
The average number that are captured in uranium resonances during the slowing-down process is ...	0.12	0.00
The average number that escape from the reactor while diffusing at thermal energies is	0.031	0.120
The average number captured by moderating and cladding materials while diffusing is	0.096	0.37
The average number captured by fuel at thermal energies is	0.75	0.47
and this yields, on the average	1.00	1.00
fission neutrons.		

The average absorption cross-section of the reactor for thermal neutrons is considerably less than in the present reactor, and consequently the lifetime of a thermal neutron, $\ell = \left[\sum_a v(1 + L^2 B^2) \right]^{-1}$ is longer. In particular, $\ell = 0.0040$ sec in the enriched system and $\ell = 0.0016$ sec in the present reactor. For a given amount of excess reactivity, the period of the enriched system will be almost three times as long as that for the natural-U reactor.

To obtain an estimate of the excess k that will be required to bring the fully enriched reactor to criticality we will want to consider the poisoning effects of xenon, temperature and experimental facilities.

For a maximum flux of $2-3 \times 10^{13}$ there is no problem of xenon override after shutdown. The loss in reactivity due to equilibrium xenon will be about 0.014. The overall temperature coefficient (in vacuum) for the enriched reactor has been found to be about -1.4×10^{-4} per centigrade degree and is due almost entirely to the temperature variation of the large probability for neutron leakage. The effect of nitrogen in the air present in the pile brings the overall coefficient to about -9×10^{-5} per centigrade degree. The reactivity change involved is about 0.014. Experiments, which were originally rated at 0.0075 in the analysis of the present BNL reactor will be more significant in the new reactor because the neutron lifetime is greater. The principal experimental facilities presently at the reactor require an additional reactivity of 360 inhour or 0.0094 in k. In the completely enriched reactor they will require some 0.023 in k. In all, the enriched reactor will possess about 5.1 percent of excess k when it is in the clean, cold state. As we shall see, our control rod system is designed to handle more than twice that amount of reactivity.

We plan to operate the reactor so that at steady state the excess reactivity in the core is kept to a minimum, i.e., is less than one-half of one percent. This will be accomplished, for example, by adjusting the active loading during the periodic shutdowns so as to compensate for changing experimental loads.

Since the BNL reactor is a research facility, it is desirable to reduce constraints and regulations over experiments to a minimum consistent with standards of safety in reactor operation. In operating the enriched reactor, we shall carefully regulate the movement of experimental apparatus during shutdowns and take special care in the evaluation of the reactivity change

produced by the insertion of a new device into the reactor, since poisons are worth so much more in the new system. The increase in importance of poisons is the result of the increased neutron lifetime and migration area in the reactor. Neutrons have a greater opportunity to "see" a given absorber during their life-cycle.

The system which we propose during the transition from the "old" to the "new" reactor contains two types of reactive materials, a core of enriched uranium and a concentric region of natural uranium, separated by a graphite "buffer zone." It would be a complex task indeed to calculate in advance the behavior of such a reactor in every contingency. Instead, we have made a study of the Brookhaven reactor as completely converted to enriched fuel, which in conjunction with available information about the present, natural-U reactor should enable one to infer much about the character of the mixed system. It is as though we were to determine upper and lower bounds for a quantity whose actual determination was very difficult. In some case, as will appear presently, we have actually calculated with the "mixed model."

An important aspect of both the mixed and the fully-enriched reactors is that the effects of burn-up must be considered. At intervals, spent fuel elements will be removed from the interior of the reactor core and fresh elements introduced at the surface of the core. In effect, there will be a quasi-continuous movement of fuel through the reactor core, and a marked variation of k_{∞} from point to point. For a flow in which fuel effectively moves from the outer, plenum faces in toward the gap, where it is removed, the region near the central cooling gap will possess lower reactivity than the region near the core faces. Since, in a large, graphite-moderated reactor the thermal flux will peak appreciably in regions where reactivity is peaked*, we may

*Briefly, because the average distance travelled by a neutron from birth to thermal absorption is small compared with the size of the reactor.

expect the thermal flux distribution along the axis of the cylindrical core to be flatter than the usual cosine distribution. This observation is borne out by the results of a one-group diffusion theory calculation, and flux distributions for the enriched reactor, for various values of fuel burn-up are shown in Fig. 3. These refer, of course, to a condition of steady-state operation. When the core of the mixed reactor is loaded with unburnt fuel, its initial flux distribution will have the cosine shape. At the steady-state situation, the axial distribution in the enriched core will vary with radial coordinate, changing from a shape at $r = 0$ close to that for a fully enriched to a shape closer to "cosine" at the edge of the enriched core. The axial distribution over most of the natural-U region will be of cosine-type.

Accurate calculations for the mixed reactor are necessary to determine the critical dimensions of the system, and the flux and power distributions to be expected at initial-loading conditions. The two reactive regions must be sufficiently isolated from one another so that the heat production rate in the natural-U elements does not exceed the present value. An examination of the flux and power distributions in such a system under various conditions is in progress. The calculations are based on two-group diffusion theory.

An important parameter in the performance of the mixed and enriched reactors is the lifetime of a fuel rod in, say, the central channel of the enriched region. For the enriched reactor, one can show that the above lifetime is $\tau = 50.9 \frac{\beta}{W}$, where β is the fractional burn-up of the fuel at removal, W is the reactor power level, in megawatts, and τ , the lifetime, is given in years of operating time. Some numerical results are, for 20 MW operation:

β (burn-up)	τ (lifetime)
0.40	1.02 yr
0.30	0.76 yr
0.20	0.51 yr

When the enriched reactor is in steady-state operation, we shall observe a non-uniform distribution of fuel nuclei in the core. Consequently, the distribution of power density will be different from the distribution of neutron flux. In fact, since there will be greater thermal flux in regions of greater fuel density, the power curve will deviate from cosine-form even more than the flux curve will, as can be seen from Fig. 4. For the loading scheme in which fresh fuel moves toward the center of the reactor core, we observe that the region of maximum heat production is located at a considerable distance from the gap. Since it is at the gap that the cooling air enters, we are, in the enriched case, at a kind of "geometrical disadvantage" relative to the present scheme. This loss in cooling efficiency is a major factor working against the achievement of still higher flux values, or still lower operating temperatures.

IV. Controls.Kinetic Behavior

The control rod system consists of 16 rods in two sets of 8 rods each. Each rod is made of 1.75 percent boron steel and has a square cross-section of 4 sq. in. One set of rods enters the pile near the southeast corner and extends northwest for a distance of 25.5 ft. when fully inserted. The rods lie in horizontal planes 2 ft. and 6 ft. above and below the median horizontal plane. In each of the planes there are 2 rods extending diagonally, each rod situated 2 ft. from the long diagonal. The second set of 8 rods is symmetric with the first set, entering near the southwest corner and running northeast, again in horizontal planes through the reactor. Two rods are used as regulating rods, the other 14 as shim or safety rods. In addition to the rod system there are four wells into which boron shot can be dumped mechanically, as an emergency safety device.

Closely linked with the control rod system are the electromechanical

systems that indicate reactor period, power level, fuel, moderator and air temperature, etc. The systems now in use are either those described in BNL-18 or improved versions thereof.

The reactor is equipped with a set of period-measuring devices which may be set so as to shut down the reactor if a period shorter than a prescribed minimum is reached. The minimum period may be elected from the range 1-20 seconds and is presently set at 6 seconds. Three of these devices are arranged in parallel and a coincidence of signals from two out of three will initiate a shutdown. The system of period meters is a full-time sensing device; it is not looked out when the reactor achieves full power.

In addition, there is a set of circuits whose "sensing" portion is an ionization chamber designed to measure reactor power level. Again, a coincidence of signals from two out of three circuits will cause all of the control rods to be simultaneously inserted. Shutdown signals are generated when the pile power exceeds a maximum value generally chosen to be 10 percent in excess of the operating level. Finally, there are some fifty additional "trips" described in detail in BNL-18, which cause the reactor to be shut down if any of a number of operating parameters deviate sufficiently from required values. In all cases, the time interval between the sensing of the danger and the full insertion of the control rods is dictated by the acceleration and travel times of the rods.

Before the transition to a reactor fully fueled by enriched uranium is completed, improved safety devices responding to a combination of a power level and period will be installed. The new devices comprise three safety channels which sense reactor power and reactor period independently. Each channel has a "scram" trip. In each, the condition for a "scram" is:

$$\frac{n}{n'} + \frac{t'}{t} \geq 1$$

where n = existing power

n' = power for a "scram" trip at infinite period

t = existing period

t' = period for a scram trip at zero power

It will be seen that the period necessary to require a trip depends on the existing power, lengthening as the power rises. A "scram" trip in a given channel will result in a reactor "scram" only if the corresponding trip occurs simultaneously in at least one other channel.

The change in reactivity produced by the control rods, when they are fully inserted, was calculated for the present BNL reactor by means of the method introduced by J. A. Wheeler. In this manner, βk was found to be 0.0435. During the years of operation that have followed these design calculations, observations have demonstrated the accuracy of the theoretical estimate. Similar calculational techniques have been applied to the completely enriched reactor, and it is found that $\beta k = 0.148$ at full insertion. The three-fold increase in the effectiveness of the control rods might be predicted by qualitative considerations; M^2 , the migration area for neutrons, is almost three times as great in the fully enriched reactor as it is in the present one. Consequently, a given control rod or poison "controls" a proportionately greater volume of the reactor. In particular, the emergency controls - the boron shot system - has its control value increased from 0.007 to 0.018.

In the mixed reactor, we intend to replace the fuel in some three dozen central channels out of a total of 670 with highly enriched material. In the actual loading operation we shall first unload more than the required

number of channels, and then load the enriched material, until the reactor reaches criticality at very low power level. Considerably fewer than three dozen enriched channels will be required. Only after a number of studies have been made of the physics of this ensemble will more enriched material be added and the reactor gradually brought up to operating power. Criticality will be achieved through the replacement of rods of natural uranium and through the withdrawal of control rods. This will be done in such a manner that at no time will the total reactivity present in the reactor core be significantly greater than that in the present BNL reactor nor will the flux in the nearest channels containing natural-U slugs be permitted to rise above its present value. In the critical state, then, the enriched core will be surrounded by an annular region devoid of uranium.

A more quantitative idea of the effect of the enriched core on the time behavior of the reactor is given by the extreme case in which one blindly substitutes enriched fuel for natural fuel in 36 channels, with control rods adjusted so that the natural-U reactor is just critical. The substitution brings the reactor beyond prompt critical, and the rising period may be calculated from a simple one-group diffusion theoretic model of reactor kinetics in which the influence of delayed neutrons is neglected. The period is found to be 0.12 seconds; the same period would result if an overall increase in k_{eff} of about 2 percent were made in the present reactor. This is of the order of the excess reactivity built into the present reactor for the purpose of counteracting temperature, xenon, etc.

In the completely enriched reactor kinetic control problems require careful attention because of the small heat capacity of the aluminum fuel plates. The details of the control rod system are given in BNL-18; we shall repeat

here that at shut-down the control rods accelerate into the reactor and reach their maximum velocity of 5.5 feet per second in 0.5 seconds. In this period of time they have moved 1.4 feet. Their total range of movement is 25.5 feet. When completely withdrawn the rods must traverse a distance of 3.5 feet before encountering a region where there is significant thermal neutron flux.

Before initiating an analysis of start-up and shut-down kinetics, one should examine the following chart which, in comparing the present and the fully enriched reactors demonstrates how difficult it is to get extremely short periods in the latter case, by an reasonable kind of accident.

<u>k</u>	<u>Present Period</u>	<u>"Enriched Period"</u>
0.5 percent (~ 180 ih)	3.4 sec	5.6 sec
1.0 percent (~ 360 ih)	0.46 sec	1.4 sec
2.0 percent (~ 720 ih)	0.12 sec	0.45 sec
4.0 percent (Borax Case)		0.2 sec

The last case (4 percent) is to be compared with the Borax experiments in which a period of 0.002 sec was obtained.

The first kinetic situation to be analyzed involves a start-up in which the control rods are withdrawn one by one so that there is a uniform increase of reactivity with time. If the reactor is equipped with an automatic control device which will shut-down the reactor when a critical power level is reached, and only this device halts a runaway, there is a minimum period that we can tolerate at automatic shut-down without melting our fuel elements while bringing the reactor back under control.

In comparing the enriched reactor with the present one, we observe first that the control rods in the enriched system are "worth" more than in the natural-U pile, the ratios of the βk 's controlled (0.148 vs. 0.0435) being

approximately equal to the ratios of the neutron lifetimes in the two systems. Also, in the simplest picture of reactor kinetics - which is a good approximation in our case, where large reactivity changes are involved - the fractional change in neutron density in unit time is given by the reactivity change divided by the neutron lifetime. Thus, if we withdraw the rods from their fully-inserted positions at the same rate in both cases we get identical rates of increase of flux. The natural-U reactor will reach critical power sooner than the enriched system because it contains a greater density of U^{235} atoms. When the enriched reactor reaches the same power, at a later time and at a higher flux level, its flux is increasing with a significantly shorter period.

The simplest quantitative study, based on the above considerations gives the Newson relation:

$$\tau_s = \left[\frac{\tau}{2r \ln(n_s/n_o)} \right]^{1/2}$$

for the reactor period at automatic shutdown (τ_s) in terms of neutron lifetime (τ), average rate of change of reactivity (r) and ratio of shut-off to initial densities n_s/n_o . To achieve the same period at shutdown in the enriched reactor as in the present one, we must restrict ourselves to slower withdrawal rates.

In particular, a more precise calculation gives:

<u>Present Reactor</u>		<u>Enriched Reactor</u>	
<u>Withdrawal Period</u>		<u>Withdrawal Period</u>	
<u>(ft/sec)</u>	<u>(sec)</u>	<u>(ft/sec)</u>	<u>(sec)</u>
1.0	0.7	---	---
0.2	3.1	---	---
---	---	0.1	2.8
---	---	0.05	5.2

for the period at 30 megawatt automatic shut-down.

The effect of the automatic shut-down upon the fuel elements depends upon the disposition of the sixteen rods at that critical time as well as upon the

nature of the fuel plates themselves. In the original analysis of the control system for the present reactor, calculations were made for the case in which, as the reactor reached the power level that initiated automatic shut-down, the controls rods were all completely out of the core. It was found that for an initial withdrawal rate of 1 ft/sec an overload of 30 times the rated power level developed before the reactor was finally reduced to zero power. It was found that the overload could be significantly reduced - so that the temperature of the central U-slugs rose no more than 25°C - by maintaining the shim control rods at half-way out positions during the startup.

Similar calculations have been made for the fully enriched reactor. They indicate that if the reactor hits its critical power at a time when all rods are completely withdrawn, the aluminum fuel plates are in grave danger of melting during the ensuing shut-down. The critical period at automatic shut-down is in this case 0.6 seconds. However, if we permit the rods to begin their shutdown motion from a position advanced some 3.5 feet from complete withdrawal, placing them at the extrapolated zero point for the flux distribution, the amount of heat dissipated in the fuel plates during shut-down is reduced by a factor of two. It might be added that under the adiabatic conditions assumed throughout these calculations, about 28 seconds of power generated at 20 MW is required to melt the aluminum plates.

In sum we can state that if rod withdrawal rates are kept below say 0.3 ft/sec in the enriched reactor during startup, there is practically no chance that an accident involving the melting of fuel plates will occur. Plans have already been agreed upon to alter the mechanism for withdrawal of control rods so that the withdrawal speed will be fixed at 0.1 feet/sec. When the alterations are complete it will be impossible for the rod speed to be changed without making a major change in the driving mechanism. A withdrawal speed of 0.1

feet/sec. corresponds to a period of 2.5-3.0 sec at 30 megawatts automatic shutdown. In addition, it is our intention not to withdraw control rods fully, but only to the advanced position mentioned above.

The mixed reactor will be characterized by a relation between period at shut-down and withdrawal rate that is quite similar to that of the present reactor. Essentially, the difference lies in a slightly higher statistical weighting of the central region for control purposes. There are aluminum fuel plates of low heat capacity present, however, and this imposes a lower bound upon allowable periods. Although maximum power density in the enriched portion of the mixed reactor is less than what it would be in the enriched reactor, we may use the same criterion of 0.6 second periods, and restrict ourselves to withdrawal rates less than, say 0.5 feet/second in the mixed system. These rates may be obtained with the present control-drive mechanism.

In all of this discussion we have assumed that only a single safety device - the trip at critical power - is operative. Actually the trip at critical period will give additional insurance against runaways.

It has been suggested that the use of steel-clad, sandwich-type plates would, because of the greater heat capacity of the plates, reduce the possibility of melting fuel elements during an overshoot. Calculations have shown, however, that for equivalent amounts of parasitic absorption (poisoning) no improvement in heat capacity is obtained. The desired increase can only be obtained at the cost of adding fuel to the system, which carries with it the disadvantages of greater power to be dissipated, shorter neutron generation time, etc.

No serious or unusual problems arise in the study of the shut-down kinetics of the enriched reactor. One can find in BNL-18 a full and quantitative

discussion of the behavior of the present, natural-U reactor during shut-down. In cases where there is concern over the behavior of the fuel element under temperature cycling (see ahead) it is desirable to reduce the pile power at shut-down in a gradual manner. When the present reactor is shut down, the metal temperature is reduced at the rate of 2° /minute under full cooling until a temperature of about 100°C is reached, when power and cooling are reduced sharply. A similar scheme will be developed for the enriched reactor.

When, at shut-down, the natural-U reactor is in a subcritical state, there is a detectable thermal neutron background, due to spontaneous fission in U^{238} and its multiplication by the lattice. Since this minimum level is perceptible to the pile operator, he will have knowledge of the flux level during all phases of the start-up operation. It is estimated that with the removal of U^{238} , the shut-down counting rate will drop by a factor of ~ 1500 . In order to preserve the rate and be protected from the possible occurrence, unknown to the operator, of short periods at low power level, a suitable neutron source will be permanently placed in the reactor core. We estimate that enough Antimony 124 and Beryllium to produce about one curie of activity (γ, n) will be adequate. The $\text{Sb}^{124}\text{-Be}$ source is discussed in Reference 4.

V. Temperature Effects

The overall temperature coefficient of reactivity for the fully enriched reactor is negative, and equal to -9×10^{-5} per $^{\circ}\text{C}$ (-3.5 ih per $^{\circ}\text{C}$). It is, because of the greater possibility for neutron leakage, more negative than in the present reactor.

In changing from a natural to an enriched reactor, a considerable amount of U^{238} is removed from the system. Thus, our enriched system no longer possesses the temperature coefficient associated with the resonance absorption

of neutrons. This "prompt" coefficient, dependent only upon the metal temperature, was about -2.0×10^{-5} °/C. (~ 0.77 in per °C.) Its significance may be judged by the fact that, roughly, an excursion of about 1000°C in fuel element temperature would be required before the prompt coefficient could bring under control a reactor suddenly graced by 2 percent of excess k. Or, if the operating metal temperature were $\sim 300^\circ\text{C}$, and one asks what is the maximum δk that could be controlled by the prompt coefficient without our melting aluminum fuel elements in the process, one sees that this is only about 0.6-0.7 percent in k. This excess reactivity would produce a period of about 2.5 sec in the enriched reactor.

In judging whether a particular reactor will tend to "runaway" or prove unstable during normal operational procedures, one must consider the many factors that influence stability. In converting from a natural-U to an enriched reactor, we lose prompt temperature coefficient, but gain in having a longer neutron lifetime. We believe that the enriched reactor to be no more hazardous than the present reactor, and we would refer the reader to the safety record of the Reactor Operations Division at BNL.

A quantitative estimate has been made of the amount of U^{238} that would have to be replaced in our enriched plates - either by distributing it uniformly throughout, or by surrounding the element with a thin foil - in order to restore the prompt coefficient. We find that in order to obtain an appreciable fraction of our present coefficient, a large amount of U^{238} would have to be added. Specifically, the addition of 240 gm of U^{238} to a two-foot fuel element containing 15 gm of enriched uranium would restore only about one-quarter of the prompt coefficient. Certainly, the addition of so large a quantity of what is essentially a poison would prevent us from achieving

our goal, a significantly increased flux level, and would change completely the characteristics of the fuel element we use.

The barometric coefficient, due to the absorption of neutrons by nitrogen atoms results in a change in k whenever there is a change in the density of air in the reactor core. The density change might be produced by a change in temperature while the air pressure distribution in the reactor is maintained, or by a change in air pressure while the reactor is at operating power. The coefficient due to temperature change is approximately $+6.5 \times 10^{-5}$ per $^{\circ}\text{C}$ (2.5 ih per $^{\circ}\text{C}$) at an air temperature of 150°C . The coefficient varies inversely as temperature. The coefficient for pressure change is about -3.8×10^{-5} per mm.Hg. (1.5 ih per mm.Hg.) at 740 mm.Hg. The temperature portion of the coefficient is generally lumped in with the graphite coefficient and responds slowly to a change in, say, pile power. The pressure coefficient would play an important role if there were a sudden failure of the cooling fans. In this case, the increase in the density of nitrogen appears in a small fraction of a second, and a δk of about -0.001 (40 ih) corresponding to a negative period of 60 seconds would be the immediate result. In time, the air in the channels would be heated sufficiently so that the positive air temperature coefficient would cause the reactor to return to equilibrium, and then become super critical.

VI. The Fuel Elements

The fuel plates described in the Introduction have been used successfully in the MTR, at an operating temperature of about 60°C . Since we plan to use them at temperatures as high as 350°C a complete test program was devised and initiated in the winter of 1955 to investigate their behavior under conditions as similar to operating conditions as we could make them. To specify some

experiments, the elements were studied for their mechanical behavior in an air stream, their heat transfer properties, their behavior under long periods of irradiation, severe radiation damage, and temperature cycling. These experiments will be described below, in greater detail. The 36-channel loading we propose, now, is a natural continuation of the testing program.

A. Aerodynamic Tests

In their normal position in an unorificed channel, new fuel elements will be subject to a thrust of approximately 1-1/2 pounds, exerted by the cooling air. A system of aluminum anchors has been designed to hold the elements in place while permitting a maximum flow of air. Under these conditions no motion of the elements has been observed.

Since the BNL reactor has a central gap, through which the cooling air is introduced, the possibility that a two foot long fuel element will bridge the gap and interfere with the air-flow must be considered. The procedure for moving fuel through the reactor was designed to insure against this possibility (see below); in addition, the air-flow is from gap to plenum, and it is clear that the chance for bridging the gap is small. Should an element protrude into, or cross the gap, our tests show that the high air velocity perpendicular to the element will not cause it to flutter. The design of the element insures that the channel involved will not be starved for air.

B. Thermal Testing

Considerable time was spent in establishing the heat transfer characteristics of the new fuel elements. Experiments were performed with electrical heat sources in a mock-up of a reactor cell and the results were adjusted to fit data obtained from measurements made on reactive elements inserted into the present core. In both cases power and temperature distributions were obtained.

A number of sandwich fuel plates were exposed for long periods of time at temperatures of 350, 400, and 450°C in static air to study their behavior under these conditions. Several of these plates were sectioned to expose the core material to the hot air.

Examination of these plates after 13000 hours exposure at these temperature conditions indicated that:

1. No pimpling, buckling, or separation of core from clad had occurred.
2. No significant change in length had occurred
3. No measurable weight gain had occurred.
4. No migration of core material to surface of plate had occurred.
5. No exfoliation or powdering of the core material occurred on the plates in which the core was exposed. In fact, the core was outlined by a thin adherent oxide film.

Several complete elements in which the end rings were welded securely in place failed mechanically after temperature cycling 417 times between temperatures of 420 and 270°C. The failure was in the form of bent plates and ruptured end rings and was probably due to the thermal expansion stresses caused by uneven heating and cooling of the elements.

The end ring joints have now been modified to allow for the differential expansion. These elements have been cycled 720 times between 420 and 270°C (one minute cooling time) and 275 times between 400 and 40°C. No adverse effects of thermal cycling have been observed in these modified elements.

C. In-Pile Tests

In July of 1955, nine highly reactive two foot elements, three of which contained 51 gm. of enriched U, and six 39 gm. were inserted into channels in

the present reactor. Three 51 gm elements and three 39 gm elements were placed in high-flux channels (92 percent of maximum flux) and the last group of three 39 gm elements was placed in a channel with a flux of about 34 percent of maximum. The 51 gm. elements, containing over three times the uranium held by the ones we propose, were chosen so as to have a fission rate in the present pile flux close to that expected in the enriched reactor. Since, for a given fractional burn-up, we have produced 51/15 and 39/15 times as many fissions in a given volume element of these plates as we would in normal plates, we may expect that fission-induced damage to the microstructure will be scaled up accordingly.

In order to calculate burn-up with some accuracy, absolute thermal flux measurements were made along the test-channels. Temperature distributions along the plates were also obtained and it was found that some of the 51 gm. plates ran to temperatures as high as 420°C. To our knowledge, this is the first time that MIR-type elements have been used at such temperatures.

A serious problem associated with operating such fuel elements at high temperature until high burn-up is achieved, is that of the diffusion of fission products through the aluminum matrix material. If diffusion is rapid, a clustering of fission product gases around impurities, or micro-fissures may result. This, in turn, would lead to blistering. Also associated with this diffusion is the possibility that fission products may penetrate through the cladding at these temperatures and be carried off by the cooling gas. Serious radioactive contamination would be the result.

Three enriched elements (39 grams of U²³⁵) were discharged on December 10, 1955 and examined in the canal after they had been in channel B-O-4-S for five and one-half months. This exposure represents 5-1/2 percent burn-up of the

fuel or 14 percent of fission products in a regular enriched fuel element. No observable pitting, blistering, or surface defects were observed on the fuel plates. Some of them were slightly warped but this effect can be attributed to handling.

The support rings at each end of the plates usually had one or two cracked joints where the ring was brazed to a plate. However, every ring seemed to be firmly held in place by the remaining joints so that no danger existed that the element would come apart. This type of joint has been changed in a later design to alleviate the condition by dove-tailing a ring with slots into plate slots at the end of each element. The plate slot is then partially closed so the ring has freedom of motion but is prevented from slipping out of position. This freedom of movement prevents buildup of stress in the bent plates.

If, during the time that the test elements were in the reactor, a serious discharge of radioactivity either in the form of particulate matter or gas had occurred, it would have been noticed. The aluminum anchors located in the test channel, downstream of the elements, were smeared to test for radioactive dust; the results were negative. There was also no unusual change in the activity of the dust removed from the cooling air by the Glastex filter. (See BNL-18, p.82, for a description of this system.) Gaseous activity is monitored by sending a portion of the exit air through an ionization chamber. There is a constant signal from the chamber owing to the decay of radioactive Argon-41 that is produced during normal operation (see below). A change of about 2 percent or more in this quantity can be detected, but at no time in the year of testing the none super-reactive elements has there been any indication that gaseous fission products were escaping from them.

The "test" just described is crude, and sets a large upper limit in the amount of gaseous fission products that might be liberated into the atmosphere by fuel elements in the completely-enriched reactor. The following experiment, not yet completed, yields a much better estimate.

MTR-type fuel plates, containing natural- instead of enriched-U were irradiated by a neutron flux of 10^{12} in the BNL reactor, while they were maintained at a temperature of 350°C . Tank helium was blown over the samples and then passed through a trap into which the gaseous fission products would condense. We find that during the initial ten days of irradiation, the amount of gaseous fission product collected in the trap is the same whether the fuel element is present in the system or not. Studies of the decay of these products indicate that 9 hr Xe^{135} and 5 day Xe^{133} are present, along with other long-lived but as yet unidentified components.

We suspect that the background of fission product activity is due to the fission of uranium that is known to be present as an impurity in the aluminum out of which the system's container is made. The fact that no fission products originating in the element are seen is consistent with the estimate, based on an approximate value for the diffusion coefficient in aluminum, that many weeks should be required for the density of fission products to rise to an equilibrium value in the sandwich element. Theoretical analysis shows also that the fraction escaping depends sensitively upon the thickness of the cladding, and the product of diffusion coefficient and decay time. Indeed, a decrease in mean life by a factor of two cuts the escape probability by a factor of 1000. This may be one of the reasons - apart from a lower fission yield - why e.g. 4.3 hr Kr^{85} will not be seen.

A similar experiment, in which 0.020 in. Al-clad sandwich-type elements were first irradiated at ORNL and later heated in the laboratory was performed

at North Carolina State College⁽⁵⁾. Data indicated that 24 hours of heating at temperatures in the range 350-450°C would cause to be driven off only about 0.01 percent of the gaseous activity produced in the element. The activity obtained was essentially 5 day Xe¹³³. If the probability for diffusion out of the cladding for an atom of Xe¹³³ were as large as 1.0 percent, we would be discharging from our enriched reactor, at 20 MW operation, about 4 atoms of Xe¹³³ for each atom of active Argon-41. Since the argon is seven times as active as the xenon, the xenon activity emitted in this case would just begin to be significant relative to the Argon-41. The case of Argon-41 will be discussed below.

A somewhat different set of experiments to study the diffusion of fission product gases will begin shortly. These involve 20 percent burnt MTR-type fuel plates that have been obtained from ORNL. Since the plates have been "cooling" for more than a year, 10 yr. Krypton should be the only gaseous product left; its rate of emanation will be studied as a function of element temperature.

D. Radiation from Molten Elements

We have obtained some information regarding the activity of our fuel elements when melted. In one experiment, a sandwich-type plate containing 2.7 gm of depleted-U in its core was irradiated in the BNL reactor for about 24 hours and then placed in an out-of-pile experimental situation somewhat similar to the one described above. Here, however, compressed air was used to sweep the sample, and an exit air filter was installed and monitored to detect particulate activity. The activity in both the gas trap and the air filter was examined as a function of element temperature. The filter activity tended to increase steadily as the temperature was increased well above the melting point. It appeared that at all times, the magnitude of the particle-activity was very small compared with the gross activity of the fuel-element itself.

The gaseous activity proved to be too great for the counting equipment used; most of it came off in the first minutes after fusion.

These results are not inconsistent with personal communications we have received from Oak Ridge scientists who observed that in melting similar elements, only a very small percentage of the stored xenon (~ 1 percent) escapes. If this is true, then the sudden melting of all the fuel in the enriched reactor would result in the sudden release of about 22500 curies of radioactivity. (See Section VIII.)

E. Handling Procedure for Enriched Elements

Operation of the fully enriched reactor will necessitate the development of completely new methods for loading and unloading fuel. In the plenum loading scheme, in which fresh fuel moves from plenum to gap, the two fuel elements closest to the gap in each half of the reactor are discharged at a predetermined degree of burn-up. The vacancies created are filled by moving the remaining fuel towards the gap and replacing the spent elements with two new ones at the outer, or plenum end of the loading.

New handling techniques are required in this type of operation. Reactor fuel accountability and fuel rotation and replacement schedules are also complicated due to the large number (~ 5000 at full loading) of elements involved.

We plan to use a punched-card system to maintain fuel rotation schedules and accountability. By processing the cards associated with each channel of fuel prior to a scheduled shutdown, we will have an automatic indication as to which elements are to be removed. This system should not only minimize the amount of labor involved in our operation of this magnitude, but should result in more efficient accounting as well.

In the enriched reactor, the discharge and replacement of spent fuel will take place at a faster rate than it does now. This fact, along with the expect-

tation of higher radiation level from the new fuel call for handling techniques that are safe and rapid. A suitable method has been evolved. Prototype tools have been developed and the techniques have been tested in a fuel-channel mock-up. The new operation employs a remotely operated push-pull tool which can be made to engage or disengage the fuel elements for the purpose of getting them into a slotted bridge tube. By positioning the fuel in this tube, the elements which have been selected for discharge can be dumped into a remotely operated tray which can be lowered to the canal chute (see BNL-18). Further manipulation with the push-pull tool enables one to rearrange the fuel to allow for the charging of new fuel in its proper position.

The entire charging and discharging operation is to be performed from the charging elevator and there should be no necessity for removal of the scanner slot shield blocks. In addition, a grouted shield plug is to be kept in the end of the bridge tube which protrudes from the south face shield. This plug will prevent the pulling of a fuel element onto the charging elevator and should also provide the operators with shielding sufficient to attenuate the gamma activity of the active fuel to tolerable levels.

Rigid regulations and procedures relating to the storage of burnt fuel in the canal will be necessary, to avoid the possibility of attaining a critical assembly. Specific procedures will be established after relevant experimental studies have been made. These can be initiated only when a sufficient quantity of the new elements are at hand.

VII. Hazards in Normal Operation

A. Reactor Shielding

The major concern here is the shielding of personnel from the high-energy X-rays produced when thermal neutrons are captured in an (n, γ) reaction in

graphite and in iron present in the shield material. The former source of γ -rays is dependent upon both fast and slow leakage flux, while the latter depends upon fast leakage alone. It has been found that for a shielding configuration similar to the one we propose, γ -rays from (n- γ) reactions in the iron predominate greatly⁽⁶⁾. Since the enriched reactor, operating at lower power, will have a smaller fast leakage flux than the natural-U reactor, the present shielding will ensure the maintenance of radiation safety.

B. The Production of Argon-41

Argon-41, which is radioactive with a 1.8 hr. half-life is produced by neutron capture in Argon-40, present to about one percent in the atmosphere and hence, in our cooling air. At present, some 20.9×10^{17} radioactive atoms are emitted daily. To a very good approximation, the number of atoms manufactured is proportional to the reactor core volume and to the thermal flux level. Consequently, in the fully enriched reactor, where the flux level is increased by a factor of five, and the core volume is about three-fifths its present value we should expect to produce about three times as much argon. More precise calculations have been made and bear out this estimate. Our lack of knowledge about the diffusion of gases through pores in the graphite effectively limits the accuracy of our computations, which should tend to overestimate absolute yields.

The radiation level on the BNL site is proportional to the production of argon, and has been the subject of considerable study as a function of location and meteorological conditions. Under adverse conditions of occasional occurrence we now experience exposures of about 3 mrep in a 7-day period at locations at the edge of the laboratory site. Since current policy permits up to 30 mrep per week off-site, the new level will be acceptable. On-site peaks in

activity will increase from about 1-3 mrep/hr. Such peaks in themselves are probably unimportant from a radiation safety point of view. According to the best present estimate, on-site levels for particularly bad weeks may be increased to about 30 mrep per week. This figure is not quite as accurate as the one for off-site levels. Areas where the exposures may exceed 30 mrep/wk are normally designated as radiation areas and personnel monitoring equipment is required. It will be appropriate to improve our knowledge of the on-site levels by measurement, as soon as final decisions on the fully enriched reactor are made.

The effect of fission product gases upon the area radiation level during normal operation has been discussed. The effects are expected to be small relative to that of Argon-41.

The effects of Carbon-14, produced by neutron capture in graphite, and entering the cooling stream as CO or CO₂, has been discussed thoroughly in BNL-18 (Supplement, pp.74-78). The conclusion drawn there, that the biological effects of our production and dissemination of Carbon-14 on humans is quite negligible, applies here as well.

VIII. Some Possible Catastrophies

Data on atmospheric dissemination of radioactive material, hydrological and earthquake data and information regarding the make-up of the area surrounding the reactor site are contained in BNL-18. We treat here some situations unique to the enriched reactor.

An important aspect of mishaps in the operation of the enriched reactor is the possibility of melting the aluminum fuel elements. It is expected that the effusion of fission-product gases will be considerably greater than normal when the fuel is in its molten state. Experiments performed at Oak Ridge indi-

cate that upon fusion about 1 percent or less of the fission rare gases present in the element is liberated in an hour of heating in a normal atmosphere. The heating carries the element well above its melting point. Of the 1 percent release, almost all occurs as the element melts. Were there melting immediately following steady-state 20 MW operation, then, some 22,500 curies (3-1/2 times the daily argon activity) of radioactive rare gases would be sent up the exhaust stack. Agitation of the melt would change these figures considerably since one would expect to have much more gas released; complete release would yield 2.25×10^6 curies. Agitation from "normal" causes would not be expected to occur until one approached the boiling point of aluminum ($\sim 2300^\circ\text{C}$).

In the Oak Ridge experiments, iodine released from the melt was also identified; it was released to the extent of 0.01 -- 0.1 percent.

We must now consider under what circumstances melting of the plates may be achieved. As mentioned above, 28 seconds of heat generation at 20 MW and adiabatic conditions would be required to melt an element. This is a long time indeed.

A more credible situation is one in which fan power fails while the reactor is in steady-state operation at, say, 20 MW. Due to the inertia of the heavy flywheels, cooling air will continue to flow through the reactor for some seconds, and when the fan motion closes entirely, there remains a natural flow of some 50,000 lbs/hr. Clearly, there is no chance of melting the elements in such a situation in the time needed for the fan-power trip to scram the pile.

A less credible situation would be one in which the power failure was accompanied by failure of all of the fourteen independent flywheel-driven hydraulic systems for rod insertion. Under these circumstances, and with experimentally determined rates for fall-off of air flow, it was calculated (see Fig. 5) that almost three minutes of heating is required to melt the

plates. This is certainly ample time for the activation of emergency control systems, or the manual insertion of the rods. It should be noted that the emission of gaseous fission products varies only slightly with temperature at temperatures below the melting point. One has to melt the element before a significant amount of gas is released.

It has been suggested that enriched fuel plate temperatures might increase - perhaps excessively - in the event of a reactor shutdown which is accompanied by a sudden loss of primary cooling-air flow. These conditions are experienced at least several times a year, particularly on occasions when there is an interruption of incoming electrical power. Convictional air flow is maintained in such cases.

We have temperature data for the super-reactive 39 gm test elements during two such shut-downs. In a typical case fuel temperatures dropped smoothly through about 200°C to a minimum in about three minutes, and then rose by a few degrees. This type of shut-down is, then, by no means hazardous.

To return to the behavior of the fully enriched reactor when the fuel elements have been melted, it is important to consider the fate of the molten Al-U mixture. Tests have been made to determine under what conditions the melt will penetrate into pores or cracks in the graphite. These tests indicate that the U-Al alloy can penetrate into graphite if the molten fuel is in a very good reducing atmosphere and there is a vacuum on the other side of the slab into which penetration is desired. In this case, the pressure required is small (1/2 - 2 psi). However, any conditions that cause a monolayer of aluminum oxide to be formed will severely hinder penetration and pressures greater than 40 psi are required to force the molten fuel into graphite.

Consequently, should an enriched fuel element melt in the reactor it is not expected that any penetration into the graphite will occur. Since the

elements are air-cooled aluminum oxide will be formed, and large pressures will be needed for penetration. These pressures are simply not available, since the maximum pressure gradient in the BNL pile is about 1-2 psi.

The molten fuel will "puddle" in the graphite channels. Since the process of puddling changes the shape of our effective fuel element, making it thicker, there will be a change in thermal utilization. The quantity f , will decrease because of the increased self-shielding of the fuel. The corresponding decrease in k_{∞} may be calculated; it is found to be about 0.4 percent. Were it possible for the molten mixture to diffuse through the graphite, there would be an increase in k_{∞} , but in amount much less than 0.4 percent. Thus, in the situation where cooling and all controls failed, the melting of the elements would shut the reactor down, with a period of about 7 seconds.

Were molten fuel to run into the gap, it would fall to the floor of the reactor structure, thus passing from the core. In this case, there would be a definite loss of reactivity with an accelerated shut-down.

It is also possible, though highly improbable, that some event might interfere with the airflow in a channel, and result in the melting of its elements. The change in reactivity would be but a fraction of an inhour and might not be noticed, but the release of ~ 20 curies of gaseous fission products certainly would be. As we discussed above, the air monitoring system reports deviations of as little as 2 percent from the normal discharge rate of about 6000 curies per day, i.e., deviations of five curies.

One is generally curious about the effect of water upon the behavior of the reactor, and assumes that it is not too incredible for that substance to find its way into the core. Water would improve the moderation of neutrons, and thus increase reactivity by decreasing fast leakage. But it also increases the parasitic absorption of thermal neutrons, producing a decrease in thermal

utilization (which is partially offset by a decrease in thermal leakage). One suspects that the parasitic effect is the smaller, and a numerical calculation for small amounts of water bears out that this is so, by a factor of ten. This theoretical conclusion is verified by experience; water has accidentally found its way into the reactor on several occasions, and its effect on reactivity was strongly and unambiguously negative.

We close this section of the report with a consideration of the "maximum credible accident". We have seen that after a prolonged shut-down the enriched reactor possesses about 2.8 percent in excess reactivity, due to temperature and xenon effects. Suppose that at start-up, the rods are withdrawn steadily, at their immutable speed of 0.1 ft/sec. so as to bring the reactor to full power as rapidly as possible. This dangerous operation may be the result of a misunderstanding between scientist and operator. If so, there will be ample time for its clarification, since the enriched reactor will not become critical for 35 minutes. The reactor period at criticality is in excess of 20 seconds.

Before becoming prompt-critical, the reactor achieves a period of 6 seconds at which point the period meter trips should initiate a shut-down. If these trips are found to be inoperative, there is still time to "scram" the reactor by sundry means, since about 40 seconds remain before the reactor reaches the 20 MW power range. At this point the high-level trips would initiate the shut-down and the consequent power overload would not be sufficient to melt the elements. The reactor period at automatic power scram is about two seconds.

Let us suppose that the period trips were deliberately eliminated, and that the high-level (power) trips fail. About eight seconds of time must elapse after automatic power scram is reached, before the fuel elements melt. There is, as we have mentioned, a significant release of gaseous fission ac-

tivity at this point. If the reactor had been shut down for some days before start-up, most of the 9 hr xenon activity stored in the elements would have decayed. At fusion, then, only krypton and 5-day xenon would be discharged, and in amounts equal, approximately, to one percent of their equilibrium quantities in steady-state 20 MW operation. A simple calculation shows that the initial release of gaseous products would be of the order of 10^4 curies. If this activity were swept up and out through the reactor exhaust stack, we would be discharging into the atmosphere about two-thirds of the daily yield of argon-41. The possibility that the active gas will seep into the portion of the pile building occupied by personnel is not a likely one so long as the exhaust fans maintain an opposing gradient in air pressure.

The reactor, with its fuel molten, continues to rise in power. In the case we are considering, the control rods have not yet been completely withdrawn, and the rising period is about one second! There are several factors that will cause the reactor to shut down. When the molten aluminum has been heated to and beyond its boiling point (2300°C) vaporized aluminum will move out from the reactor core impelled either by its own vapor pressure, or by the airflow in the channels. The vapor will carry uranium atoms (or UO_2) with it, effectively redistributing the reactor fuel throughout a larger volume, thereby depressing the reactivity of the system. As a quantitative estimate, distribution of our uranium throughout the 25 foot cube produces a $\delta k \sim 10$ percent.

While the molten fuel is being heated, it radiates a considerable amount of energy to the surrounding graphite. At 2000° , the graphite receives of the order of 100 megawatts. Since oxidation of graphite is significant at temperatures $\geq 1000^{\circ}\text{C}$, the puddles of molten aluminum may "burn" into the graph-

ite, perhaps coalesce, with a decrease in overall reactivity.

When temperatures close to the boiling point of aluminum are reached, there should be a considerable increase in the emission of krypton and xenon; One might get as many as 10^6 curies released - i.e., all that were stored in the elements. In addition, we should consider the activity produced during the accident, which may be estimated as follows:

To bring our aluminum adiabatically to its boiling point, about 10^{10} calories or 0.75×10^{20} fissions are required. This represents the burn-up of approximately one-half gram of U^{235} . Since heat will have been lost to air and graphite we may take as an upper limit, 1.50×10^{20} fissions. This yields 5600 curies of 9 hr xenon and 290 curies of 5 day xenon. Fission activity produced during the incident is considerably less than the amount of stored activity that is released.

Surely this incident, in which control rods continue to be withdrawn, but do not respond to shut-down signals, and in which there is complete failure of instrumentation and personnel, resembles more a maximum incredible than a maximum credible accident.

IX. Peroration

The Reactor Department of Brookhaven National Laboratory requests that it be permitted to proceed, by installing a 36-channel loading of enriched fuel, in its plan for the ultimate conversion of the present BNL reactor into a completely enriched high-flux reactor. The change will greatly enhance the quantity and quality of neutron-physics research done at the Laboratory.

Studies show that the enriched reactor possesses the intrinsic safety of a long generation time, making it almost impossible to obtain short periods during operation. Ruptured cartridges and the accompanying release of par-

ticulate activity will be a thing of the past when the new elements are installed. The enriched reactor does not possess a negative prompt temperature coefficient, and when shut down does possess more (controlled) reactivity than we have at present. It is believed nevertheless, that with thoughtful operation there will be little danger of an accident.

In the more than five year history of the operations of the BNL reactor there has never been anything resembling an "incident" or unsafe operating condition. The control system has never failed to perform in accordance with the design specifications in preventing the establishment of unacceptable conditions. The control system imposes an automatic emergency shutdown, "scram", whenever any component of the procedures is violated. The violation limits are set far from really unsafe conditions, an example being the period measuring circuits which do not permit periods shorter than six seconds. A history of all the scrams which have occurred is presented in Figure 6. It is seen that there were a large number of shutdowns in the early days due to equipment failure, but that improvements greatly reduced their number later. The most significant aspect of this tabulation is that since January 1952 there have been only two emergency shutdowns which occurred because of violations of permissible limits established. It is notable also that there has never been a shutdown because of high radiation levels caused by the argon-41 constituent in the air coolant discharged from the stack. It might further be added that the operating record shows that the reactor is at high power more than ninety percent of the time throughout the five years. It is hardly necessary to say that the two supplementary, manually-activated shutdown systems, built into the original design, have never been used.

We are proud of our operating record to date, though humble when confronted with the problems of reactor safety. Yet we can see no reason to believe that our safety record would have been much different, had the new fuel elements been in use.

References

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(Note: Figure 1 was not included in this report)

BNL ENRICHED FUEL ELEMENT

END VIEW OF ELEMENT
FULL SCALE

SECTION THRU FIN BEFORE BENDING
DOUBLE SCALE

FIGURE 2

FLUX, PLENUM LOADING

FOR VARIOUS DEGREES
OF FUEL BURN-UP

FIGURE 3

43

44

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Maximum Temperature of Fuel Element Under Failure of
Coolant and Control Rods

(Constant Reactivity)

TABLE I
EMERGENCY SHUTDOWN HISTORY OF THE BROOKHAVEN REACTOR

[illegible]

**E.S.D. during permissible process conditions
 +E.S.D. during prohibited process conditions
 A Time at which high level trips were improved
 B Time when protective guards were installed over ESD buttons
 C Time when ESD circuits were made self-resetting on occasions of false signals.

FIGURE 6