Bordx I

28 fuel elements ^{ed} containing

138.6 gm ^{235}U in 18 Alclad Al-U alloy

3.88 Kg 25   4.16 Kg U(93)

NCT-04
#40

Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors

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INTRODUCTION

One of the important characteristics of a nuclear reactor is the degree of hazard which it creates in the surrounding area. If the usefulness of reactors, either for research or for power production, is to be exploited effectively, the hazard must be minimized, since isolation of the reactor compromises its utility and increases its cost. It is, therefore, important to find means for evaluating the hazards of specific reactors and methods of improving the safety of reactors in general.

The ultimate question in an evaluation of reactor safety is the question of what will happen if the reactor is inadvertently made supercritical and allowed to "run away" without any artificial limitation of its power. For, although safety devices which impose artificial limitations will certainly be provided for in the reactor design, the possibility of their failure as well as the consequences of their finite speed of operation must be recognized.

In general it can be said that the reactivity of a reactor will be related to its power level once the power has become sufficiently high to cause significant changes in the temperatures of the reactor parts. At moderate power levels this dependence can be such as to cause the reactivity either to decrease or to increase with power level, depending on the de-

sign of the specific reactor, but eventually at some power level any runaway reactor will become subcritical, through some degree of disassembly of itself if not by other means (Fig. 1). The safety question has to do with how violent the energy release becomes before the eventual shutdown is achieved.

The more important ways by which increasing power can cause a reactor to lose reactivity are by expansion of the fuel, by heating and expansion of the moderator, and, if strong resonance absorbers are present, by Doppler broadening of the resonances. In many cases the unknowns in the magnitudes of the applicable effects and in their speeds of operation make safety evaluations quite uncertain.

The most straightforward way of evaluating the unknown aspects of the reactor shutdown process is by observing experimental runaways of actual reactors. Some experiments of this kind on two reactor types which have particularly favorable power-limitation characteristics, the solid-fuel, water-moderated reactor and the water-moderated homogeneous reactor, are reported here. In addition to the instrumental measurements reported here, motion picture records, which add materially to the information on the safety characteristics of the reactors, are available.

Homogeneous reactors can, in general, be made to have negative temperature coefficients of reactivity. The negative coefficient results primarily from thermal expansion of the fuel solution, which decreases not only the density of the moderator, but that of the fuel as well. The coefficient is quite large for small reactors with high neutron leakage. Furthermore, since the heat of fission is liberated directly in the fuel solution, the action of the negative coefficient is very rapid, and insofar as such reactors can limit their power by temperature coefficient alone, they can be expected to be quite effectively protected against destructive runaways.

The solid-fuel water-moderated reactor may get a certain degree of power limitation from the Doppler coefficient if it contains a large fraction of U^{238}. Beyond this, power limitation comes from the moderator temperature coefficient and expulsion of moderator from the reactor core by the formation of steam at the hot fuel element surfaces. The bulk temperature of the moderator does not change rapidly enough
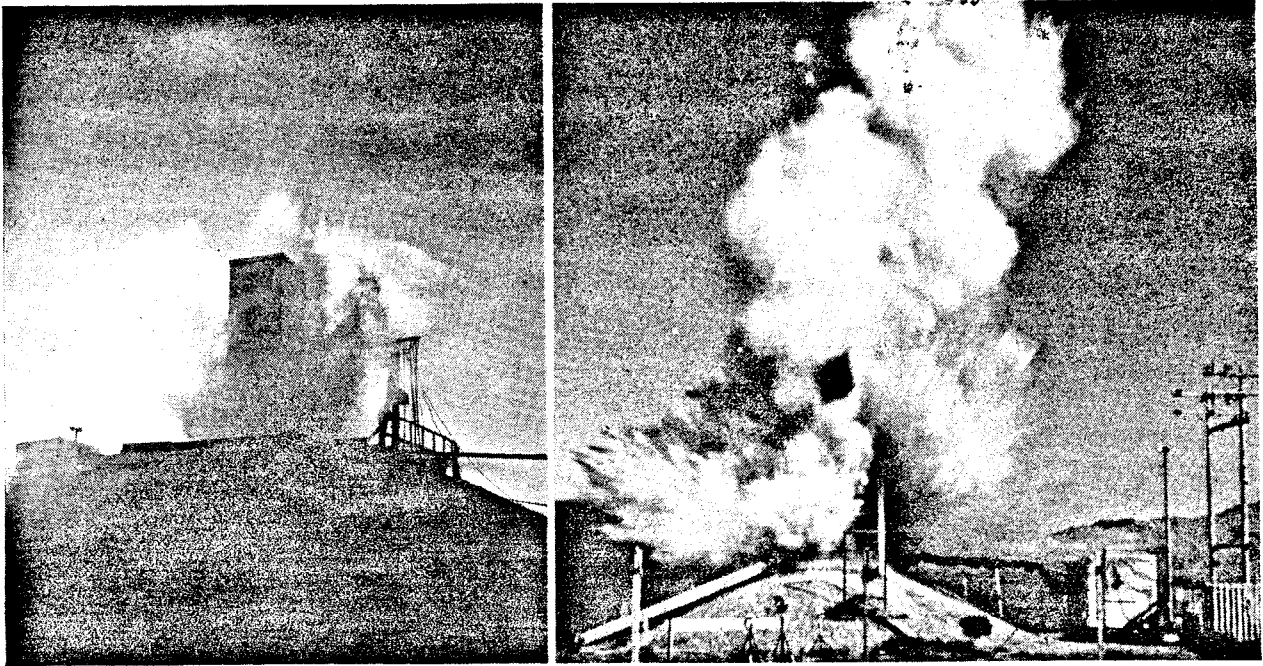
Figure 1. Effects of large reactivity additions to a water-moderated reactor; left, reactor shutting itself down safely by water expulsion after being made supercritical by 2.1% k_{eff}; right, reactor destroyed by melting of fuel plates after being made supercritical by 3.3% k_{eff}

to make the temperature coefficient effective against rapid power increases, but early laboratory experiments by Untermyer and later ones by West, Weills, Hooker, and Schiltz showed that the expulsion of moderator by steam can be very rapid.[1,2] The reactor experiments, initiated at the suggestion of Untermyer, confirmed the effectiveness of the process. They were begun in the early summer of 1953 by Argonne National Laboratory as part of an experimental program on boiling water-moderated reactors. The effectiveness of boiling as a safety process was proved, and the following year the severity of the experiments was increased to the point of planned destruction of the reactor. A new reactor was built in 1954 with which similar experiments were made at elevated pressures.

In July and August of 1953 safety experiments were run on the Los Alamos "Supo" reactor by Lyon, Kasten, and others of Oak Ridge National Laboratory and King, Zabel, and others of Los Alamos Scientific Laboratory. Shortly thereafter a program of more drastic safety experiments was run on the Homogeneous Reactor Experiment at Oak Ridge National Laboratory by Paré, Visner and others. Although these experiments were not carried to conditions as severe as those used for the experiments on solid-fuel reactors, they demonstrated a high degree of inherent self-limitation of power in the homogeneous systems.

EXPERIMENTS WITH HOMOGENEOUS REACTORS

The runaway behavior of the non-boiling homogeneous reactor, whose power is limited by the temperature coefficient of reactivity, is the most straightforward of those investigated. Reactors of this type may have very high negative temperature coefficients of reactivity which result primarily from the expansion of fuel solution out of the reactor proper as the temperature increases.

Figure 2 is a diagram of the Homogeneous Reactor Experiment, which was used for the tests described here. The core is normally full of the fuel solution of enriched uranium in water. If the solution heats and expands, the displaced fraction of the solution is accommodated by the pressurizer tank, where it contributes nothing to reactivity. The temperature coefficient of the reactor was about -0.1% k_{eff} per degree C. The volume of the core was 50 liters, and the effective prompt neutron lifetime was 7.5×10^{-5}
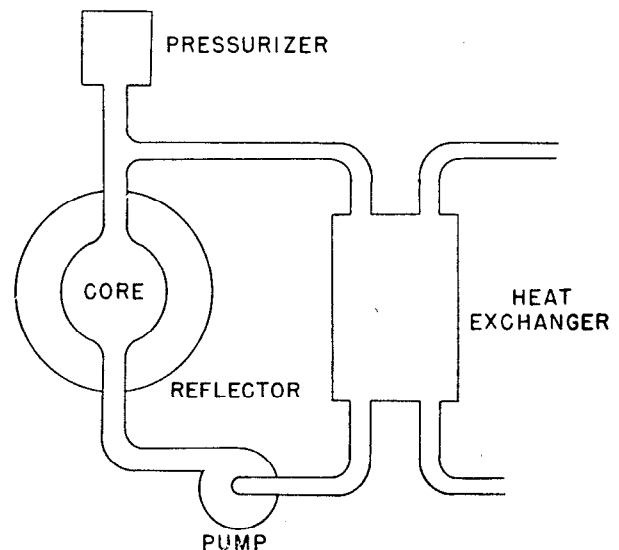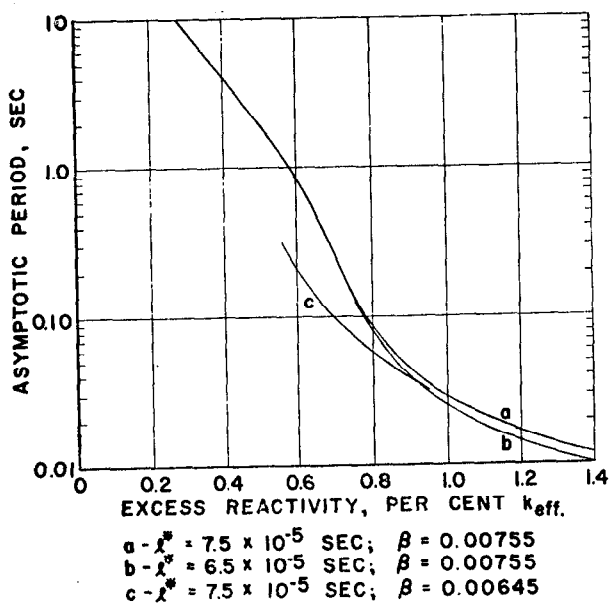

Figure 2. Diagram of homogeneous reactor system

Figure 3. Relation between excess reactivity and reactor period

a - $\ell^* = 7.5 \times 10^{-5}$ SEC; $\beta = 0.00755$
b - $\ell^* = 6.5 \times 10^{-5}$ SEC; $\beta = 0.00755$
c - $\ell^* = 7.5 \times 10^{-5}$ SEC; $\beta = 0.00645$

sec. The relation between excess reactivity and asymptotic reactor period is given by curve c, Fig. 3. The deficiency in delayed neutron fraction results from the circulation of the fuel solution. The reactor was pressurized to a pressure of 1000 psi.

If reactivity is added to such a reactor according to some law of time variation, $k_a(t)$, resulting in a period which is short compared to the residence time of the fuel solution in the reactor, then the law which expresses the variation of reactivity with time is simply

$$k(t) = k_0 + k_a(t) + C \int_0^t P(\tau)d\tau \qquad (1)$$

where C is a characteristic constant of the reactor which includes the heat capacity and the (negative) temperature coefficient of reactivity, and $P(\tau)$ is the instantaneous reactor power. This equation must, of course, be coupled with the usual differential equations characteristic of the kinetics of the neutron chain reaction to specify the variation of reactor power with time. If the power increase is fast enough that the compressibility of the fuel solution is important, still further relations must be included to describe the dynamics of the system. Kasten has treated these considerations at some length.[3] Regardless of the complications which may occur in specific reactors, the safety experiments which have been made indicate that the fundamental situation is reasonably well understood.

The reactor used for the experiments was not provided with special means for increasing reactivity rapidly, and hence the experimental situations were those relatively complex ones which would be characteristic of practical reactor accidents. Reactivity was increased experimentally by several methods: withdrawal of a weak control rod; increase of fuel concentration; raising of the reflector level;

rapid cooling of the circulating fuel; and pumping of precooled fuel solution into the reactor proper. The latter method gave the largest and fastest reactivity changes, and the results obtained by it are the ones reproduced here.

The experiments were made by stopping the circulating pump (Fig. 2), cooling the fuel solution in the heat exchanger to about 100°C, and then restarting the pump to inject the cooled solution rapidly into the reactor core, which had been maintained at a temperature of about 180°C. The severity of the experiment was adjusted by adjusting the initial power level of the reactor before the cold solution was injected. Figure 4 shows the measured power variations for two different initial power levels. Naturally, the lower initial level allows the greater increase in reactivity before the reactor begins to shut itself down and results in the higher maximum power. The temporary power decrease immediately after the pump starts is a result of the initial loss of delayed neutron emitters as the old fuel solution is displaced by the new.

This type of power transient can be characterized by specifying both the rate of addition of reactivity by the inflow of cold solution and the minimum reactor period reached during the transient. Figure 5 contains a set of theoretical curves giving the maximum power reached as a function of the minimum period reached during the transient, with rate of reactivity addition as the parameter. The experimental results are plotted in the same figure, with the rate of reactivity addition indicated by each. In view of the relatively complex experimental conditions, the agreement with theory is satisfactory.
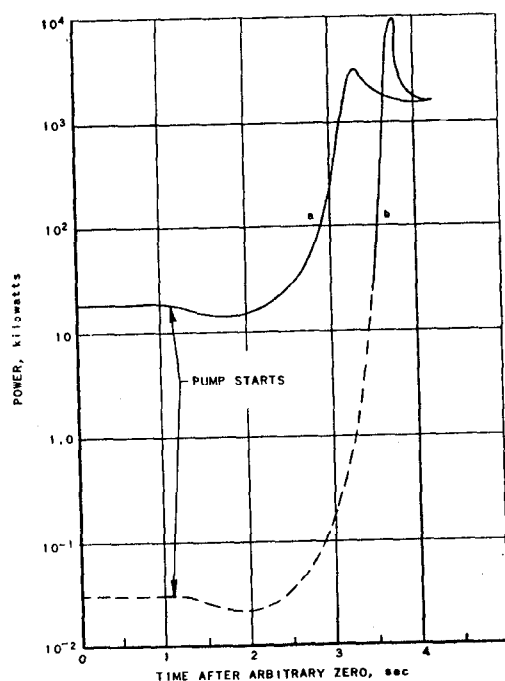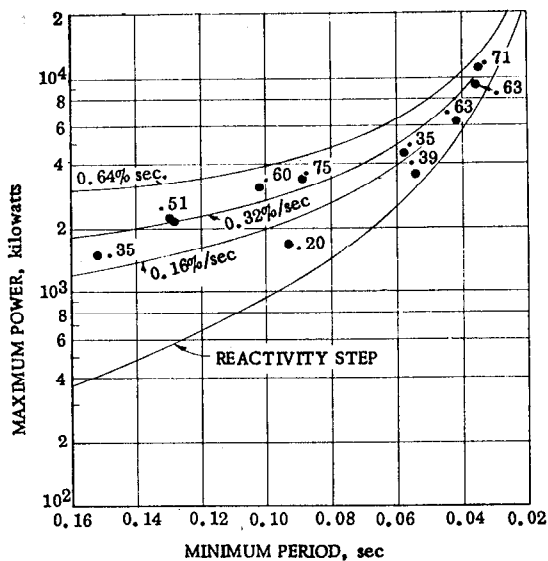

Figure 4. Power variations during injection of cold fuel solution in homogeneous reactor experiment

THE CURVES ARE COMPUTED FOR VARIOUS RATES OF REACTIVITY ADDITION. THE RATES OF ADDITION USED ARE MARKED BY THE EXPERIMENTAL POINTS.

Figure 5. Maximum power as a function of minimum period for homogeneous reactor power transients

The experiments indicated, further, that for the conditions tested the reactor would settle down to steady operation after the transient. The power level would then be determined simply by the rate of heat removal from the heat exchanger: the reactor power would be self-regulating via the temperature coefficient of reactivity.

The Supo reactor differs from the Homogeneous Reactor Experiment in that its fuel solution is not circulated; heat is removed by a cooling coil in the reactor vessel. Furthermore, the reactor vessel is not completely full of solution. The temperature coefficient is nevertheless strongly negative: about — 0.024% k_{eff} per degree C. The reactor has been described by King.[4]

The safety experiments were made in connection with an investigation of boiling operation of the reactor. The reactor was operated at powers of several kilowatts as a boiler, and the power under this condition fluctuated, but was self-regulating. Sudden reactivity additions up to about 0.4% k_{eff} were made, under conditions of both boiling and non-boiling operation. In both cases the reactor power was self-limiting, but the excursion was terminated more rapidly under boiling conditions. The experiments indicated that there is no very long time delay in the formation of steam bubbles in a homogeneous reactor once the solution has reached saturation temperature. Once this is known, it is to be expected that steam would be by far the more effective shutdown agent for long-period power excursions at atmospheric pressure. For whereas about 5000 calories of heat are required to produce 1 cm³ of effective void in liquid water by thermal expansion, only about 0.3 calorie is required to evaporate sufficient water to produce 1 cm³ of steam at atmos-

pheric pressure. It is by no means evident, however, that the same situation would hold for very short period transients or at very high pressure.

EXPERIMENTS ON SOLID FUEL REACTORS

In the experiments made with solid-fuel, water-moderated reactors the expulsion of water by steam formation was the important process in transient limitation of the power. Since a quantitative theory of the process has not been developed, it is necessary to present the results and the experimental conditions in some detail.

The experiments were made in two different reactors which were also used for investigation of the steady-state characteristics of boiling reactors. The pertinent differences between the two reactors lay in their core characteristics. These differences will be described, but differences in the mechanical details of the two reactors will be ignored.

Description of the Reactors

Figure 6 is a cutaway drawing of the first reactor, which was constructed outside and which was operated remotely from a control station half a mile away. The reactor tank was contained in a larger shield tank of ten-foot diameter which was sunk part-way into the ground and had earth piled around it for additional shielding. Adjacent to the shield tank was a pit with concrete walls in which was installed equipment for filling and emptying the reactor and shield tanks, and for preheating the water in the reactor tank. The reactor tank, four feet in diameter and about thirteen feet high, contained the reactor core, which consisted of an adjustable number of plate-type fuel elements held at the bottom by a supporting grid and at the top by a removable cover grid.

In operation the reactor tank was filled with water to a height of three to four and one-half feet above the top of the core; this water constituted the reflector, moderator, and coolant. The shield tank was filled with water only when the reactor was shut down.

The reactor contained five cadmium control rods which were operated by drive mechanisms located in the rectangular housing above the shield tank. The connection from the mechanism to the rods was through spring-loaded magnetic couplings. These couplings could be released in unison or individually, allowing the rods to drop freely downward under the acceleration of the springs plus gravity. When released, the center control rod dropped out of the reactor core to apply the excess reactivity used for the experiments. The other four rods when released dropped into the reactor core to terminate the experiments. Each rod traversed the length of the core in about 0.2 sec.

The fuel elements were made of aluminum-clad, aluminum-uranium alloy plates, of 60 mils total thickness, fastened into aluminum side plates to make boxes roughly 3 inches square. Figure 7 is a
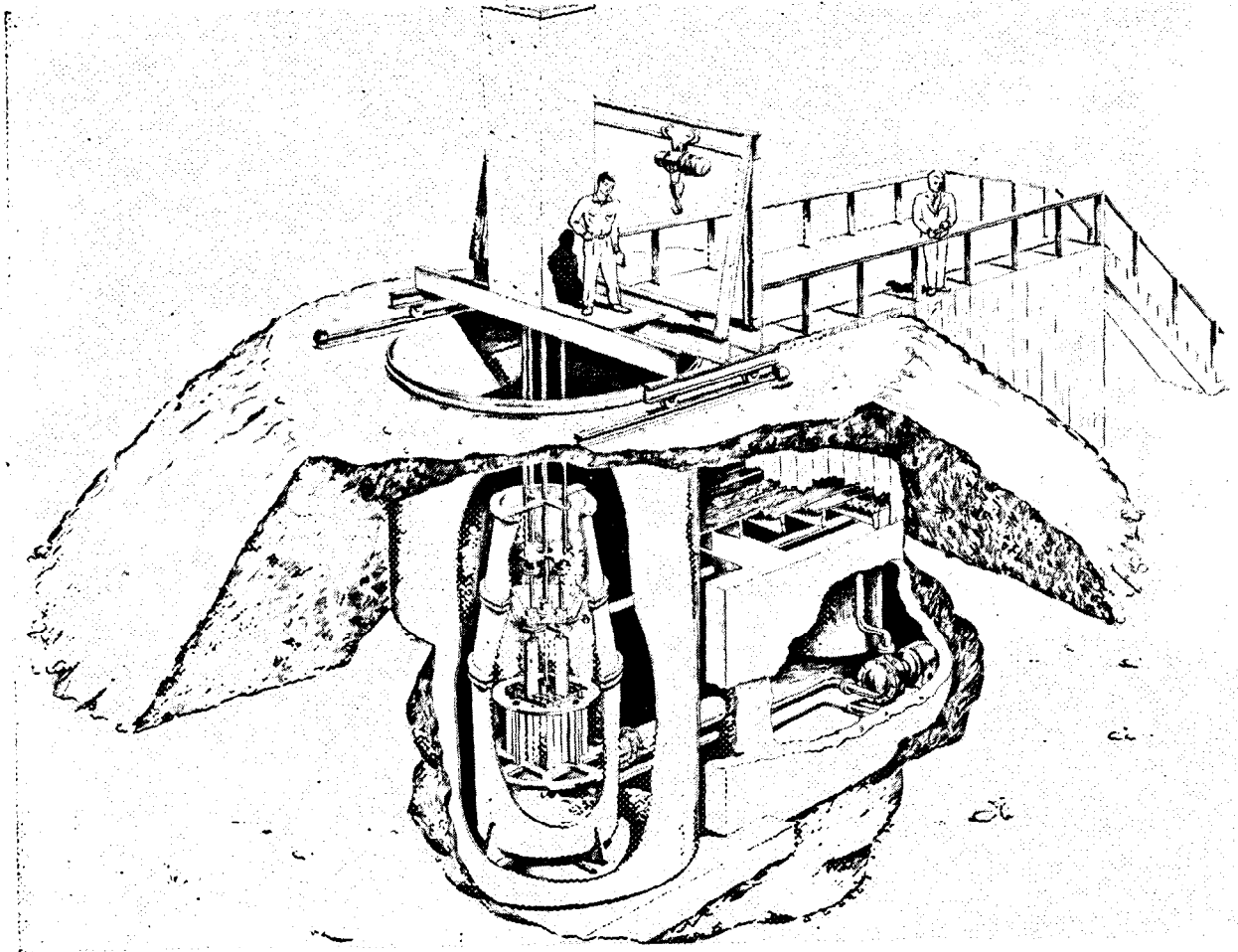
Figure 6. Cutaway drawing of reactor I

drawing of a fuel element for one of the reactors. The other reactor used elements of identical outside dimensions, but each element contained only ten of the fuel plates.

The two reactors will be designated I and II. Each reactor was loaded, for any given experiment, with the number of fuel elements which would give a convenient amount of reactivity. Typical loadings for the two reactors are diagrammed in Figure 8. Reactor II contained several elements of higher uranium content around its periphery to flatten the power distribution.

The relationship between asymptotic reactor period and excess reactivity for the reactors is given in Fig. 3 (curves a and b). For larger excess reactivities the period (τ) is given by

$$\tau = \frac{l^*}{k_{ex}(1 - \beta) - \beta}$$ (2)

where β is the total delayed neutron fraction and l^* the effective neutron lifetime. Other characteristics of the two reactors are summarized in Table I.

Typical Transient Behavior of the Reactors

The experiments were made by the following procedure. The reactor water temperature was adjusted

to the desired value, and the reactor was made critical at a low power (about 1 watt) by appropriate positioning of the control rods. The center control rod was then dropped out of the reactor core. The initial power was sufficiently low and the speed of rod ejection was sufficiently high so that in almost all cases the rod was completely out of the core and the reactor period reached its stable value before
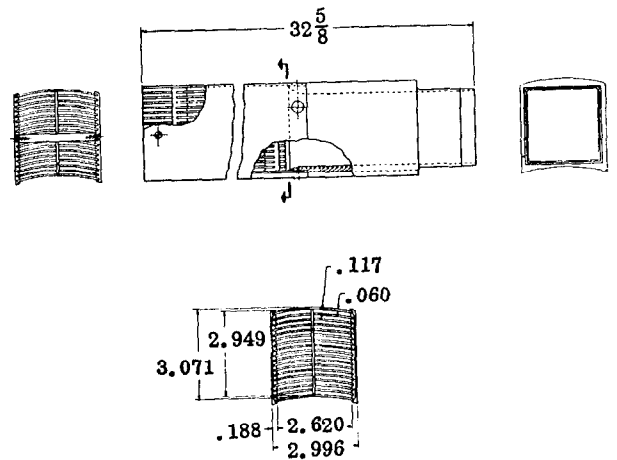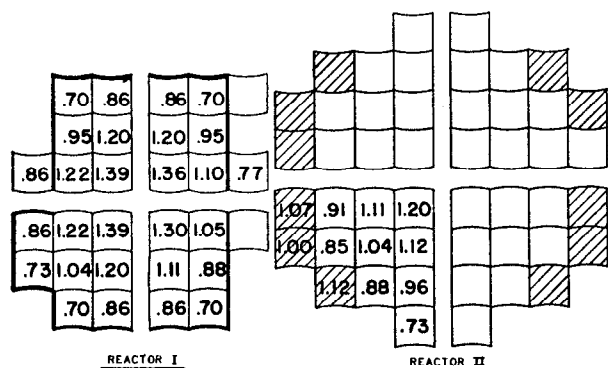

Figure 7. Standard fuel assembly

the reactor power had risen high enough to produce significant thermal effects. The power was allowed to continue to rise until the formation of steam in the reactor core reduced the reactivity below criticality and caused the power to fall to a low value. After it was evident that the power had been safely limited by the formation of steam, the remaining four control rods were dropped into the reactor to terminate the experiment, referred to as a power excursion. By proper adjustment of the number of fuel elements in the reactor core and of the positions of the four outer control rods, the reactor could be made critical with the center control rod inserted to any desired degree in the core. The magnitude of excess reactivity applied by ejection of the center rod could thus be adjusted at will.

Figure 9 is a reproduction of a typical chart from the multichannel magnetic oscillograph which recorded the data on the experiments. In this case the applied excess reactivity was 1.4% k_{eff} and reactor I was used. The neutron flux (proportional to reactor power) was recorded over about three decades by three different neutron-sensitive ion chambers working through logarithmic amplifiers. The stable reactor period (τ) is indicated by the three ion chamber records as 0.0096, 0.0107, and 0.0109 second, respectively. The temperature of one of the fuel plates, which was situated at roughly the highest flux position in the core, is recorded by two fast-response thermocouples. Both of the couples were located near the position of maximum neutron flux; one was installed on the surface of the plate, the other at its central plane. There is little difference between the two temperatures, because of the high thermal conductivity of the thin plate.

The ion chambers, which were calibrated in terms of absolute power by thermal methods, indicate that the reactor power reached a maximum value of 220 megawatts before the formation of steam checked

Table I. Comparison of Reactors I and II

	Reactor I	Reactor II
Ratio, $\dfrac{\text{volume aluminum in core}}{\text{volume water in core}}$	0.626	0.422
U^{235} content per fuel element	138.6 gm	93.4 gm or 157.3 gm
Number of fuel plates per element	18	10
Plate spacing (between center lines)	0.177 inch	0.324 inch
Measured reactivity loss with temperature increase:		
80°F to 200°F	0.82% k_{eff}	0.45% k_{eff}*
80°F to 280°F	1.93% k_{eff}	0.76% k_{eff}*
80°F to 420°F	- -	1.57% k_{eff}*
Calculated loss of reactivity caused by replacement of 10% of core water by steam, at 200°F	2.4% k_{eff}	1.0% k_{eff}*
Effective neutron lifetime (l^*)	6.5×10^{-5} sec	7.5×10^{-5} sec*

* These values apply to the case in which the reactor is loaded only with fuel elements of low uranium content.

the rise. Further generation of steam reduced the reactivity below the critical value, and caused the power to decrease very rapidly to a value of about 0.2 megawatt.

Once the initial power excursion has been checked by boiling in the reactor the specific power variation depends both quantitatively and qualitatively upon the amount of excess reactivity to which the reactor was initially subjected, and upon the bulk temperature of the reactor water. Figure 10 contains tracings from excursion records for reactor I similar to those of Fig. 9, but of longer duration. They summarize the typical behavior of the reactor for various amounts of applied excess reactivity when the reactor water is initially at saturation temperature. When the excess reactivity applied is low, corresponding to a reactor period of about 0.03 sec or longer, the reactor power after the initial surge settles down to a relatively steady value in the neighborhood of half a megawatt (top curve, Fig. 10). For this type of excursion the self-regulating characteristic of the reactor operates rapidly enough to stabilize the power at a steady value characteristic of the amount of applied excess reactivity. After the power has reached this steady value, further reactivity could, of course, be applied, and the reactor would continue to operate stably in steady boiling at a higher power.

If the excess reactivity which is applied by ejection of the control rod exceeds that corresponding to a period of 0.02 or 0.03 sec, the initial power excursion is followed by a series of qualitatively similar excursions of smaller amplitude, which occur at intervals of about 1 sec (second curve, Fig. 10). The amplitudes of the successive excursions, although they vary in an irregular manner, have no sustained tendency to increase or decrease. This

REACTOR I   REACTOR II

EACH BOX REPRESENTS A FUEL ELEMENT.

THE SHADED BOXES IN REACTOR II CONTAINED FUEL PLATES OF HIGHER FUEL LOADING (SEE TABLE I)

THE NUMBERS IN EACH BOX INDICATE THE RATIO OF THE TOTAL POWER PRODUCED IN THAT ELEMENT TO THE AVERAGE POWER PER ELEMENT; THE MEASUREMENTS ON REACTOR I WERE MADE WITH A SLIGHTLY DIFFERENT LOADING THAN THAT DIAGRAMMED.

THE HEAVY OUTLINE IN REACTOR I ENCLOSES THE CLEAN COLD CRITICAL LOADING. REACTOR II WAS CRITICAL AT ROOM TEMPERATURE WITH A 6 BY 8 ELEMENT ARRAY OF THE MORE LIGHTLY LOADED ELEMENTS (TABLE I)

THE AXIAL MAXIMUM/AVERAGE POWER RATIO IS 1.3 FOR BOTH CORES

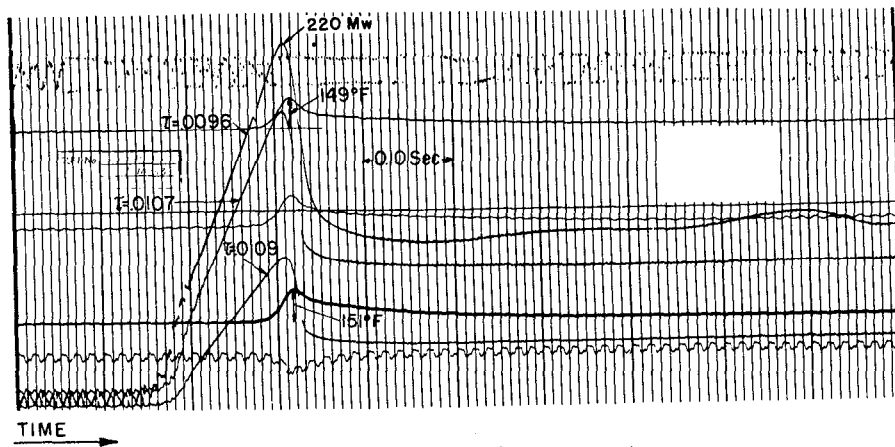Figure 8. Core loading diagrams for reactors I and II

TIME

Figure 9. Typical record of power excursion

type of operation will hereafter be referred to as "chugging."

When the applied excess reactivity was greater than that corresponding to about a 0.01-sec period, the chugging was no longer observed, and the power after the first surge remained at a low value. This permanent shutdown was no doubt the result of expulsion of sufficient water from the reactor tank to partially uncover the reactor core. The occurrence of this behavior in other reactors would, of course, depend upon the specific design of the reactor in question.

When the applied excess reactivity was increased to about 2% k_{eff} to give periods in the 0.005-sec range, the qualitative behavior of the reactor power remained the same, but the fuel plate temperature did not drop immediately after the power surge (bottom curve, Fig. 10). The fuel plate temperature remained high for almost a second after the power surge and then decreased by small jumps, as though the plate had been blanketed by steam for some time after the power excursion.

Experiments of this type were not carried to periods shorter than about 0.013 sec when the reactor water was cooler than the saturation temperature. With this condition, which will hereafter be called the subcooled condition, chugging was never experienced. In all cases after the initial power excursion the reactor power stabilized at some more or less steady value. Figure 11 is a record of such an experiment on reactor I, in which the initial period was 0.014 sec. Although the reactor power oscillated after the excursion, the oscillation amplitude was very much less than that for typical chugging operation.

Some of the details of the nuclear and thermal behavior of the reactor during a power excursion are illustrated by Figs. 12a and 12b, which apply to the subcooled reactor I for excursions of two different periods. The curves show the time variation of reactor power, on a linear scale, and the fuel plate surface temperature. In these experiments the fuel plate, with thermocouple attached, was coated with a thermally insulating plastic over a section of

its length. The temperature of this section of the plate is also included in the figures. This temperature, except for the effect of a small heat loss through the thermal insulation, is proportional to the total nuclear energy liberated in the plate. The point at which there is a sharp deviation between the temperature of the bare section and that of the insulated section evidently marks the beginning of rapid steam formation at the bare plate surface. Up until this time the temperature of the bare plate, like that of the insulated plate, is roughly proportional to the time integral of the reactor power.

Since no single fuel plate can produce sufficient steam to shut the reactor down it is evident that the time relationship between peak reactor power and peak temperature of the bare fuel plate, as well as the ratio of maximum bare plate temperature to maximum insulated plate temperature, will depend upon the local value of power density at the plate location relative to the power density elsewhere in the reactor. The plate used for these measurements was located at or near the maximum power density, and the thermocouples were installed near the point of maximum power density in the plate. Nevertheless, Fig. 12a indicates that for long-period excur-
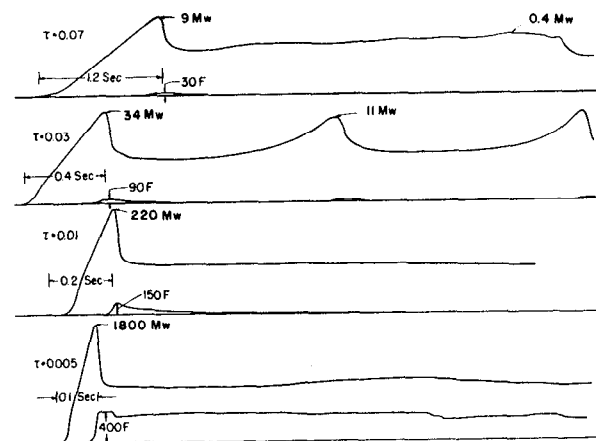

Figure 10. Representative records of excursions at saturation temperature with various excess reactivities

TIME AFTER ARBITRARY ZERO, sec
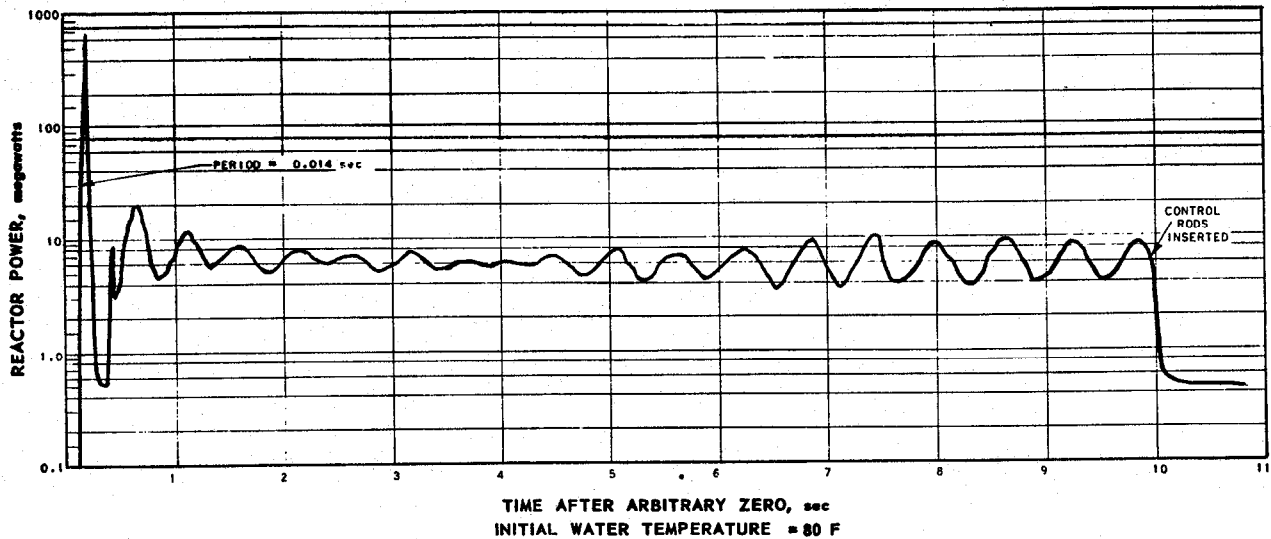INITIAL WATER TEMPERATURE = 80 F

Figure 11. Reactor power variation during 10-second run following initial excursion of 14-millisecond period

sions other plates were responsible for the first formation of steam, since power began to decrease before plate temperature reached saturation.

Effects of Several Variables on Power, Energy and Fuel Plate Temperature

Both the total nuclear energy liberation and the maximum fuel plate temperature reached during a power excursion depend upon the amount of excess reactivity involved in the excursion. In the following presentation, the reciprocal of the stable reactor period is used to characterize the excess reactivity. The relation between the two is given in Fig. 3. In Fig. 13 the total energy† liberated by the power

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† Throughout the section on solid fuel reactors the term energy will be used to refer to the prompt fission energy which is converted to heat in the fuel plates. The total energy release, including delayed emission and energy converted to heat directly in the reactor water, is about 15% higher. A consistent definition is used for reactor power. By the total energy of a power excursion is meant the energy liberation up to the time of the first minimum in reactor power.

excursion and the maximum fuel plate surface temperature are plotted as functions of the reciprocal period for the case in which the reactor water was at saturation temperature before the excursion began. The shapes of the two curves are quite similar; in fact, the maximum fuel plate temperature rise is roughly proportional to the energy of the excursion for all periods shorter than about 0.03 sec, and the peak temperature corresponds to the temporary storage in the fuel plate of 60 to 70 per cent of the total energy of the excursion.

The energy liberation and maximum fuel plate temperature for the condition in which the reactor was initially at room temperature are given, for reactor I, in Fig. 14. The plotted temperature is the maximum above saturation temperature at atmospheric pressure rather than the total temperature rise. The fuel plate temperature is somewhat higher (and the energy release is much higher) for a given reactor period than in the case of saturated reactor water.

PERIOD = 0.134 sec.
INITIAL TEMPERATURE = 80°F

a

PERIOD = 0.021 sec
INITIAL TEMPERATURE = 82°F
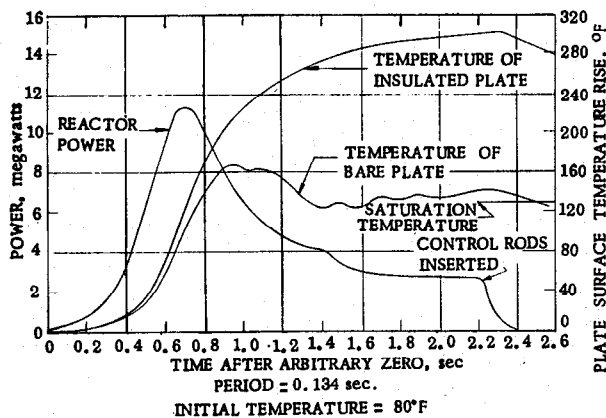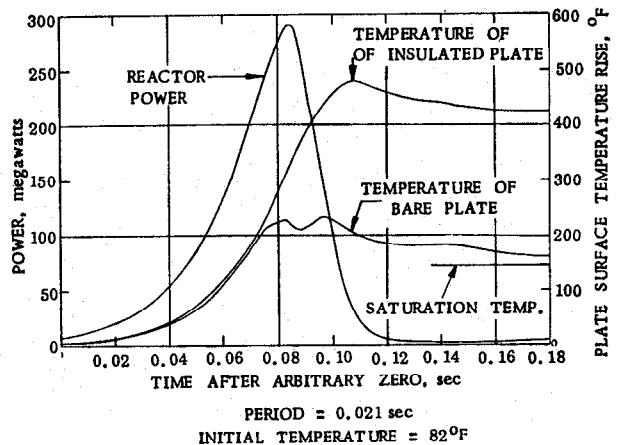
b

Figure 12. Power and fuel plate surface temperature rise during power excursions

Figure 13. Maximum temperature rise of fuel plate surface and total energy release during power excursions of various exponential periods. Reactor I, at saturation temperature and atmospheric pressure
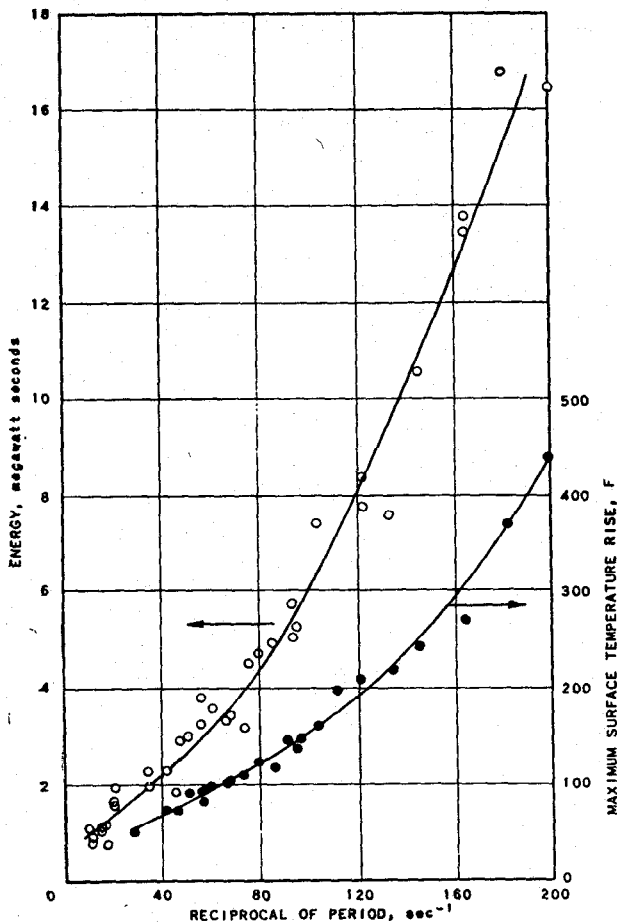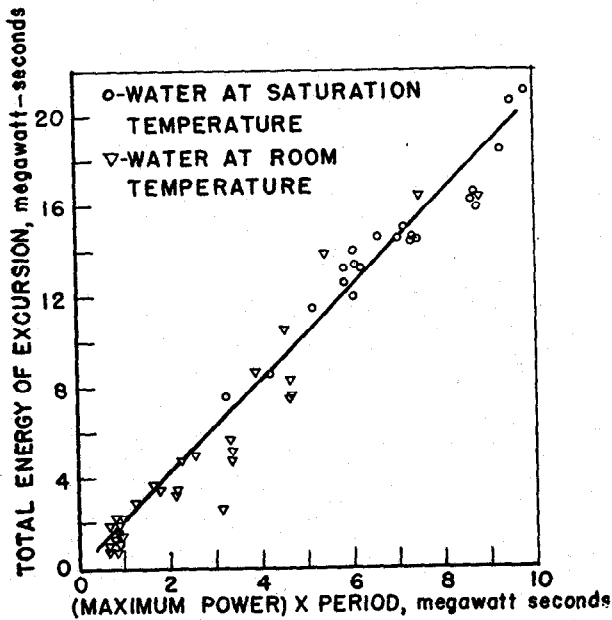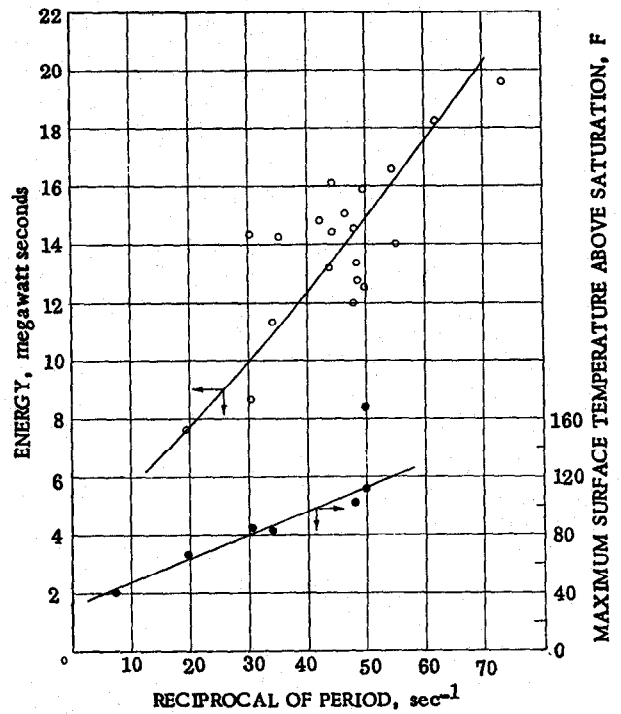

Figure 14. Maximum temperature of fuel plate surface and total energy release during power excursions of various exponential periods. Reactor I, at room temperature (approx. 80°F) and atmospheric pressure

Figure 15. Relation of total energy, maximum power, and period. Reactor I

For the cases of both saturation temperature and room temperature water the total energy liberated during the power excursion was nearly proportional to the product of the maximum power and the period. These quantities are plotted in Fig. 15 for the experiments made with reactor I. The slope of the line is about two.

In comparing the behavior of the two reactors, I and II, which differed in core size, it is more informative to compare energy density or energy per fuel plate than to compare total energy. The latter comparison has been made. In Fig. 16 the energy release per fuel plate is compared, as a function of reciprocal period, for the two reactors: Fig. 16a is for the case of saturated reactor water, and Fig. 16b for the case of room temperature water. The dashed portion of Fig. 16b is an upper limit only. The plotted energy release is that of the fuel plate in the position of highest neutron flux. In comparing the behavior of the two reactors, reference should be made to Table I. Note that the volume of water associated with each fuel plate is about twice as great for reactor II as for reactor I. If the energy release per unit volume of water is compared, the values for reactor II are only slightly higher than those for reactor I, despite the fact that the steam coefficient of reactivity is more than twice as great for reactor I as for reactor II.

In reactor II the investigation of power transients was extended to reactor pressures as high as 300 psi. The pressurization of the reactor was by the vapor from the reactor water. Consequently, only the satu-
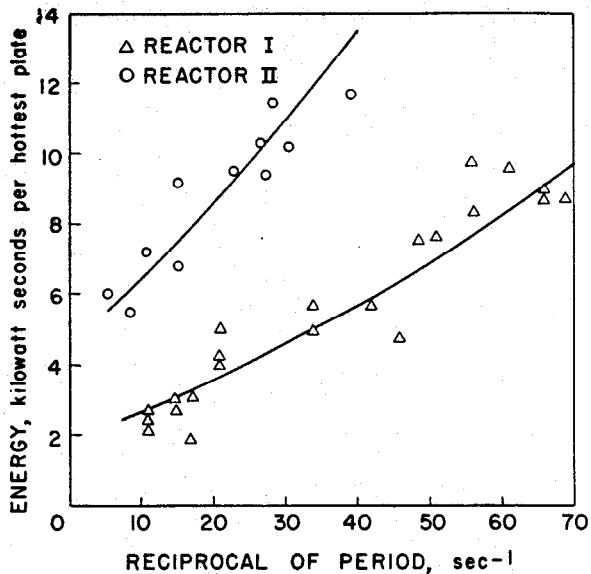
REACTOR WATER AT SATURATION TEMPERATURE
AND ATMOSPHERIC PRESSURE
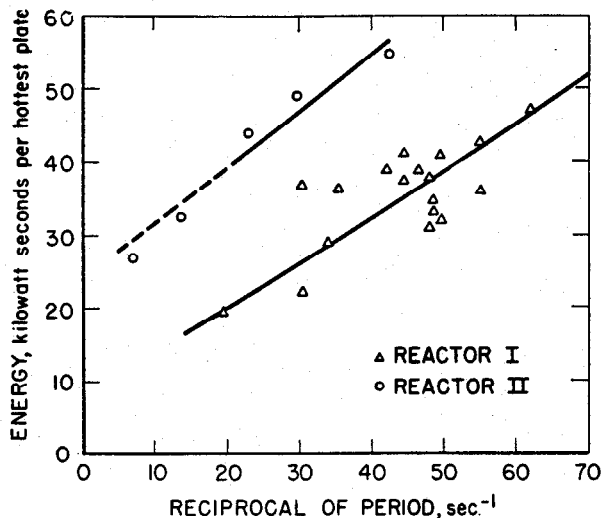
REACTOR WATER AT ROOM TEMPERATURE
AND ATMOSPHERIC PRESSURE

Figure 16. Energy release of hottest fuel plate during power excursions in reactors I and II; (a) left, reactor water at saturation temperature and atmospheric pressure; (b) right, reactor water at room temperature and atmospheric pressure

rated condition could be investigated. The power excursions were run with the reactor tank completely closed. In no case did the pressure in the steam space above the reactor water rise by more than about 5 psi as a result of an excursion. The effect of pressurization is to decrease both the energy released in an excursion of given period and the maximum temperature rise of the fuel plates (Fig. 17).
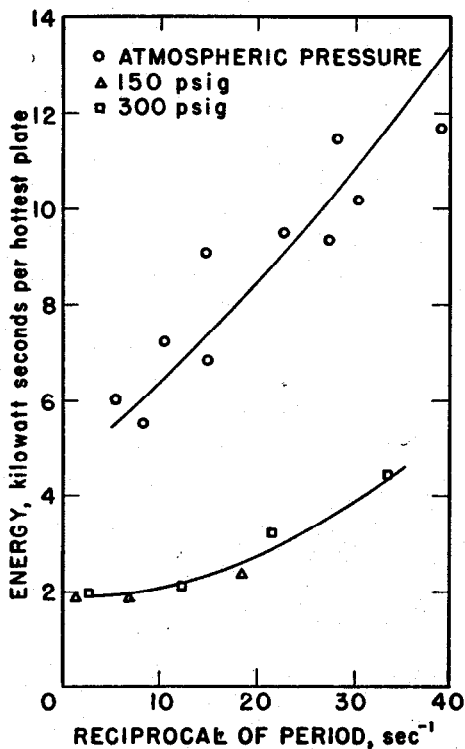
WATER AT SATURATION TEMPERATURE

REACTOR II
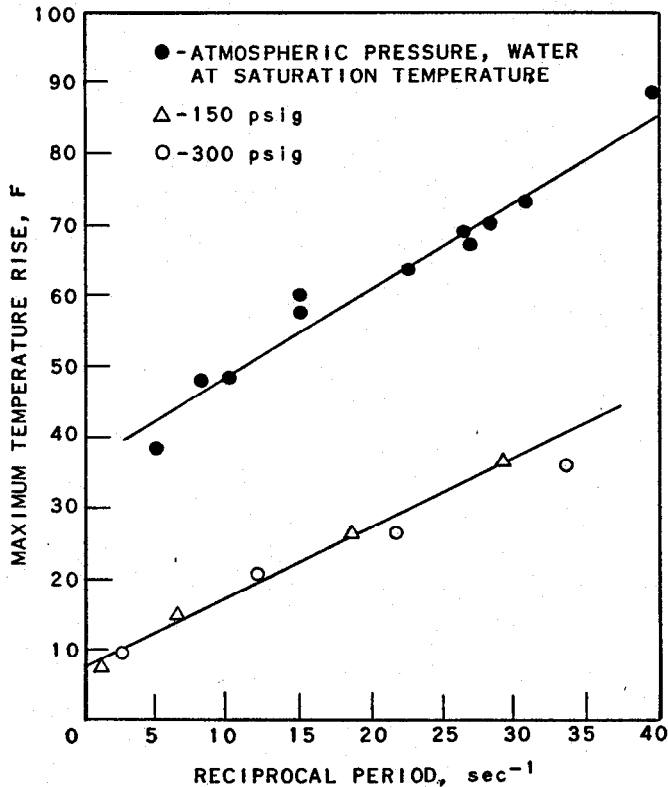
REACTOR II

Figure 17. a, left, energy release of hottest fuel plate during power excursions at different pressures; b, right, maximum fuel plate temperature rise during power excursions: Comparison of unpressurized and pressurized cases

The Destructive Experiment

In the short period experiments with reactor I at atmospheric pressure, the steam pressure which built up in forcing the water rapidly from the reactor resulted in permanent deformation of the fuel plates. Because of this effect it was not possible to extend the experiments to periods shorter than about 0.005 sec without damaging the reactor to the point where it became unusable. Despite this mechanical damage the maximum temperatures reached by the fuel plates did not approach the melting temperature. It was decided that the reactor, which by this time had fulfilled its other purposes, should be sacrificed in an experiment which was violent enough to melt the fuel plates. For this purpose a control rod worth 4% k_{eff} was completely ejected from the reactor core. To increase the severity of the experiment it was run with the reactor water at room temperature. Although the ejection of the rod required only about 0.2 sec, the rod was only about 80 per cent out of the core when the reactor power reached its peak value. The minimum period resulting from the ejection was 0.0026 sec.

The power excursion melted most of the fuel plates. The pressure resulting from the molten metal in contact with the reactor water burst the reactor tank and ejected most of the contents of the shield tank into the air. The sound of the explosion at the control station, half a mile away, was comparable to that resulting from the explosion of 1 to 2 pounds of 40 per cent dynamite on the bare ground at the same distance. Figure 1b, taken from motion picture records of the experiment, shows one stage of the explosion, as compared to the nondestructive ejection of water from the reactor when the period of the excursion was 0.005 sec.

The total energy release during the excursion as determined by calibrated cobalt foils in the reactor core, was 135 megawatt seconds. Other data taken during the excursion are less reliable because of the violent mechanical effects of the explosion. Figure 18 is a reproduction of the transient record. The absolute values on the power curve (A) may be in error by 30 or 40 per cent, and the shape of the decreasing portion may not be correct. The fuel-plate thermocouple (curve B) failed long before peak power was reached. Curve C is a temperature record from an insulated fuel plate installed some distance outside the reactor core. It was used for an auxiliary determination of total energy and does not give good transient information, as it was connected to a relatively slow recorder. A pressure transducer, which was installed in the reactor tank adjacent to the reactor core, failed before it recorded a pressure of significant magnitude (curve D). Analysis of the mechanical damage of the transducer, however, indicated that the peak pressure was at least as high as 6000 psi, and was probably higher than 10,000 psi.

It was evident from examination of the reactor debris that many of the fuel plates had been prac-
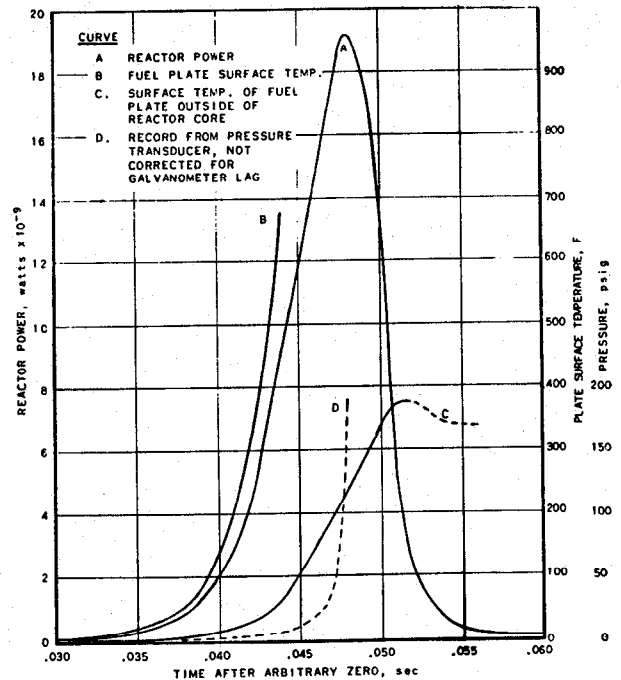

Figure 18. Replot of records from destructive experiment

tically completely melted. Others, evidently those at the edge of the core, had been only partly melted, and portions of them remained fastened to the side plates of the fuel elements (Fig. 19). Some of the fragments which had evidently been molten appeared as spongy metallic globules (Fig. 20). Other fragments appeared to have been molten inside, while the outside clad remained solid (Fig. 21).

Most of the heavy debris fell to the ground near the shield pit. The control rod drive mechanism, which weighed about a ton, fell on the side of the earth shield, after having been thrown about 30 feet into the air. Recognizable fuel plate fragments were thrown as far as 200 feet from the reactor site. Surveys of the total fission-product radioactivity of all the debris indicated that practically all of the fuel originally in the reactor could be accounted for within a radius of 350 feet around the original reactor location. Although these measurements necessarily lacked precision, they showed that no large fraction of the reactor core material left the site in the form of airborne material. At the time of the experiment, the wind velocity was 8 miles per hour at ground level, and 20 miles per hour at 250 feet above ground. Fifteen minutes after the experiment the total beta plus gamma activity level, 3 feet above ground, at a point 0.8 mile directly downwind of the reactor, was 5 mr/hr. At all points farther from the reactor the effects of fall-out were less than this value. Momentarily during the explosion, a gamma dose rate in excess of 400 mr/hr was indicated on a survey meter half a mile from the reactor. This indication decayed rapidly; the total dose received at the half-mile point (cross wind) was less than 10 mr.

Both the observed radiation intensities and the mechanical damage were roughly consistent with the measured nuclear energy release of 135 megawatt seconds. Although the explosion was spectacular, its effects were comparable to those which could be caused by a moderate amount of chemical explosive. The destruction of the reactor tank was not surprising, since it was constructed of relatively thin (½-inch) steel. Most of the equipment outside the shield tank was either undamaged or repairable, and much of it, including the control rod drive mechanism, was decontaminated, reconditioned, and re-used on reactor II.

There was no evidence that the power-limitation process in the destructive experiment differed qual-
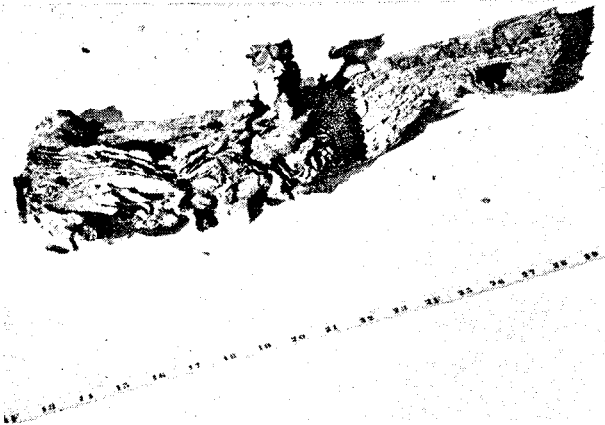

Figure 19. Fuel element side plate with attached cluster of fuel plate fragments
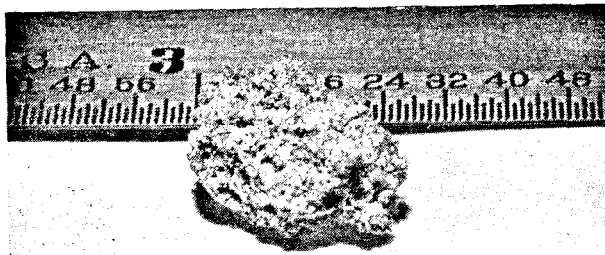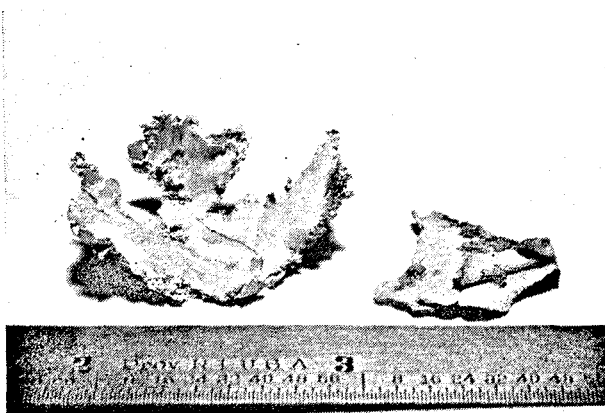

Figure 20. Pellet of spongy aluminum-uranium mixture

Figure 21. Fragments of fuel plates

RECIPROCAL OF REACTOR PERIOD, sec⁻¹
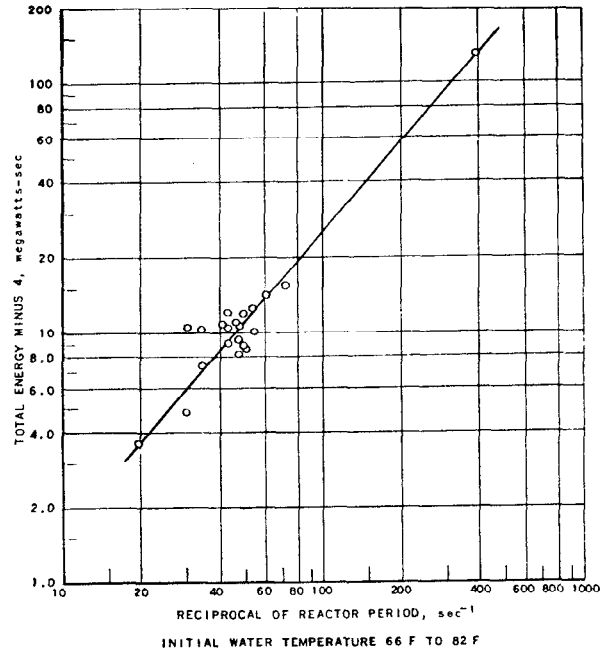
INITIAL WATER TEMPERATURE 66 F TO 82 F

Figure 22. Total energy of excursion minus energy required (4 Mw-Sec) to raise temperature of center of average plate to the boiling point

itatively from that which was effective in the earlier, nondestructive experiments. It was quite evident that the nuclear power release was terminated at an early stage of the explosion; indeed, high-speed motion pictures recorded the light flash emitted by the reactor as it reached high power and showed that it was extinguished before any ejected material appeared above the top of the shield tank. The flash lasted about 0.003 second. The energy stored in the fuel plates as sensible heat and latent heat of fusion during the relatively short nuclear power burst was, of course, released during the much longer explosion process.

Figure 22 is a plot of energy released as a function of reciprocal period for all the power excursions made with cold water in reactor I. In plotting the curve, a constant energy of 4 megawatt-seconds has been subtracted from each value. This is the quantity of energy which has been released up to the time when the center temperature of the average fuel plate has reached the atmospheric boiling point of water. The highest point on the curve, which refers to the destructive experiment, does not appear inconsistent with the other values.

Discussion

The experiments that have been described prove that the reactors investigated possess a high degree of inherent safety, and indicate that it is possible to design practical reactors of these types which will be safe against any reactivity accident which can occur in practice. In this connection, the consistent and regular behavior of the reactors during the experiments is reassuring. Although such behavior would be expected in the case of the homogeneous

non-boiling reactor, anomalies would not have been surprising in the cases where shutdown depends on the rapid formation of steam. Actually, in the entire series of some two hundred experimental runaways, no inconsistencies of behavior were observed which could not be attributed to instrumental errors resulting from rather difficult experimental conditions.

A few general remarks may be made in connection with the application of the results to the estimation of the runaway behavior of other reactors. The behavior of the homogeneous, non-boiling reactor may be considered to be typical of the case in which the loss of reactivity is roughly proportional to the time integral of the transient reactor power. Thus the behavior of power as limited by a negative "prompt"‡ temperature coefficient in a solid-fuel reactor would be expected to be qualitatively quite similar. Quantitatively, of course, such "prompt" coefficients are generally small and their effectiveness is limited. In inferring the behavior of other steam-limited reactors from the results presented here, it should be remembered that the reactors used had fuel plates of such high thermal conductance that in all the non-destructive experiments the metal of the plate represented no important impedance to the transfer of heat to the water. Certainly the transient behavior will be strongly modified and the effectiveness of steam limitation of power will be decreased if a departure is made from this condition, as it may well be in certain power reactor designs. In such a case the difference between the behavior of the reactor with saturated water and with subcooled water may be a very important one, and reactors designed for operation as boiling reactors may have important safety advantages.

No complete theoretical treatment of the transient limitation of power by steam has yet been developed. Attention should be called to some of the general experimental results which will have important bearing on the formulation of such a theory. Perhaps the most obvious is that there is no apparent simple relation between the energy liberated during a transient and the energy content of a steam volume of the size necessary to remove the applied excess reactivity. For example, the heat of vaporization of sufficient steam to fill the entire core volume of reactor I at atmospheric pressure was only 0.087 megawatt-second, very much less than the energy generation in any of the experiments. Furthermore, in extending the experiments from reactor I to reactor II, the total energy release did not increase in proportion to the volume of steam required to produce a given decrease of reactivity. Finally, when the experiments were extended from atmospheric pressure to 300 psig (Fig. 17) the energy release for a transient of given period *decreased* by almost

a factor of 3, whereas the energy content of unit volume of steam *increased* by a factor of 16.

Certainly one of the important considerations is that although the energy required to vaporize a significant volume of steam is small, the temperature differences required to transfer this heat to the water in the short time available during the transient may be large. The heat capacity of all the fuel plates of reactor I was 0.05 megawatt-second per degree F, and the heat capacity of all the water in the core was 0.15 megawatt-second per degree F. Thus the establishment of steep temperature gradients required the expenditure of significant quantities of energy. A further consideration is that relatively high steam pressures must be built up to expel the water from the reactor core rapidly enough to terminate the short-period transients. A few pressure measurements were made in the reactor core which indicated that the peak transient pressure increase was about 15 psi during a transient of period 0.034 sec, and of the order 100 psi during the transient of period 0.005 sec. Consequently, even when transients are run with the reactor water at the ambient saturation temperature, the water is effectively in the subcooled state during the power excursion. Not only must the fuel plates be heated to temperatures corresponding to the transient saturation condition, but steam, once it forms at the hot fuel plate surface, may recondense in the cooler water.

Approximate theoretical treatments of the steam transient have been made by various workers to extend the results of the experiments to other reactor designs. Golian *et al.*§ have assumed that steam is formed in a laminar layer immediately adjacent to the fuel plate, and that the thickness of the layer grows by conductive transfer of energy across the layer to the water boundary. Edlund and Noderer¶ employ a model in which the transient pressure rise plays an important part. It is assumed that the heat transferred to the water is contained in a layer adjacent to the fuel plate, the average temperature of which equals the transient saturation temperature. The layer, which contains water and steam, grows in thickness as though by a thermal conductive process, but with a thermal diffusivity determined empirically from the experiments. Both approaches have shown reasonable agreement with the results of the experiments on reactor I.

The semi-empirical approach to extension of the experimental data is facilitated by the circumstance that all of the experimental power transients have quite similar shapes if time is measured in the non-dimensional unit of asymptotic reactor period. This characteristic is illustrated by Fig. 15. Despite this regularity, however, detailed examination of the power curves reveals significant differences in behavior as the conditions of the experiment are changed. In Fig. 23 the power curves for three

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‡ The "prompt" coefficient is that component of the temperature coefficient of reactivity which depends on fuel-element temperature alone. It results from thermal expansion of the fuel element and Doppler broadening of resonances.

§ US Naval Research Laboratory.
¶ Oak Ridge National Laboratory.

TIME AFTER ARBITRARY ZERO, sec

CURVE a – POWER GENERATION IN HOTTEST
FUEL PLATE.
CURVE b – SURFACE TEMPERATURE OF HOTTEST
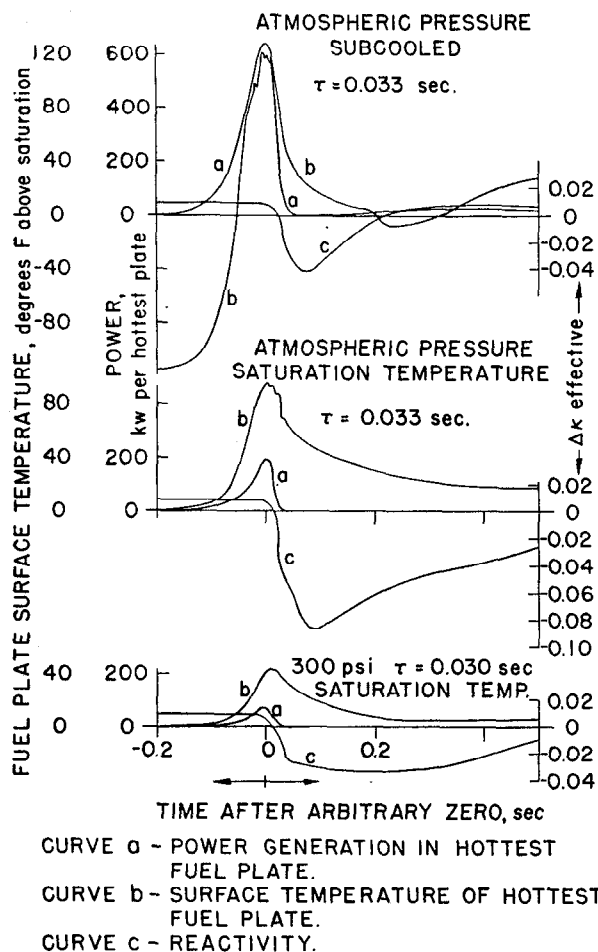FUEL PLATE.
CURVE c – REACTIVITY.

Figure 23. Time variation of power, fuel plate temperature, and
reactivity during power excursions on reactor II

different reactor II transients of about the same period are reproduced along with the fuel-plate temperature records. The power curves have been analyzed to yield the variation of reactivity with time. The differences in magnitude of the power and temperature variations with subcooling and with pressure are striking, but it is interesting to note also the reactivity variations. At atmospheric pressure and saturation temperature the reactor shuts itself down by more than 8% k_{eff}. In the subcooled condition the degree of subcriticality achieved is considerably less, and the recovery to criticality is

rapid, no doubt because of condensation of the steam. At 300 psi the degree of shutdown is again small, evidently because of the relatively low heat storage in the fuel plates. The recovery to criticality is, however, quite slow. A striking characteristic in all cases is the small fraction of the total reactivity change which suffices to stop the initial power rise. As is to be expected theoretically, this is equal to the initial prompt excess reactivity.

The limitation of power in fast transients is one aspect of the self-regulating behavior of reactors with strong negative power coefficients of reactivity. Experience with the solid-fuel boiling reactors (e.g., Fig. 10) has shown that the self-regulation is normally stable but that a type of instability (chugging) can result if the reactor is subjected to sufficiently large reactivity variations. Fortunately, even under chugging conditions the power is limited and does not reach a dangerous level. It is no doubt true for all self-regulated reactors that for some amplitude of reactivity excitation the characteristic rates of change of power are too rapid for the self-regulating process to maintain control, and instability will result. In the boiling reactors it is believed that the rate of escape of steam from the core, rather than the rate of formation of steam, represents the limiting time constant in the regulating process. Consequently, in subcooled boiling operation, where steam can leave by condensation, the range of stability is increased (Fig. 11). In any case, however, the range of stable self-regulation of the reactors investigated is more than adequate for practical use and probably exceeds that which can be attained practically with artificial control systems.

REFERENCES

1. Untermyer, Samuel, Direct Steam Generation for Power, Nucleonics, Vol. 12 No. 7, 43 (July, 1954).

2. Lottes, P. A., Boiling Studies at Argonne Relative to Boiling Reactors, Proceedings, 1955 Conference on Nuclear Engineering, University of California, Los Angeles; California Book Co., Ltd., Berkeley, California.

3. Kasten, Paul R., Homogeneous Reactor Safety, Proceedings, 1955 Conference on Nuclear Engineering, University of California, Los Angeles; California Book Co., Ltd., Berkeley, California.

4. King, L. D. P., The Los Alamos Homogenous Reactor, Supo Model, LADC-1081 (Feb. 2, 1952).